Name: Answer Key **Total Points:** 100

Emp. ID: _____ **Points Received:** _____

Date: 10/01/2005 **Grade:** _____

Exam / Quiz Title: OCT05NRC : October 2005 NRC Exam

Open Reference [ ] Yes [X] No

Reference Material that may be used to support this open reference exam:

1. EOP/SAG Graphs 2. EOP 2A, 5A 3. 2.0.5 4. 5.7.1 5. T.S. 3.1.7
6. T.S. 3.4.1 7. T.S. 3.5.1 8. T.S. 3.8.1 9. TRM T3.4.1 10. 2.4VAC Att 3
11. 2.4SDC Att 5

GUIDELINES

1. Allotted time to complete the exam / quiz is **6** Hrs.

2. ALL questions shall be directed to the proctor. Students *shall not* discuss the questions among themselves until all examinees have completed the exam / quiz.

3. Restroom trips are limited and only one examinee at a time may leave the room.

4. Verify that all questions have been answered prior to turning in your exam/quiz.

5. To pass the exam/quiz, you must achieve an overall grade of **80%** or greater.

6. After you have completed the exam/quiz, you must sign the statement on the cover sheet indicating that the work is your own and you have not received or given assistance in completing the exam / quiz. This must be done after you have finished.

I have neither given nor received assistance during the administration of this examination/quiz. (Proctor assistance excluded) All work on this exam is my own.

Examinee Signature: _____

Prepared by: _____ First Grader: _____

Approved by: _____ Second Grader: _____
(If required)

Approved by: _____
(If required)

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
1	21300	00	08/07/2005	10/01/2005	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
Abnormal/Emergency Procedures	INT0320124, Operational implication operation in Natural circulation when power is greater than 1%.

Related Lessons
INT0320124 CNS Abnormal Procedure (RO) Reactor Recirculation

Related Objectives	
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	
(B)(10)	Administrative, normal, abnormal, and emergency operating procedures for the facility.
(B)(7)	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Related Skills (K/A)	
295001.AK1.01	Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: (CFR: 41.8 to 41.10) Natural circulation. (3.5/3.6)

QUESTION: 1 21300 (point(s))

Given the following conditions during a startup:

- ➢ Reactor power is 20%.
- ➢ Reactor Recirculation pump "A" is idle.
- ➢ Reactor Recirculation pump "A" is ready to start.
- ➢ Bus 1D is supplied via the 1DS breaker.

Annunciator **RRMG B BKR 1DS TRIP**, 9-4-3/A-5 alarms.

What action is required next?

a. Start Recirculation pump "A".

b. Scram and enter procedure 2.1.5.

c. Perform rapid shutdown per procedure 2.1.4.1.

d. Inform Reactor Engineering that Stability Exclusion Region has been entered.

ANSWER: 1 21300

b. Scram and enter procedure 2.1.5.

The trip of the 1DS places the plant in natural circulation. A reactor scram is required if power is greater than 1% and in natural circulation. Immediate operator action of 2.4RR states " If both RR pumps are tripped and reactor power > 1% rated thermal. SCRAM **and** Enter Procedure 2.1.5."

Although determining the required action is cognitive level 1 determining that the recirculation pump trip places the plant in natural circulation implies understanding (cognitive level 2).

Distractors:

a. is incorrect as no procedure allows the restart of a RR pump in NC at power.

c. is incorrect as a reactor scram is required.

d. is incorrect because even though this would be an advisable action the question asked for the next required action. The next action should be to scram.

Source: Modified 13406

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
2	2256	01	08/12/2004	10/01/2005	Licensed Operator	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	COR0010102, Degraded Voltage effect on 1F/1G and reactor scram or not

Related Lessons
COR0010102 AC Electrical Distribution

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

Related References	
5.3GRID	Degraded Grid Voltage
(B)(8)	Components, capacity, and functions of emergency systems.

Related Skills (K/A)	
295003.AK3.05	Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: (CFR: 41.5 / 45.6) Reactor SCRAM. (3.7/3.7)

QUESTION: 2 2256 (1 point(s))

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

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

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

a. Diesel Generators and the reactor is scrammed.

b. Diesel Generators and the reactor is **NOT** scrammed.

c. Emergency Transformer and the reactor is scrammed.

d. Emergency Transformer and the reactor is **NOT** scrammed.

ANSWER: 2 2256

a. Diesel Generators and the reactor is scrammed.

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

b. is incorrect because the the reactor is scrammed.

c. is incorrect because the DG is supplying.

d. is incorrect because the DG is supplying and the reactor is scrammed.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
3	21302	00	07/23/2005	10/01/2005	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	COR0021202, What is the reason a reactor scram occurs when this loss of DC power happens.

Related Lessons
COR0021202 INTERMEDIATE RANGE MONITOR

Related Objectives	
COR0021202001030B	Describe the interrelationships between IRM subsystem and the following: Reactor protection system
COR0021202001030C	Describe the interrelationships between IRM subsystem and the following: Operating the mode switch
COR0021202001030E	Describe the interrelationships between IRM subsystem and the following: APRM (including scram signal)
COR0021202001050B	Describe the IRM system design features and/or interlocks that provide the following: Reactor scram signals
COR0021202001070B	Predict the consequences of a loss or malfunction of the following would have on the IRM system: 4/48 VDC
COR0021202001090B	Given plant conditions, determine if the following IRM actions should occur: Reactor Scram.

Related References	
(B)(6)	Design, components, and functions of reactivity control mechanisms and instrumentation.
(B)(7)	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Related Skills (K/A)	
295004.AK3.03	Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER: (CFR: 41.5 / 45.6) Reactor SCRAM: Plant-Specific (3.1/3.5)

QUESTION: 3 21302 (1 point(s))

Given the following conditions:

- ➢ Reactor power is currently 6%.
- ➢ The **Reactor Mode Switch** is in **RUN**.
- ➢ IRMs B and D are failed upscale.
- ➢ IRM B is bypassed.
- ➢ **24/48V DIV 1** is now lost.

What additional action or condition results in a **full reactor scram and why**? (Use Technical Specification values.)

a. Loss of MCC-L causing the loss of RPS MG set A.

b. Reactor power rises above 15% causing an APRM upscale trip.

c. Loss of 24/48V DIV 2 causing the INOP trip on the B side IRMs.

d. When reactor power is lowered below 3% due to the APRM/IRM companion trip.

ANSWER: 3 21302

d. When reactor power is lowered below 3% due to the APRM/IRM companion trip.

The failure of IRMs B and D result in an upscale trip for IRM B and D. Since the Reactor Mode switch is in RUN the IRM upscales do not generate a trip. Only B IRM is bypassed (it is only possible to bypass one or the other). Add to this now the loss of Division 1 24/48 VDC that causes INOP on the Division 1 IRMs. Since the mode switch is in RUN a scram is not generated by this loss as the IRM inop is bypassed in run. When power is lowered to 3% the APRM downscales come in and now IRM companions are either upscale or inop on both sides so a scram results.

Distractors:

a. is incorrect because no B side half scram is present.,

b. is incorrect because the mode switch is in RUN.

c. is incorrect because this would only result in a scram if the mode switch were in position other than RUN.

Question Source: Modified 21065

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
4	12134	01	03/12/2002	10/01/2005	None	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	INT0320127O0O0100 CNS Abnormal Procedures (RO) Turbine/Generator

Related Lessons	
INT0320127	CNS Abnormal Procedures (RO) Turbine/Generator

Related Objectives	
INT0320127O0O0100	Given plant condition(s), determine from memory if a manual reactor scram or an emergency shutdown from power is required due to the event(s).

Related References	
(B)(10)	Administrative, normal, abnormal, and emergency operating procedures for the facility.
2.4TURB	Main Turbine Abnormal

Related Skills (K/A)	
2.4.49	Ability to perform without reference to procedures those actions that require immediate operation of system components and controls. (CFR: 41.10 / 43.2 / 45.6) (4.0/4.0)

QUESTION: 4 12134 (1 point(s))

Reactor power is 90% when rising thrust bearing metal temperatures were noted. Thrust bearing metal temperature on computer point **T079 was 220°F and T080 is 230°F.** The crew also noted that lube oil cooler outlet temperature was **145°F.**

Which action is **required next**?

a. Reduce turbine load by 10%.

b. Commence a normal shutdown.

c. Scram the reactor and trip the turbine.

d. Reduce lube oil cooler outlet temperature.

ANSWER: 4 12134

c. Scram the reactor and trip the turbine.

Turbine thrust bearing metal temperature is greater than 225°F and at 90% power reactor power is above the setpoint for 9-5-2/C-4, TSV & TCV CLOSURE TRIP BYP annunciator a reactor scram and then a turbine trip are required.

a. is incorrect because a turbine trip is required.
b. is incorrect because a turbine trip is required.
d. is incorrect because a turbine trip is required.

Source: Direct

K/A: 2.4.49 as applied to 295005.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
5	13673	01	06/13/2003	10/01/2005	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	INT032-01-04, CNS Administrative Procedures and 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	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: 5 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: 5 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

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
6	21295	00	07/14/2005	10/01/2005	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	INT0320134, Why is the reactor scrammed after control room is abandoned?

Related Lessons
COR0023402 Alternate Shutdown (LO)

Related Objectives	
COR0023402001050B	Predict the consequences a malfunction of the following would have on the Alternate Shutdown System: AC Distribution system

Related References	
5.1ASD	Shutdown From Outside The Control Room
(B)(10)	Administrative, normal, abnormal, and emergency operating procedures for the facility.
5.4FIRE-SD	Fire Induced Shutdown From Outside Control Room

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

QUESTION: 6 21295 (1 point(s))

Given the following conditions:

- ➢ The Plant is operating at 100% power.
- ➢ Two (2) people in the Control Room are overcome by fumes of unknown origin.
- ➢ The Control Room is abandoned before the reactor can be scrammed.

Why is the reactor scrammed after abandonment?

a. Immediately reduce the staffing requirements.

b. In order to also generate Group 1, 2, 3, 6, and 7 isolations.

c. So that the plant can be taken to cold shutdown (Mode 4).

d. To place the plant in an **immediately** more stable condition.

ANSWER: 6 21295

c. So that the plant can be taken to cold shutdown (Mode 4).

Since the control is abandoned operation cannot continue. Once the control room is abandoned the goal is to put the unit in Mode 4 (cold shutdown). This cannot be accomplished without first scramming the reactor.

Reference: 5.1ASD, 5.4FIRE-SD

10CFR55.41(b)(10)

Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
7	9033	01	05/03/2002	10/01/2005	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	COR0021902, REC, LOOP - which pump will be running

Related Lessons
COR0021902 REACTOR EQUIPMENT COOLING

Related Objectives	
COR0021902001030B	Describe the interrelationship between the REC System and the following: REC Pumps
COR0021902001060D	Given a specific REC malfunction, determine the effect on any of the following: Standby REC pump operation
COR0021902001110B	Given plant conditions, determine if any of the following should occur: Standby pumps automatic start

Related References	
COR0021902	REC
2.3_M-1	Panel M - Annunciator M-1
5.2REC	Loss Of REC
(B)(4)	Secondary coolant and auxiliary systems that affect the facility.

Related Skills (K/A)	
295018.AK3.04	Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: (CFR: 41.5 / 45.6) Starting standby pump. (3.3/3.3)

QUESTION: 7 9033 (1 point(s))

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

- "A", "B", and "C" pumps running in **NORMAL**.
- "D" secured in **STANDBY.**
- Both REC heat exchangers are in service.
- A loss of **ALL** off-site power occurs.

What is the REC system line up after one (1) minute?

a. REC pump "D" is running supplying critical loads **only**.

b. REC pump "D" is running supplying loads from the B heat exchanger **only**.

c. REC pumps "A", "B", and "C" running and are supplying the critical loops **only**.

d. REC pumps "A", "B", and "C" running and are supplying **both** heat exchangers.

ANSWER: 7 9033

a. REC pump D is running supplying critical loads **only**.

The D pump selector switch was in STANDBY, therefore it will start when emergency power becomes available. Only the critical loops will be supplied because of the low header pressure isolations.

Distractors:

b. Pump D will be running but both heat exchanger isolation valves will have isolated on low header pressure.
c. Pumps A, B, and C will not re-start.
d. Pumps A, B, and C will not re-start.

Source: Direct

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
8	2226	00	08/13/1999	10/01/2005	Licensed Operator	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
Abnormal/Emergency Procedures	INT032013 What requires entry into the emergency procedure.

Related Lessons
INT0320136 CNS Abnormal Procedures (RO) Miscellaneous

Related Objectives	
INT0320136L0L0100	Given plant condition(s), determine from memory the appropriate Abnormal/Emergency Procedure(s) to be utilized to mitigate the event(s).

Related References	
5.2AIR (B)(10)	Loss of Instrument Air Administrative, normal, abnormal, and emergency operating procedures for the facility.

Related Skills (K/A)	
2.4.4	Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures. (CFR: 41.10 / 43.2 / 45.6) (4.0/4.3)
LINK ONLY TO EOP/AOP LESSONS/QUESTIONS	

QUESTION: 8 2226 (1 point(s))

The plant is operating at power and a power ascension is in progress.

What condition, by itself, requires **direct** entry into EP 5.2AIR, Loss of Instrument Air?

a. SERVICE AIR PRESSURE falls just below the green band.

b. When the first Control Rod drift alarm annunciates following a loss of air.

c. Annunciator A-4/C-4, STATION AIR COMPRESSOR TROUBLE A A-4/C-5 alarms.

d. When the first Control Rod ACCUM light illuminates on the Full Core Display following a loss of air.

ANSWER: 8 2226

a. SERVICE AIR PRESSURE below green band.

The entry conditions for 5.2AIR are either INSTRUMENT AIR PRESSURE below green band or SERVICE AIR PRESSURE below green band.

Distractors:

b. is incorrect because this is not an entry condition.

c. is incorrect because this is not an entry condition.

d. is incorrect because this is not an entry condition.

Source: Direct

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
9	21303	00	08/13/2005	10/01/2005	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
Abnormal/Emergency Procedures	INT0320126, Ability to operate or monitor Reactor Recirculation system during a loss of SDC.

Related Lessons
ACD0070307 Thermal Hydraulics (GP)

Related Objectives
ACD00703070013600 Describe means by which the operator can enhance natural circulation.

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

Related Skills (K/A)	
295021.AA1.05	Ability to operate and/or monitor the following as they apply to LOSS OF SHUTDOWN COOLING: (CFR: 41.7 / 45.6) Reactor recirculation. (3.0/3.0)

QUESTION: 9 21303 (1 point(s))

The plant is shutdown with "A" Loop of RHR in SDC with "A" RHR pump operating. A rupture occurs between RHR-MO-17 and RHR-MO-18 causing reactor water level to fall to 0". A SDC isolation and trip of “A" RHR pump results. The following plant conditions are noted:

- ➢ Reactor vessel head is installed.
- ➢ Reactor water level is now recovered to 40" and steady.
- ➢ Reactor pressure is 5 psig slowly rising.
- ➢ Both RR pumps are idle.
- ➢ Reactor Head vents are closed.

What actions are required?

a. Raise reactor water level and start a RR pump.

b. Open the head vents and use feed with condensate system.

c. Lineup and start the "B" loop of RHR for decay heat removal.

d. Bleed steam to the condenser until reactor pressure is at 0 psig.

ANSWER: 9 21303

a. Raise reactor water level and start a RR pump.

A loss of SDC has occurred due to a rupture on the common RHR SDC suction line to the Reactor. This makes the other loop of RHR unavailable for SDC. 2.4SDC and TS require the restoration of flow through the reactor. Starting a RR pump and raising level are important and required steps to restoration.

Distractors:

b. is incorrect as the head vents are required to be closed when pressure is >0 psig.
c. is incorrect as the common suction line is inoperable due to a rupture.
d. is incorrect as the condenser is not used until level is raised and the steam lines are plugged.

Source: Modified 16796

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
10	21362	00	09/09/2005	10/01/2005	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	COR0011802 , Relationship between refueling accident and radiation monitoring equipment.

Related Lessons
COR0011802 OPS Radiation Monitoring

Related Objectives	
COR0011802001110D	Predict the consequence of the following items on the Radiation Monitoring System: Refuel floor handling accidents/operations

Related References	
(B)(11)	Purpose and operation of radiation monitoring systems, including alarms and survey equipment.

Related Skills (K/A)
295023.AK2.03 Knowledge of the interrelations between REFUELING ACCIDENTS and the following: (CFR: 41.7 / 45.8) Radiation monitoring equipment (3.4/3.6)

QUESTION: 10 21362 (1 point(s))

Which radiation monitor is used to monitor the release of radioactive material from a refueling accident?

a. Reactor Building Ventilation Rad Monitor.

b. Elevated Release Point (ERP) Rad Monitor.

c. Refueling Floor Area Radiation Monitor (ARM).

d. Refueling Floor Continuous Air Monitor. (CAM).

ANSWER: 10 21362

b. Elevated Release Point (ERP) Rad Monitor.

The accident results in a group 6 isolation and the start of SGT. SGT discharges to the ERP and the ERP Kaman monitors the release.

a. is incorrect because a group 6 isolation would occur.

c. is incorrect because this provides no indication of release only area radiation levels.

d. is incorrect because this provides no process flow information only area airborne.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
11	21305	00	07/23/2005	10/01/2005	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
Emergency Operating Procedures	COR0022302, Relationship between drywell pressure and the torus spray valve.

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

Related References	
(B)(4)	Secondary coolant and auxiliary systems that affect the facility.
(B)(7)	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Related Skills (K/A)	
295024.EK2.13	Knowledge of the interrelations between HIGH DRYWELL PRESSURE and the following: (CFR: 41.7 / 45.8) Suppression pool spray: Plant-Specific. (3.8/3.8)

QUESTION: 11 21305 (1 point(s))

Given the following conditions:

- ➢ A LOCA has occurred.
- ➢ The Residual Heat Removal System is operating in the LPCI Mode.
- ➢ Drywell pressure is **5.0 psig.**
- ➢ The CONTMT COOLING VLV CONTROL PERMISSIVE Switch is placed in **MANUAL.**
- ➢ The CONTMT COOLING 2/3 CORE VALVE CONTROL PERMISSIVE switch is in **OFF.**

What additional condition prevents opening the RHR Torus spray inboard throttle Valves MOV-38A(B)?

a. Drywell pressure falls to 1.7 psig.

b. Loss of 125 VDC Reactor Building starter rack.

c. Reactor water level is -140 inches on Wide Range Instruments.

d. The Suppression Pool Cooling /Torus Spray Outboard Valve MOV-39 is opened first.

ANSWER: 11 21305

a. Drywell pressure falls to 1.7 psig.

EXPLANATION OF ANSWER: a. Correct. If Drywell pressure falls below 2.3 psig, the Spray Valve Control logic will de-energize and the Spray Valves will not open. b. this SR does not provide power to these valves c. Backwards. Level must be < -35" FZ (This corresponds to -193” instrument zero according to Tech Spec Bases 3.3.5.1 page B-3.3-103) to prevent valve operation. d. This switch is spring return to OFF. A contact in the Spray Valve Control logic is closed when the switch is in the OFF AFTER MANUAL position.

REFERENCE: STCOR0022302 Residual Heat Removal Page 18 Section II.F Rev 13; PR 2.2.69 Residual Heat Removal System Page 9-10 Section 4 Rev 57

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
12	21356	00	08/31/2005	10/01/2005	Licensed Operator	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
Abnormal/Emergency Procedures	COR0021602, Determine the effect a high reactor pressure has on water level.

Related Lessons
COR0021602 OPS NUCLEAR PRESSURE RELIEF

Related Objectives	
COR0021602001040D	Given a Nuclear Pressure Relief system component manipulation, predict and explain the changes in the following parameters: Reactor water level
COR0021602001070E	Given a specific NPR malfunction, determine the effect on any of the following: Reactor over-pressurization

Related References	
(B)(5)	Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for

Related Skills (K/A)	
295025.EA2.06	Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE: (CFR: 41.10 / 43.5 / 45.13) Reactor water level. (3.7/3.8)

QUESTION: 12 21356 (1 point(s))

Following a transient and reactor scram the following conditions exist:

- ➢ Reactor water level is 53" (NR) and steady.
- ➢ All SRVs are currently closed.
- ➢ LLS is not available.
- ➢ Reactor pressure is 1050 psig and rising at 5 psi/minute.
- ➢ RCIC is running at capacity and is the ONLY injection source.

How does reactor water level respond as reactor pressure rises above the setpoint for the lowest set relief valve?

Reactor water level immediately…

a. rises, then RCIC trips and water level then falls.

b. rises, then falls and RCIC remains in operation.

c. falls, then rises and RCIC remains in operation.

d. falls until the SRV closes and RCIC remains in operation.

ANSWER: 12 21356

a. rises, then RCIC trips and water level then falls.

Reactor water level is just below the RCIC turbine trip. As reactor pressure rises the lowest set relief valve will lift. This will result in a substantial swell and the trip of the RCIC turbine. After the swell transient is over level falls due to the inventory loss through the now open SRV.

Distractors:

b. is incorrect because level would swell tripping RCIC.

c. is incorrect as the immediate vessel level response is swelling of level.

d. is incorrect as the immediate vessel level response is swelling of level.

Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
13	21306	00	07/23/2005	10/01/2005	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
Emergency Operating Procedures	INT0080618, Interpret SP Level and Reactor Pressure for HCTL.

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

Related Objectives	
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.
INT0080613001040B	State the basis for primary containment control actions as they apply to the following: Primary Containment Control Systems
INT00806180010300	Given plant conditions and the EOP and SAG Graphs Flowchart, determine if operation is within the allowed region of a graph.
INT00806180020400	Using the Cautions provided in the EOP and SAG Flowcharts, explain the bases behind each of the Cautions.

Related References	
(B)(8)	Components, capacity, and functions of emergency systems.
(B)(10)	Administrative, normal, abnormal, and emergency operating procedures for the facility.

Related Skills (K/A)	
295026.EK3.01	Knowledge of the reasons for the following responses as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: (CFR: 41.5 / 45.6) Emergency/normal depressurization. (3.8/4.1)

QUESTION: 13 21306 (1 point(s))

An accident occurred that resulted in operation in the unsafe region of the HCTL curve.

Suppression pool level is 12’ and Suppression pool temperature is 212°F.

IF an Emergency RPV Depressurization is performed at this time, what potential consequence could result?

a. The downcomers may fail due to excessive cyclic stress.

b. The Safety Relief Valve (SRV) tail pipe supports may fail.

c. The Torus-to-Drywell Vacuum Breaker capacity may be exceeded.

d. The Primary Containment Pressure Limit (PCPL) may be exceeded.

ANSWER: 13 21306

d. The Primary Containment Pressure Limit (PCPL) may be exceeded.

Current conditions place the plant on the unsafe side of the HCTL Curve. An ADS initiation now may result in exceeding the PCPL due to insufficient energy absorption capacity to handle a blowdown.

a. is incorrect as the blowdown is directed through the SRVs T quenchers not the downcomers.
b. is incorrect The maximum level where a blowdown would not cause damage to the SRV tail pipe supports is 16 feet.
d. is incorrect Torus to Drywell Vacuum Breakers are designed for LOCA energy release.

Source: Bank (production TM 21460)

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
14	5349	01	07/24/2003	10/01/2005	NRC Style Question	RO: SRO: NLO:	Y Y N

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

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

Related Lessons	
INT0080605	OPS FLOWCHART 1A - RPV CONTROL/RPV PRESSURE
INT0080618	OPS EOP AND SAG GRAPHS AND CAUTIONS
INT0080609	OPS EOP FLOWCHART 1A - RPV CONTROL, RPV LEVEL

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.
INT00806180020400	Using the Cautions provided in the EOP and SAG Flowcharts, explain the bases behind each of the Cautions.
INT00806050011200	Given plant conditions, assess if RPV water level can be determined or not.
INT00806090011300	Given plant conditions, assess if RPV water level can be determined or not.

Related References	
5.8	Emergency Operating Procedures (EOPs)
(B)(8)	Components, capacity, and functions of emergency systems.

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

QUESTION: 14 5349 (1 point(s))

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

- ➢ Reactor pressure is 470 psig (lowering slowly.)
- ➢ Indicated Wide Range Reactor water level is -120" (steady.)
- ➢ Drywell pressure is 5.5 psig (rising slowly.)
- ➢ Drywell temperature is 350° F (all points) (steady.)

What is the status of Wide Range Reactor Level Instrumentation?

Wide Range Reactor Level Instrumentation is . . .

a. accurate **AND** can be used for trending.
Actual Reactor level is -120".

b. **NOT** accurate **BUT** can be used for trending.
Actual Reactor level is <u>higher than</u> -120".

c. **NOT** accurate **BUT** can be used for trending.
Actual Reactor level is <u>lower than</u> -120".

d. **NOT** accurate **AND CANNOT** be used for trending.

ANSWER: 14 5349

c. **NOT** accurate **BUT** can be used for trending.
Actual Reactor level is <u>lower than</u> -120".

Provide EOP Graphs 15A, B, C, D & E

EXPLANATION OF ANSWER: c. is correct. 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. The elevated Drywell temperatures are causing indicated level to be erroneously high and actual Reactor level will be lower than -120".

Foils:
a. Indicated level is erroneously high and actual Reactor level will be lower than 120".
b. Actual Reactor level will be lower than -120".
d. 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.

REFERENCES: EOP Graphs 1 & 15, CAUTION 1
INT0080618 INT0080605

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
15	21296	00	07/14/2005	10/01/2005	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
Emergency Operating Procedures	INT0080618, What torus temperature first makes operation in the unsafe region of HCTL?

Related Lessons
INT0080618 OPS EOP AND SAG GRAPHS AND CAUTIONS

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

Related References	
5.8	Emergency Operating Procedures (EOPs)
(B)(10)	Administrative, normal, abnormal, and emergency operating procedures for the facility.

Related Skills (K/A)	
295030.EA2.02	Ability to determine and/or interpret the following as they apply to LOW SUPPRESSION POOL WATER LEVEL: (CFR: 41.10 / 43.5 / 45.13) Suppression pool temperature (3.9/3.9)

QUESTION: 15 21296 (1 point(s))

Following an accident the following plant conditions exist:

- ➢ RPV pressure is 800 psig and stable.
- ➢ Torus level is 10 ft. and stable.
- ➢ Torus temperature is rising.

What is the first average torus water temperature that requires an Emergency Depressurization?

a. 180°F

b. 195°F

c. 205°F

d. 215°F

ANSWER: 15 21296

b. 195°F

800 psig and 195°F is the first condition that puts operation in the unsafe region of the Heat Capacity Temperature Limit Graph.

Distractors:

a. is incorrect because this is not in the unsafe region of the HCTL Graph and therefore an ED is not required. This answer would be correct if reactor pressure was 1000 psig.

c. is incorrect because this is not the first temperature that requires ED. ED is required at 185°F. This answer would be correct if reactor pressure was 600 psig.

d. is incorrect because this is not the first temperature that requires ED. ED is required at 185°F. This answer would be correct if reactor pressure was 400 psig.

<u>Provide EOP Graph 7 to the candidate.</u>

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
16	21307	00	07/23/2005	10/01/2005	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	INT0080613, What RHR lineup is used to refill the SP?

Related Lessons
COR0022302 RESIDUAL HEAT REMOVAL INT0080613 OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL

Related Objectives
NONE

Related References	
5.8.14	Suppression Pool Make Up Systems
(B)(4)	Secondary coolant and auxiliary systems that affect the facility.
(B)(10)	Administrative, normal, abnormal, and emergency operating procedures for the facility.

Related Skills (K/A)	
295030.EA1.06	Ability to operate and/or monitor the following as they apply to LOW SUPPRESSION POOL WATER LEVEL: (CFR: 41.7 / 45.6) Condensate storage and transfer (make-up to the suppression pool): Plant-Specific (3.4/3.4)

QUESTION: 16 21307 (1 point(s))

Entry into the Emergency Operating Procedures (EOPs) has been made because of low suppression pool level. The CRS has directed that suppression pool makeup using RHR LOOP A per 5.8.14.

What is the RHR lineup that is used to makeup water to the suppression pool?

a. Water from pressure maintenance flows though RHR-MO-16A, RHR LOOP A MINIMUM FLOW VALVE to the suppression pool.

b. Water from the Condensate storage tank flows through RHR-98, CONDENSATE SUPPLY PUMP A SUCTION valve to the suppression pool.

c. Water from pressure maintenance flows through RHR-MO-38A, TORUS SPRAY OUTBD VLV and RHR-MO-34A, SUPPR POOL COOLING INBD THROTTLE VLV to the suppression pool.

d. Water from pressure maintenance flows through RHR-MO-39A, SUPPR POOL COOLING/TORUS SPRAY OUTBD VLV and RHR-MO-34A, SUPPR POOL COOLING INBD THROTTLE VLV to the suppression pool.

ANSWER: 16 21307

d. Water from pressure maintenance flows through RHR-MO-39A, SUPPR POOL COOLING/TORUS SPRAY OUTBD VLV and RHR-MO-34A, SUPPR POOL COOLING INBD THROTTLE VLV to the suppression pool.

The lineup used to add water to the suppression pool is from pressure maintenance through the SUPPR POOL COOLING/TORUS SPRAY OUTBD VLV and SUPPR POOL COOLING INBD THROTTLE VLV to the suppression pool.

Distractors:

a. This lineup is not used by 5.8.14 and the check valve on the discharge of the pump would prevent water flow to the SP.

b. This lineup is not used with RHR to fill the SP.

c. This lineup if it could be lined up would put water in the drywell.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
17	21809	00	10/05/2005	12/31/2003	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
Abnormal/Emergency Procedures	INT0080605, CRD flow is maximized to the RPV, where and how?

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

Related Objectives	
INT00806050011000	Given plant conditions and EOP flowchart 1A, RPV CONTROL, determine required actions.

Related References	
5.8.4	Alternate Injection Subsystems (Table 4)

Related Skills (K/A)	
2.1.30	Ability to locate and operate components / including local controls. (CFR: 41.7 / 45.7) (3.9/3.4) **NRC EXAM ONLY**

QUESTION: 17 21308 (1 point(s))

During an accident CRD flow is maximized to the RPV. Suction filter differential pressure has risen to >15 psid.

What is required?

Where does this action take place?

a. Swap suction to demineralized water by opening CM-68, CM/DM CROSS TIE TO CRD HYD PP. SUCT and closing CM-29 CRD HYD PPS SUPP.
Reactor Building 881 level SE QUAD.

b. Swap suction to demineralized water by opening CM-68, CM/DM CROSS TIE TO CRD HYD PP. SUCT and closing CM-29 CRD HYD PPS SUPP.
Reactor Building 903 SE.

c. Throttle closed CRD-170, CRD PP'S DISC. THROTTLE.
Reactor Building 903 SE.

d. Throttle closed CRD-170, CRD PP'S DISC. THROTTLE.
Reactor Building 881 level SE QUAD.

ANSWER: 17 21308

a. Swap suction to demineralized water by opening CM-68, CM/DM CROSS TIE TO CRD HYD PP. SUCT and closing CM-29 CRD HYD PPS SUPP.
Reactor Building 881 level SE QUAD.

If suction filter differential pressure becomes > 15 psid, swap suction to demineralized water system as follows:

Open CM-68, CM/DM CROSS TIE TO CRD HYD PP. SUCT (R-881-SE QUAD).

Close CM-29, CRD HYD PPS SUPP. (R-881-SE QUAD).

Distractors:

b. is incorrect because the valves that are required to be manipulated are in the SE Quad 881.

c.and d are incorrect because throttling the discharge is not required.

Modified 2278

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
18	21309	00	08/13/2005	10/01/2005	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
Emergency Operating Procedures	INT0080610, Can the Annunciator be used to go past FS/L-8.

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

Related Objectives	
INT00806100010800	Given plant conditions and EOP flowchart 7A, RPV LEVEL (FAILURE TO SCRAM), determine required actions.

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

Related Skills (K/A)	
295037.EA2.01	Ability to determine and/or interpret the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: (CFR: 41.10 / 43.5 / 45.13) Reactor power. (4.2*/4.3*)
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: 18 21309 (1 point(s))

The plant was at 100% power when an ATWS occurred. Reactor water level was intentionally lowered due to **Table 17 conditions**. SLC injection was initiated with an initial SLC tank level of 75%. Water level is now intentionally being lowered.

The following conditions were initially noted:

- Reactor water level is 120" (Corrected FZ)
- Reactor pressure is 900 psig and steady.
- Drywell pressure is 1.5 psig and steady **SRV's are being cycled** for pressure control.
- APRMs indicate between 10% and 12% power.

The following times/parameters were noted as the event progressed:

12:00 Reactor water level reaches 100" (Corrected FZ).
12.02 Annunciator APRM DOWNSCALE, 9-5-1/C-8 alarms and level is +85(FZ) and power is 10%.
12:04 Reactor power lowers to 2% by APRM indication.
12:06 Reactor water level lowers to -25 (Corrected FZ)

What is the **earliest** injection can be commenced?

a. 12:00

b. 12:02

c. 12.04

d. 12:06

ANSWER: 18 21309

c. 12.04

The APRM downscales can not be used to determine that power is less than 3% as their setpoint is well above this value. The first time condition of EOP-7A step FS/L-8 is met is 12:04 when power is less than 3%.

Distractors:
a. is incorrect because at 12:00 no conditions of FS/L-8 are met.
b. is incorrect because at 12:00 no conditions of FS/L-8 are met.
d. is incorrect because injection could be commenced prior to this time.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
19	21310	00	08/07/2005	10/01/2005	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	Plant ventilation that should be operated during period of a high offsite release.

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	
(B)(10)	Administrative, normal, abnormal, and emergency operating procedures for the facility.
(B)(13)	Procedures and equipment available for handling and disposal of radioactive materials and effluents.

Related Skills (K/A)	
295038.EA1.06	Ability to operate and/or monitor the following as they apply to HIGH OFF-SITE RELEASE RATE: (CFR: 41.7 / 45.6) Plant ventilation. (3.5/3.6)

QUESTION: 19 21310 (1 point(s))

The plant is shutdown for a refueling outage. The radwaste and turbine building HVAC are shutdown but available. A fuel handling accident occurs resulting in a group 6 isolation and SGT initiation. The Shift Manager declares a Site Area Emergency due to high offsite release rate.

What EOP action is appropriate for the radwaste and turbine building HVAC?

a. Leave both Turbine Building and Radwaste Building HVACs OFF.

b. Start BOTH the Turbine Building and the Radwaste Building HVAC.

c. Start the Radwaste HVAC and leave the Turbine Building HVAC OFF.

d. Start the Turbine Building HVAC and leave Radwaste Building HVAC OFF.

ANSWER: 19 21310

b. Start BOTH the Turbine Building and the Radwaste Building HVAC.

EOP-5A directs that the Turbine Building and Radwaste building HVAC be started.

a. is incorrect as they should be started.

c. is incorrect because the Turbine Building HVAC should also be started.

d. is incorrect because the Radwaste Building HVAC OFF should also be started.

Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
20	10617	02	02/18/2002	10/01/2005	None	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
None	INT0320134, 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).

Related References	
5.4POST-FIRE (B)(10)	Post Fire Operational Information Administrative, normal, abnormal, and emergency operating procedures for the facility.

Related Skills (K/A)
600000.AK2.04 Knowledge of the interrelations between PLANT FIRE ON SITE and the following: Breakers / relays / and disconnects (2.5/2.6)

QUESTION: 20 10617 (1 point(s))

Given the following conditions:

- ➢ A reactor scram occurs due to a spurious Group 1 isolation.
- ➢ RCIC and HPCI are started for level/pressure control.
- ➢ One fire is reported in a cable tray in the Reactor Building Southwest Quad.
- ➢ Another fire is reported in Reactor Building Southeast Quad preventing entry into the quad.
- ➢ Procedure 5.4 POST-FIRE has been entered.
- ➢ A valid Group 4 isolation signal is received, but no valves close due to the signal.

Based on the conditions above, which choice below describes the method to be utilized to mitigate the failure of the Group 4 isolation?

a. Close HPCI-MO-15 from MCC-R.

b. Close HPCI-MO-16 from EE-STR-125B.

c. Close RCIC-MO-15 from Control Room panel 9-4.

d. Close RCIC-MO-16 from Control Room panel 9-4.

ANSWER: 20 10617

a. Close HPCI-MO-15 from MCC-R.

b. incorrect. This method would be used if the fire were in the NW quad (Att. 12).
c. incorrect. Closing RCIC-MO-15 will have no effect on Group 4 failure.
d. incorrect. Closing RCIC-MO-16 will have no effect on Group 4 failure.

REFERENCE: Procedure 5.4.Post Fire

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
21	21312	00	08/13/2005	10/01/2005	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
Abnormal/Emergency Procedures	INT0320132, Knowledge of system status criteria that require the notification of plant personnel.

Related Lessons
INT0320132 CNS Abnormal Procedures (RO) Off Gas/Vacuum

Related Objectives	
INT0320132J0J0100	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).
INT0320132K0K0100	Given plant condition(s), determine from memory if a Main Turbine trip is required due to the event(s).
INT0320132L0L0100	Given plant condition(s), and the applicable Abnormal/Emergency Procedure, determine the correct subsequent actions required to mitigate the event(s).

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

Related Skills (K/A)	
2.1.14	Knowledge of system status criteria which require the notification of plant personnel (such as Reactivity Management Events). (CFR: 43.5 / 45.12) (2.5/3.3)

QUESTION: 21 21312 (1 point(s))

During operation a loss of condenser vacuum occurred due to the loss of a single CW pump. When conditions stabilized the following parameters were noted:

- ➢ Condenser vacuum is 24"Hg vacuum.
- ➢ Barometric Pressure is 29.92"Hg abs.
- ➢ Reactor power is 35% following a power reduction to maintain vacuum.

Three minutes after the loss of the CW pump the crew started an additional CW pump and restored vacuum to normal.

What is required, if anything, at this time?

a. No actions are required.

b. Scram the reactor then trip the turbine.

c. Trip the turbine and enter procedure 2.2.77.

d. Notify Turbine Engineering to determine if turbine inspections are required.

ANSWER: 21 21312

d. Notify Turbine Engineering to determine if turbine inspections are required.

2.4 VAC requires that if absolute pressure enters 5 Minute Exit Area of Attachment 3, Maximum Permissible Condensing Pressure, for any amount of time, contact Turbine Engineering for determination of turbine inspection requirements. In this event pressure did enter the 5 minute exit area and the notification is required.

Distractors:

a. is incorrect because the notification is required.

b. is incorrect no requirement to scram or trip the turbine has been met.

c. is incorrect as no requirement to immediately shutdown the turbine exists.

Source: New

Provide Attachment 3 of 2.4VAC to the Candidate.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
22	1252	01	07/24/2005	10/01/2005	Licensed Operator	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	COR0022202, How does the RR pumps respond to a low reactor water level?

Related Lessons
COR0022202 REACTOR RECIRCULATION

Related Objectives	
COR0022202001130C	Given plant conditions, determine if any of the following should occur: Recirculation pump runback to the dual limiter.
COR0022202001100L	Describe the Reactor Recirculation system and/or Recirculation Flow Control system design features and/or interlocks that provide for the following: Recirculation Pump Runback

Related References	
(B)(4)	Secondary coolant and auxiliary systems that affect the facility.
(B)(7)	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Related Skills (K/A)
295009.AK2.03 Knowledge of the interrelations between LOW REACTOR WATER LEVEL and the following: (CFR: 41.7 / 45.8) Recirculation system (3.1/3.2)

QUESTION: 22 1252 (1 point(s))

The plant was operating normally at 90% power when a single reactor feed pump trips. Reactor water level drops initially to +25" (NR) then recovers to its normal operating value.

The Reactor Recirculation pumps are…

a. tripped.

b. operating at ≈ 45% speed.

c. operating at ≈ 22% of rated speed.

d. operating with the scoop tubes locked out.

ANSWER: 22 1252

b. operating at ≈ 45% speed.

Reactor water level was below the low level alarm setpoint with only a single feed pump in operation. This results in both RR pumps running back to 45%.

a. is incorrect because they would trip.

c. is incorrect as they would only runback to 45%.

d. is incorrect as the scoop tubes should not be locked.

REFERENCE: STCOR002-22-02, page 29, section II.H.8, rev. 15.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
23	21298	00	07/15/2005	10/01/2005	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	INT0080618, Why is ED required if SP level can not be maintained greater than 9.6 ft.

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

Related Objectives	
INT0080613001040C	State the basis for primary containment control actions as they apply to the following: Graphs reference on Flowchart 3A
INT0080613001040A	State the basis for primary containment control actions as they apply to the following: Specific setpoints
INT0080613001040B	State the basis for primary containment control actions as they apply to the following: Primary Containment Control Systems

Related References	
(B)(10)	Administrative, normal, abnormal, and emergency operating procedures for the facility.
(B)(8)	Components, capacity, and functions of emergency systems.

Related Skills (K/A)	
295010.AK1.01	Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL PRESSURE: (CFR: 41.8 to 41.10) Downcomer submergence: Mark-I&II. (3.0/3.4)

QUESTION: 23 21298 (1 point(s))

An accident occurred that resulted in elevated drywell pressure and a lowering primary containment water level. Why is emergency depressurization required when PC water level cannot be maintained above 9.6 feet?

a. Steam condensation at the HPCI exhaust is no longer assured.

b. Drywell non-condensable levels over 1% are no longer assured.

c. Steam condensation at the SRV T-quenchers is no longer assured.

d. Steam suppression at the opening of the downcomers is no longer assured.

ANSWER: 23 21298

d. Steam suppression at the opening of the downcomers is no longer assured.

The RPV is not permitted to remain at pressure or at power if suppression of steam discharged from the RPV cannot be assured. A primary containment elevation of 9.6 feet is the opening of the downcomers, the point at which steam suppression can longer be assured.

Distractors:

a. is incorrect because the HPCI exhaust is uncovered at 11 ft. Uncovery of the HPCI exhaust does not require emergency depressurization.

b is incorrect because the drywell non-condensible consideration is based on a lowest value of torus pressure and not suppression chamber level.

c. is incorrect because steam condensation at the SRV T- quenchers is assured if level remains above 6 feet.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
24	21357	00	09/03/2005	10/01/2005	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
Abnormal/Emergency Procedures	INT0320123, Why is a scram required for an inadvertent reactivity addition.

Related Lessons
INT0320123 CNS Abnormal Procedures (RO) Reactivity

Related Objectives	
INT0320123G0G0100	Given plant condition(s), determine from memory all immediate operator actions required to mitigate the event(s).
INT0320123H0H0100	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	
(B)(5)	Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for
(B)(10)	Administrative, normal, abnormal, and emergency operating procedures for the facility.
2.4RXPWR	Reactor Power Anomalies

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: 24 21357 (1 point(s))

A plant startup is in progress with reactor power at 20%. The power ascension is stopped when the following occur:

- ➢ APRM indication rises to 30% then falls to 23%.
- ➢ Annunciator LPRM UPSCALE, 9-5-1/B-7 alarms and clears after 15 seconds.
- ➢ Several LPRM Upscale lights on the full core display illuminate then extinguish.
- ➢ Core Flow is relatively constant throughout the event.

What is required and what is assumed to have occurred?

a. Scram the reactor per 2.4RXPWR because a rod drop accident has occurred.

b. Scram the reactor per 2.4RXPWR because a core shroud crack has developed.

c. Reduce power per 2.1.10 because a rod drop accident has occurred.

d. Reduce power per 2.1.10 because a core shroud crack has developed.

ANSWER: 24 21357

a. Scram the reactor because a rod drop accident has occurred.

Per procedure 2.4RXPWR LPRM upscales require that the reactor be scrammed and to assume that a rod drop accident has occurred.

Distractors:

b. is incorrect because the indications are of a rod drop accident.

c. is incorrect because a reactor scram is required due to the LPRM upscale alarms.

d. is incorrect because a reactor scram is required due to the LPRM upscale alarms and indications are that a rod drop accident has occurred.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
25	21314	00	07/30/2005	10/01/2005	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	INT0320104 , Identify the RWCU control specified in the Alarm Response Procedure required to be operated following an inadvertent isolation.

Related Lessons
COR0012002 OPS Reactor Water Cleanup

Related Objectives	
COR0012002001040C	Describe the interrelationship between the RWCU system and the following: CRD System
COR0012002001130G	Given a RWCU component manipulation, predict and explain the changes in the following parameters: RWCU system pressure

Related References	
(B)(9)	Shielding, isolation, and containment design features, including access limitations.
(B)(10)	Administrative, normal, abnormal, and emergency operating procedures for the facility.

Related Skills (K/A)	
2.4.50	Ability to verify system alarm setpoints and operate controls identified in the alarm response manual. (CFR: 45.3) (3.3/3.3)

QUESTION: 25 21314 (1 point(s))

The plant is at rated power when annunciator RWCU HI SPACE TEMP 9-4-2/A-5, alarms followed by an a group 3 isolation. The cause of the isolation is due to failure of temperature detectors in the RWCU area.

What RWCU action is required and why?

a. Throttle open RWCU-MO-74 to prevent thermal binding of that valve.

b. Reset the isolation and crack open RWCU-MO-15 to prevent thermal binding of that valve.

c. Throttle open RWCU-MO-74 to prevent over-pressurizing the RWCU pump seals with CRDH mini-purge.

d. Reset the isolation and crack open RWCU-MO-15 to prevent over-pressurizing the RWCU pump seals with CRDH mini-purge.

ANSWER: 25 21314

c. Throttle open RWCU-MO-74 to prevent over-pressurizing the RWCU pump seals with CRDH mini-purge.

Both the AP and 2.1.22 state that MOP-74 be opened in order to prevent over pressurization of the pump seals with CRDH mini purge.

Distractors:

a. is incorrect because RWCU-MO-74 not in jeopardy of thermal binding but opening the valve will prevent overpressure of the seals,

b. is incorrect because RWCU-MO-15 is not cracked open to prevent thermal binding.

d. is incorrect because MO-74 is opened to prevent over pressurization.

Modified PTM 311

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
26	21315	00	08/13/2005	10/01/2005	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Emergency Operating Procedures	INT0080607, Interpret high containment water level and determine the required actions.

Related Lessons	
INT0080607	OPS EOP FLOWCHART 2A - EMERGENCY RPV DEPRESSURIZATION & STEAM COOLING

Related Objectives	
INT00806070010700	Given plant conditions and EOP flowchart 2A, EMERGENCY RPV DEPRESSURIZATION/STEAM COOLING, determine required actions.

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

Related Skills (K/A)	
295029.EA2.03	Ability to determine and/or interpret the following as they apply to HIGH SUPPRESSION POOL WATER LEVEL: (CFR: 41.10 / 43.5 / 45.13) Drywell/containment water level. (3.4/3.5)

QUESTION: 26 21315 (1 point(s))

An accident occurred that resulted in a high suppression pool water level. When suppression pool water level reached 16 feet the crew commenced emergency depressurization using RWCU with the filters bypassed, the reactor head vent and HPCI.

An hour later the following plant conditions were noted:

- ➢ Reactor pressure is 55 psig and slowly lowering
- ➢ Torus pressure is 25 psig and steady.
- ➢ Primary Containment Water level is 43 ft and rising.
- ➢ Reactor Water level is 45" (NR) and steady.

With no major changes in plant status foreseen, what action is required?

a. Isolate RWCU.

b. Close the head vent.

c. Raise HPCI steam flow.

d. Open the inboard MSL Drain valve.

ANSWER: 26 21315

d. Open the inboard MSL Drain valve.

Step RC/P-9 contains an override that if PC water level is anticipated to go over 44 ft that inboard MSL drain should be opened. This preserves this flowpath for later use.

Distractors:

a. is incorrect as no requirement to isolate RWCU is present.

b. is incorrect as the head vent may be maintained open to maintain the reactor depressurized.

c. is incorrect as the conditions for ED are met.

Provide EOP-2A to the candidate.

Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
27	21316	00	07/30/2005	10/01/2005	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	COR0022802, Monitor SGT System following Auto initiation from a group 6 isolation.

Related Lessons
COR0022802 OPS STANDBY GAS TREATMENT

Related Objectives	
COR002280200108OA	Describe the Standby Gas Treatment design features and/or interlocks that provide for the following: Automatic system initiation

Related References	
(B)(7)	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Related Skills (K/A)	
295035.EA1.02	Ability to operate and/or monitor the following as they apply to SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE: (CFR: 41.7 / 45.6) SBGT/FRVS. (3.8/3.8)

QUESTION: 27 21316 (1 point(s))

A Group 6 isolation occurs due to high radiation in the reactor building ventilation exhaust plenum. Following the automatic start of SGT the following parameters and control positions were noted:

- ➢ Reactor building pressure is 0.40 H_2O vacuum.
- ➢ SGT train A Filter train pressure drop is 11" of water D/P.
- ➢ SGT train B Filter train pressure drop is 11" of water D/P.
- ➢ SGT-DPCV-546A or SGT-DPCV-546B control switches are in AUTO.

What is the expected position/response of SGT differential pressure control valves (SGT-DPCV-546A or SGT-DPCV-546B) and the SGT fan inlet vortex dampers?

SGT differential pressure control valves…

a. are full open and remain full open and the vortex dampers are full open and remain full open.

b. are full open and remain full open and the vortex dampers close to obtain 10" H_2O D/P across their filter trains.

c. modulate to maintain 0.25" H_2O vacuum in the Reactor Building and the vortex dampers are full open and remain full open.

d. modulate to maintain 0.25" H_2O vacuum in the Reactor Building and the vortex dampers close to obtain 10" H_2O D/P across their filter trains.

ANSWER: 27 21316
b. are full open and remain full open and the vortex dampers close to obtain 10" H_2O D/P across their filter trains.

A group 6 isolation signal (the reactor building ventilation exhaust plenum radiation) signal causes the differential pressure control valves (SGT-DPCV-546A or SGT-DPCV-546B) to open and remain open until the signal is reset providing their respective control switches are in AUTO. The fan vortex control system limits air stream flow through each filter train so that total pressure drop across the train remains less than 10" of water D/P.

a. is incorrect because the vortex dampers would not be full open.
c. is incorrect because the D/P control valves are open and remain open until the isolation is reset.
d. is incorrect because the D/P control valves are open and remain open until the isolation is reset.
Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
28	21317	00	07/30/2005	10/01/2005	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
Systems	COR0022302, Predict the change in DG loading for an RHR system control manipulation.

Related Lessons
COR0022302 RESIDUAL HEAT REMOVAL

Related Objectives	
COR0022302001060I	Given an RHR control manipulation, predict and explain changes in the following: Emergency Diesel Generator loading

Related References	
(B)(14)	Principles of heat transfer thermodynamics and fluid mechanics.

Related Skills (K/A)	
203000.A1.08	Ability to predict and/or monitor changes in parameters associated with operating the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) controls including: (CFR: 41.5 / 45.5) ?Emergency generator loading (3.7/3.8)

QUESTION: 28 21317 (1 point(s))

Following a loss of coolant accident and a loss of offsite power the operator stopped RHR loop A pumps that automatically started. Five minutes later LPCI injection is required. At 09:59 the operator starts both RHR pump A and C.

The following actions/events occur:

10:00 RHR pumps A and C are running on minimum flow.
10:01 RHR-MO-27A is throttled open until LPCI injection flow is 8,000 gpm.
10:02 RHR pump C is secured and LPCI injection flow is 7000 gpm from RHR A pump only.
10:03 RHR-MO-27A is closed and RHR loop A Minimum flow valve opens.

At what time referenced above is **DG#1** loading at its **greatest**? (Consider only the effects of RHR on DG loading).

a. 10:00

b. 10:01

c. 10:02

d. 10:03

ANSWER: 28 21317

c. 10:02

The highest DG load occurs when the pump that is powered by DG #1 is at its greatest load. This would occur when the RHR pump was doing the most work. The highest flow rate through the A RHR pump occurs when C RHR pump is secured.

Distractors:

a. is incorrect because at this time RHR pump A flow is only at minimum.

b. is incorrect because with RHR pump C running A pump flow is less than at 10:02.

d. is incorrect because RHR pump flow is less than at 10:02

Source: NEW

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
29	21318	00	07/30/2005	10/01/2005	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
Abnormal/Emergency Procedures	INT0320136, Predict the effect of a loss of SDC on reactor pressure.

Related Lessons
INT0320136 CNS Abnormal Procedures (RO) Miscellaneous

Related Objectives	
INT0320136P0P0100	Given plant condition(s), and the applicable Abnormal/Emergency Procedure, determine the correct subsequent actions required to mitigate the event(s).

Related References	
(B)(4)	Secondary coolant and auxiliary systems that affect the facility.
(B)(10)	Administrative, normal, abnormal, and emergency operating procedures for the facility.

Related Skills (K/A)	
205000.K3.01	Knowledge of the effect that a loss or malfunction of the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) will have on following: (CFR: 41.7 / 45.4) Reactor pressure (3.3/3.3)

QUESTION: 29 21318 (1 point(s))

The plant is shutdown for refueling.

- ➢ The reactor has been shutdown for 24 hours.
- ➢ The reactor vessel head has not been removed.
- ➢ Reactor Water level is 58".
- ➢ A SDC System high pressure isolation occurs due to an instrument malfunction.
- ➢ The isolation signal cannot be reset, and no other method of decay heat removal can be established.
- ➢ Reactor water temperature is 150°F.

How long after the isolation is reactor pressure first expected to rise above 0 psig?

a. 0.6 hours

b. 1.1 hours

c. 1.9 hours

d. 2.1 hours

ANSWER: 29 21318

b. 1.1 hours

With the loss of shutdown cooling and the inability to establish any other method of decay heat removal Attachment 5 of 2.4SDC is used to establish the time to boiling. Once boiling starts the reactor pressurizes. From the graph temperature is greater than boiling in 1.0 hours. At this point the reactor would begin pressurizing so 1.1 hours would be the answer that you would expect to see pressure rising.

a. is incorrect as boiling would not yet be expected. This would be obtained if 170°F were used instead of 150°F.

c. is incorrect because the reactor is expected to begin pressurizing prior to this time and therefore this is not when it is FIRST expected.

d. is incorrect because the reactor is expected to begin pressurizing prior to this time and therefore this is not when it is FIRST expected.

Provide to the candidate Attachment 5 of 2.4SDC.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
30	20862	00	06/25/2004	10/01/2005	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
Systems	COR0021102, Ability to operate the HPCI system following the reset of an automatic initiation.

Related Lessons
COR0021102 OPS High Pressure Coolant Injection System

Related Objectives	
COR0021102001080K	Describe the HPCI design features and/or interlocks that provide for following: Turbine and pump lubrication

Related References	
791E271	HPCI System Elementary Diagram
(B)(4)	Secondary coolant and auxiliary systems that affect the facility.
(B)(10)	Administrative, normal, abnormal, and emergency operating procedures for the facility.

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)
(DO NOT LINK RO OR SRO MATERIAL [other than 2.1.1 and 2.1.4 material] TO THIS K/A. More specific K/As are available.)	

QUESTION: 30 20862 (1 point(s))

The plant was operating at power when a loss of feedwater occurred. The reactor scrammed and HPCI automatically initiated. The following conditions were noted:

- ➢ Drywell pressure is 0.5 psig.
- ➢ Reactor water is +35"(NR)

The crew determined that HPCI operation is no longer needed.

The initiation signal was reset; the AOP control switch was placed to STOP then back to AUTO. The HPCI turbine trip pushbutton is then depressed.

What is the AOP status before and after the turbine trip pushbutton is depressed?

Before the turbine is tripped the AOP is…

a. idle and remains idle during the HPCI coast down.

b. running and remains running after the turbine is tripped.

c. idle and starts immediately after the turbine trip pushbutton is depressed.

d. idle and starts when oil pressure from the shaft pump drops to approximately 30 psig.

ANSWER: 30 20862

a. idle and remains idle during the HPCI coast down.

Auxiliary oil pump will not restart automatically on turbine coast down unless either an initiation signal is present or control switch is positioned to START.

Distractors:

b. is incorrect because the pump is currently idle and remains idle following the trip.

c. is incorrect because the pump remains idle following trip of HPCI.

d. is incorrect as the pump will not start on low pressure.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
31	21319	00	07/30/2005	10/01/2005	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	COR0021102, Knowledge of the power supplies to HPCI system components.

Related Lessons
COR0021102 OPS High Pressure Coolant Injection System

Related Objectives
NONE

Related References	
(B)(7)	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Related Skills (K/A)	
206000.K2.02	Knowledge of electrical power supplies to the following: (CFR: 41.7) System pumps: BWR-2,3,4 (2.8*/3.1*)

QUESTION: 31 21319 (1 point(s))

What HPCI equipment is effected/deenergized if a loss of 250 VDC HPCI Starter Rack occurs? Assume HPCI was in a normal standby lineup prior to the loss.

a. HPCI system pumps and motive power to all valve motors, except MO-15, 16, 17 and 58.

b. HPCI initiation and isolation logic is lost and it would prevent HPCI trip on high water level.

c. HPCI control power for HPCI system pumps and valves and motive power for MO-17 and 58.

d. HPCI "B" logic relays and the Group 4 isolation logic B lights, turbine trip power, auto initiation; and MO-14 would not automatically reposition.

ANSWER: 31 21319

a. HPCI system pumps and motive power to all valve motors, except MO-15, 16, 17 and 58.

250 VDC HPCI Starter Rack is used to provide the motive force for all HPCI system pumps and all valve motors, except MO-15, 16, 17 and 58. A loss of this starter rack would prevent operation of those pumps and valves.

Distractors:

b. is incorrect as this HPCI equipment is what would be lost if AA2 (125 VDC) were lost.

c. is incorrect as this HPCI equipment is what would be lost if BB2 (125 VDC) were lost.

d. is incorrect as this HPCI equipment is what would be lost if 125 VDC HPCI SR were lost.

Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
32	19180	01	01/23/2004	10/01/2005	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	COR0020602, Core Spray leak detection.

Related Lessons
COR0020602 CORE SPRAY

Related Objectives	
COR0020602001090C	Predict the consequences of the following items on the Core Spray System: Core Spray line break

Related References	
2.3_9-3-3	Panel 9-3 - Annunciator 9-3-3
(B)(2)	General design features of the core, including core structure, fuel elements, control rods, core instrumentation, and coolant flow.

Related Skills (K/A)	
209001.A4.11	Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) System flow (3.7/3.6)

QUESTION: 32 19180 (1 point(s))

The plant is operating at full power when the following occur:

- ➢ Annunciator 9-3-3/A-5, CORE SPRAY B BREAK DETECTION alarms.
- ➢ NO other annunciators alarm.
- ➢ The d/p indicating switch (CS-DPIS-43B) reads +4.0 psid.

How will "B" Core Spray respond during a large break LOCA?

Water from the "B" Core spray system will flow . . .

a. into the Drywell.

b. inside the core shroud.

c. into secondary containment.

d. into the downcomer region of the vessel.

ANSWER: 32 19180

d. into the downcomer region of the vessel.

Justification: The alarm and d/p reading indicate the break is outside the shroud but inside the reactor.

Distractors:

a. The indicated d/p would be pegged high (+1000 psig). The dPIS-43A/B, -10 to +15 psid.
b. The indicated d/p would be low -3.5 psig.
c. The instrument measures d/p downstream of the check valve inside the primary containment.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
33	21320	00	07/31/2005	10/01/2005	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	COR0022902, Ability to monitor SLC pump discharge following system initiation.

Related Lessons
COR0022902 STANDBY LIQUID CONTROL

Related Objectives	
COR0022902001050H	Describe the SLC design features and/or interlocks that provide for the below: Dampening of positive displacement pump discharge oscillations

Related References	
(B)(8)	Components, capacity, and functions of emergency systems.
(B)(14)	Principles of heat transfer thermodynamics and fluid mechanics.
(B)(7)	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Related Skills (K/A)	
211000.A3.01	Ability to monitor automatic operations of the STANDBY LIQUID CONTROL SYSTEM including: (CFR: 41.7 / 45.7) Pump discharge pressure: Plant-Specific (3.5/3.5)

QUESTION: 33 21320 (1 point(s))

The plant has experienced an ATWS and the CRS directs the RO to initiate Standby Liquid Control System (SLC) injection. The RO attempted to initiate the both pumps but only A SLC pump started. The following indications were noted:

- ➢ SLC "A" SQUIB CONTINUITY light is extinguished.
- ➢ SLC "B" SQUIB CONTINUITY light is lit.
- ➢ Reactor pressure is 950 psig and steady.
- ➢ SLC pump discharge pressure is greater than reactor pressure and has rapid fluctuations between 950 psig and 1050 psig.

What is the cause of these fluctuations?

SLC pump A…

a. squib valve has failed to fire.

b. suction is completely obstructed.

c. discharge relief valve is stuck open.

d. discharge accumulators have no gas pressure.

ANSWER: 33 21320

d. discharge accumulators have no gas pressure.

Accumulators located on the discharge piping of the SLC pumps, dampen the pulsations of the positive displacement pumps. The accumulators are constructed of a steel vessel with a butyl synthetic bladder. The upper side of the bladder is charged with nitrogen gas to 775 psig and the underside receives pulsations from the pump discharge. The loss of this gas pressure would result in larger fluctuations in discharge pressure.

Distractors:

a. is incorrect as this would result in fluctuations but the pressure would be fluctuating at the relief valve setpoint of 1540 psig.
b. is incorrect as this would prevent the development of any discharge pressure.
c. is incorrect as this would cause pressure to be lower than reactor pressure.

Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
34	21321	00	07/30/2005	10/01/2005	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	COR0022102, RPS Design feature that prevents supply RPS bus from multiple sources.

Related Lessons
COR0022102 REACTOR PROTECTION SYSTEM

Related Objectives	
COR0022102001040C	Describe the RPS design features and/or interlocks that provide for the following: Multiple power source to RPS bus prevention

Related References	
(B)(6)	Design, components, and functions of reactivity control mechanisms and instrumentation.
(B)(7)	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Related Skills (K/A)	
212000.K4.03	Knowledge of REACTOR PROTECTION SYSTEM design feature (s) and/or interlocks which provide for the following: (CFR: 41.7) The prevention of supplying power to a given RPS bus from multiple sources simultaneously (3.0*/3.1*)

QUESTION: 34 21321 (1 point(s))

RPS A is being supplied from alternate power when the RPS MG set is restored to service and operability. When RPS is transferred from its alternate source to the MG set what component prevents simultaneous powering of RPS A from two power sources?

a. RPS BUS A PWR TRANSFER switch.

b. RPS MG set MOTOR ON pushbutton.

c. Electrical Protection Assembly (EPA) circuit breakers.

d. RPS Alternate Feed Transformer supply breakers and RPS MG supply breaker Auto Bus Transfer device.

ANSWER: 34 21321

a. RPS BUS A PWR TRANSFER switch.

All transfer operations must be done manually through local controls and the RPS BUS A PWR TRANSFER switch in the Control Room. During the power transfer operation, the RPS power panel must be de-energized since the RPS MG and the alternate source cannot be paralleled. De-energizing the respective trip channel will cause a half scram, and part of Primary Containment Isolation System (PCIS) to operate since the PCIS is also powered from the RPS power panels.

Distractors:

b. This control only starts the RPS MG set.

c. The Electrical Protection Assembly (EPA) circuit breakers automatically isolate the RPS power panels from their respective power supply if the supplied voltage or frequency is not within the limits which allow for safe operation of the system.

d. RPS Alternate Feed Transformer supply breakers and RPS MG supply breaker Auto Bus Transfer device.

Source: Direct INPO 19659

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
35	5353	01	07/30/2005	10/01/2005	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	COR0021202, Effect of operating controls on the status of IRM rod block.

Related Lessons
COR0021202 INTERMEDIATE RANGE MONITOR

Related Objectives	
COR0021202001090A	Given plant conditions, determine if the following IRM actions should occur: Rod Block.
COR0021202001030C	Describe the interrelationships between IRM subsystem and the following: Operating the mode switch

Related References	
(B)(7)	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
4.1.2	Intermediate Range Monitoring System
(B)(6)	Design, components, and functions of reactivity control mechanisms and instrumentation.

Related Skills (K/A)	
215003.A1.04	Ability to predict and/or monitor changes in parameters associated with operating the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM controls including: (CFR: 41.5 / 45.5) Control rod block status (3.4/3.4)

QUESTION: 35 5353 (1 point(s))

Given the following:

- ➢ A Reactor startup in progress.
- ➢ All IRMs are on range 9 and reading between 9 and 11 on the lower scale.
- ➢ The Reactor Mode switch is in START & HOT STBY.
- ➢ IRM A fails downscale.

Which choice below describes Rod Block status for the present conditions **AND** after the Reactor Mode Switch is in RUN? (Use Tech Spec values).

a. **NO** Rod Block exists.
Placing the Reactor Mode switch in RUN will result in a Rod Block.

b. A Rod Block exists.
The Rod Block will clear after placing the Reactor Mode switch in RUN.

c. **NO** Rod Block exists.
Placing the Reactor Mode switch in RUN will **NOT** result in a Rod Block.

d. A Rod Block exists.
The Rod Block will **NOT** clear after placing the Reactor Mode switch in RUN

ANSWER: 35 5353

b. A Rod Block exists.
The Rod Block will clear after placing the Reactor Mode switch in RUN.

EXPLANATION OF ANSWER: b. Correct. An IRM downscale will generate a rod block. All IRM SCRAMs, Rod Blocks and Alarms are bypassed when the Mode Switch is in RUN, except Upscale or INOP with the companion APRM downscale. a,c. An IRM downscale will generate a rod block with the Mode Switch not in RUN. d. All IRM SCRAMs, Rod Blocks and Alarms are bypassed when the mode switch is in RUN, except Upscale or IPOP with the companion APRM Downscale.

REFERENCE: STCOR0021202 Intermediate Range Monitor Page 16 Section III.B Rev 9; PR 4.1.2 Intermediate Range Monitoring System Page 3 Section 6 Rev 14

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
36	21322	00	07/30/2005	10/01/2005	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	COR0011002, Operational implications of IRM Detector operation.

Related Lessons
COR0011002 INTRODUCTION TO NEUTRON MONITORING

Related Objectives	
COR0011002001030A	Concerning the Neutron Monitoring System: State which detectors operate in the ionization region and which detectors operate in the proportional region of the gas conductivity curve.
COR0011002001030B	Concerning the Neutron Monitoring System: State the reason for operating a detector in the specified region of the gas conductivity curve.
COR0011002001050A	Describe how changes in each of the following affect detector sensitivity: Operating voltage
COR0011002001050B	Describe how changes in each of the following affect detector sensitivity: Detector gas pressure
COR0011002001050C	Describe how changes in each of the following affect detector sensitivity: Amount of active coating

Related References	
(B)(6)	Design, components, and functions of reactivity control mechanisms and instrumentation.

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

QUESTION: 36 21322 (1 point(s))

What feature of the Intermediate Range System (IRM) helps reduce the sensitivity of the IRM detectors allowing power indication above the source range?

a. IRM detectors inner coating is enriched in U-234.

b. Lower voltage is applied to the SRM detectors than the IRM detectors.

c. Higher argon pressure is used in the SRM detectors than the IRM detectors.

d. IRM detectors inner U-235 coating is thicker in IRM detectors than SRM detectors.

ANSWER: 36 21322

c. Higher argon pressure is used in the SRM detectors than the IRM detectors.

Differences between SRM and IRM Ion Chamber Detectors

SRMs have more uranium than the IRMs.

SRMs have a higher gas pressure than the IRMs.

SRMs operate at a higher voltage than do the IRMs.

The SRMs are, due to the above listed reasons, more sensitive than the IRMs.

Distractors:

a. is incorrect because the IRM detectors inner coating is NOT enriched in U-234.

b. is incorrect because the SRM is supplied with higher voltage than the IRM.

d. is incorrect because the IRM detectors inner U-235 coating is not thicker than that of the SRMs.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
37	21323	00	07/30/2005	10/01/2005	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	COR0023002, Monitor and operate SRM back panel switches.

Related Lessons
COR0023002 SOURCE RANGE MONITOR SUBSYSTEM

Related Objectives	
COR0023002001050D	Given an SRM control manipulation, predict changes in the following: Control rod block status

Related References	
(B)(6)	Design, components, and functions of reactivity control mechanisms and instrumentation.

Related Skills (K/A)	
215004.A4.05	Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) SRM back panel switches, meters, and indicating lights (3.1/3.2)

QUESTION: 37 21323 (1 point(s))

The following conditions exist:

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

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

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

a. Partially withdraw SRM A detector.

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

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

d. Ensure the Reset Switch on Panel 9-12 is maintained in the Mid Position while the Function switch is in STANDBY.

ANSWER: 37 21323

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

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

Distractors:

a. is incorrect as this would have no effect on the rod block.
b. is incorrect because the this would bypass the detector retract permit and SRM downscale rod blocks not the function switch out of operate.
d. is incorrect because this switch does not bypass the INOP generated by the function switch out of operate.

Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
38	21360	00	09/06/2005	10/01/2005	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	COR0020102, What will cause the APRM to generate a scram signal.

Related Lessons
COR0020102 AVERAGE POWER RANGE MONITOR

Related Objectives	
COR0020102001050D	Describe the interrelationships between the Average Power Range Monitor System and the following: Local Power Range Monitoring System (LPRM)
COR0020102001100B	Predict the consequences a malfunction of the following would have on the Average Power Range Monitor System: LPRM detectors
COR0020102001100C	Predict the consequences a malfunction of the following would have on the Average Power Range Monitor System: Trip units

Related References	
(B)(6)	Design, components, and functions of reactivity control mechanisms and instrumentation.

Related Skills (K/A)	
215005.K4.02	Knowledge of AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM design feature (s) and/or interlocks which provide for the following: (CFR: 41.7) Reactor SCRAM signals (4.1*/4.2)

QUESTION: 38 21360 (1 point(s))

Given the following conditions:

- ➢ The plant is operating at 55% power
- ➢ Average Power Range Monitoring (APRM) Channel "C" currently has 13 "good" LPRM input signals

The following events occur:

16:00 One (1) of the "good" LPRM inputs fails downscale.
16:05 A second "good" LPRM input fails downscale.
16:10 Mode Switch for both LPRMs that failed downscale are placed to Calibrate.
16:15 RPS "A" is deenergized.

When does annunciator 9-5-1/A-7, APRM RPS CH A UPSCALE TRIP OR INOP **first** alarm?

a. 16:00

b. 16:05

c. 16:10

d. 16:15

ANSWER: 38 21360

c. 16:10

EXPLANATION OF ANSWER: c. Correct. The INOP trip Unit will trip when only 11 (<12) good inputs remain. Placing the LPRM mode switches out of "Operate" removes those LPRM inputs from the APRM channel count circuit. Since this leaves only 11 good LPRM inputs, an INOP trip occurs. a,b. A downscale failure of an LPRM does not remove the input from the count circuit of the destination APRM. d. This would eventually lead to an APRM Inop but it happens five minutes after the LPRMs caused one.

REFERENCE: COR0020102 Average Power Range Monitoring Page 9 Section II.C, Fig. 2 Rev 12; PR 2.3.2.27 Alarm Procedure Panel 9-5-1 Page 9 Section A-7 Rev 24

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
39	16402	01	05/19/2004	10/01/2005	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	COR0021802, RCIC - How does RCIC respond to a loss of AA-2?

Related Lessons
COR0021802 OPS Reactor Core Isolation Cooling

Related Objectives	
COR0021802001060C	State the electrical power supply to the following RCIC Items: Flow controller
COR0021802001100B	Predict the consequences of the following on the RCIC system: AC and/or DC Electrical power failure

Related References	
2.3_9-4-1	Panel 9-4 - Annunciator 9-4-1
(B)(7)	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Related Skills (K/A)	
217000.K6.01	Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC): (CFR: 41.7 / 45.7) Electrical power (3.4/3.5)

QUESTION: 39 16402 (1 point(s))

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

What effect does this power failure have on the RCIC system if a valid initiation condition occurs?

a. RCIC **WILL** automatically start and inject at rated flow, but **CANNOT** be manually tripped from the control room.

b. RCIC **WILL** automatically start and inject at rated flow, but **WILL NOT** shutdown automatically on high RPV water level.

c. RCIC **WILL NOT** automatically start, but it **CAN** be manually started and made to inject at rated flow from the control room.

d. RCIC **WILL NOT** automatically start and **CANNOT** be manually started and made to inject at rated flow from the control room.

ANSWER: 39 16402

d. RCIC **WILL NOT** automatically start and **CANNOT** be manually started and made to inject at rated flow from the control room.

GE Drawing 791E264, Elementary Control Diagram.

Distracter b: RCIC will not start since power was lost to relay which opens MO-131.
Distracter c: RCIC cannot be started because power was lost to the flow controller.
Distracter d: RCIC will not start because power was lost to relay which opens MO-131 and it cannot be manually started because power was lost to the flow controller.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
40	14668	01	05/27/2004	10/01/2005	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	COR0021602, Interlocks of ADS Logic Control

Related Lessons
COR0021602 OPS NUCLEAR PRESSURE RELIEF

Related Objectives	
COR0021602001080E	Predict the consequences a malfunction of the following would have on the NPR system: A.C. power

Related References	
10CFR55.41	Written examinations: Operators
791E253	Automatic Blowdown System
791E261	RHR Elementary Diagram
791E265	Core Spray Elementary Diagram
(B)(7)	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
(B)(8)	Components, capacity, and functions of emergency systems.

Related Skills (K/A)	
218000.K4.03	Knowledge of AUTOMATIC DEPRESSURIZATION SYSTEM design feature (s) and/or interlocks which provide for the following: (CFR: 41.7) ADS logic control (3.8/4.0)

QUESTION: 40 14668 (1 point(s))

The plant was operating at power when a loss of coolant accident occurred. The reactor scrammed and following the turbine trip, offsite power was lost. HPCI and RCIC failed to start and attempts to manually start them were unsuccessful. **DG-1 started and loaded**, but **DG-2 failed to start** and could not be manually started. ADS actuated after the timers timed out and **all six ADS valves opened**. The following plant conditions were present:

- Reactor Pressure 450 psig and lowering
- Reactor is - 120" Wide Range and lowering.
- CS pump 1A running
- RHR pumps 1A and 1B running

Which of the following conditions would result in the **closure of all the ADS valves**?

a. Loss of AA-2.

b. Loss of BB-2.

c. Loss of DG-1.

d. Level restored to >113".

ANSWER: 40 14668

c. Loss of DG-1.

The loss of DG-1 will result in the loss of the running CS and RHR pumps and the loss of the AC interlock. When this occurs the K6A and K7A relays in both ADS logics drop out and the ADS valves close.

a. is incorrect. Because the loss of AA-2 would only result in the loss of the A side logic. The B side logic would keep the ADS valves open. The loss of BB-2 would cause the B logic to automatically transfer to AA-2 and the ADS valves would remain open.

b. is incorrect. because the loss of BB-2 would cause the B logic to automatically transfer to AA-2 and the ADS valves would remain open.

d. is incorrect. because once TDPU 2E-K5A(B) picks up the ADS valves open and the low water level is sealed in, so even if level is restored the ADS valves would remain open.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
41	20802	00	05/31/2004	10/01/2005	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
Systems	COR0021802, RCIC Automatic Initiation/Manual Isolation Circuit

Related Lessons
COR0021802 OPS Reactor Core Isolation Cooling

Related Objectives	
COR0021802001120B	Given plant conditions, determine if the following RCIC actions should occur: RCIC system isolation
COR0021802001080F	Describe the RCIC system design features and/or interlocks that provide for the following: Manual initiation

Related References	
2.2.67	Reactor Core Isolation Cooling System
(B)(9)	Shielding, isolation, and containment design features, including access limitations.

Related Skills (K/A)	
223002.A1.01	Ability to predict and/or monitor changes in parameters associated with operating the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF controls including: (CFR: 41.5 / 45.5) System indicating lights and alarms (3.5/3.5)

QUESTION: 41 20802 (1 point(s))

The plant was operating at 100% power when a loss of 4160V busses 1A and 1B occurred. RCIC and HPCI **automatically** started when level lowered to - 42" (WR). After a few minutes the following plant conditions were noted:

- ➢ RPV water is 15" (WR).
- ➢ RPV pressure is 810 psig.
- ➢ Drywell pressure is 2.5 psig.

The RCIC Isolation Pushbutton is depressed. (This is the only RCIC related action taken during the transient to this point).

What is the status of the RCIC isolation valves and Group 5 indicating lights?

a. **Only** the RCIC Inboard Steam Line Isolation valve (MO-15) closes and **only** the A side Group 5 lights on 9-5 extinguish.

b. **Only** the RCIC Outboard Steam Line Isolation valve (MO-16) closes **only** the B side Group 5 lights on 9-5 extinguish.

c. **Both** the RCIC Inboard (MO-15) and the RCIC Outboard Steam Line Isolation (MO-16) valves close and **all** Group 5 lights on 9-5 remain lit.

d. **Neither** the RCIC Inboard (MO-15) nor the RCIC Outboard Steam Line Isolation (MO-16) valves close and **all** Group 5 lights on 9-5 remain lit.

ANSWER: 41 20802

d. **Neither** the RCIC Inboard (MO-15) nor the RCIC Outboard Steam Line Isolation (MO-16) valves close and **all** Group 5 lights on 9-5 remain lit.

An initiation signal must be present for the isolation pushbutton to cause the isolation. Since the initiation signal is reset no isolation occurs and the 9-5 lights do not change state.

a. is incorrect as no RCIC valves reposition and no lights change state.

b. is incorrect as no RCIC valves reposition and no lights change state.

c. is incorrect as no RCIC valves reposition.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
42	21326	00	08/13/2005	10/01/2005	NRC Style Question	RO: SRO: NLO:	Y Y Y

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

Topic Area	Description
Systems	INT0080607, Relationship between the SRVs and the Suppression pool.

Related Lessons	
INT0080607	OPS EOP FLOWCHART 2A - EMERGENCY RPV DEPRESSURIZATION & STEAM COOLING

Related Objectives	
INT00806070010800	Given plant conditions and EOP flowchart 2A, EMERGENCY RPV DEPRESSURIZATION/STEAM COOLING, state the reasons for the actions contained in the steps.

Related References	
(B)(3)	Mechanical components and design features of the reactor primary system.
(B)(14)	Principles of heat transfer thermodynamics and fluid mechanics.

Related Skills (K/A)	
239002.K1.07	Knowledge of the physical connections and/or cause-effect relationships between RELIEF/SAFETY VALVES and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) Suppression pool

QUESTION: 42 21326 (1 point(s))

With the reactor at near rated pressure, what is the **minimum** suppression pool level that SRVs can safely be opened?

a. 4.8 ft.

b. 6.0 ft.

c. 9.6 ft.

d. 11.0 ft.

ANSWER: 42 21326

b. 6.0 ft.

6 feet is the top of the SRV t-quenchers. The SRVs can be opened at this level but not below. EOP-2A, Emergency Depressurization allows the use of the SRVs at 6 ft.

Distractors:

a. is incorrect as SP level is below the t-quenchers.

c. is incorrect as this is not the minimum the question asked for the minimum is below 9.6 ft.

d. is incorrect as this is not the minimum the question asked for the minimum is below 11.0 ft.

Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
43	738	00	10/06/1997	10/01/2005	Licensed Operator	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	COR0021602 Nuclear Pressure Relief

Related Lessons
COR0021602 OPS NUCLEAR PRESSURE RELIEF

Related Objectives	
COR0021602001020B	State the electrical power supply to the following NPR components: SRV solenoids

Related References	
(B)(7)	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Related Skills (K/A)	
239002.K2.01	Knowledge of electrical power supplies to the following: (CFR: 41.7) SRV solenoids (2.8*/3.2*)

QUESTION: 43 738 (1 point(s))

How is the **normal** power supply provided to the Low-Low Set (LLS) logic channels?

The LLS logic channels are NORMALLY powered by 125 VDC panel ...

a. AA2 via the associated SRV solenoid fuses.

b. BB2 through the associated SRV solenoid fuses.

c. AA2 for Channel A and BB2 for Channel B via the ADS logic fuses.

d. BB2 for Channel A and AA2 for Channel B via the ADS logic fuses.

ANSWER: 43 738

a. AA2 via the associated SRV solenoid fuses.

Both LLS logic channels are normally powered from Panel AA2, with an alternate supply from Panel BB2, through the normal power supply fuses for the associated SRVs.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
44	19729	01	08/06/2005	10/01/2005	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	COR0023202, The effect that a loss of the main steam flow input will have on the Reactor Water Level Control System.

Related Lessons
COR0023202 OPS REACTOR VESSEL LEVEL CONTROL

Related Objectives	
COR0023202001060I	Predict the consequences of the following on the RVLC system: Loss of one or more main steam flow inputs

Related References	
(B)(7)	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
2.4RXLVL	RPV Water Level Control Trouble
(B)(5)	Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for

Related Skills (K/A)	
259002.K6.03	Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR WATER LEVEL CONTROL SYSTEM: (CFR: 41.7 / 45.7) Main steam flow input (3.1/3.1)

QUESTION: 44 19729 (1 point(s))

The plant is operating at **25%** power with RFC-LC-83 (MASTER LEVEL CONTROLLER) tape set at 35" when the "C" Main Steam Flow instrument fails downscale.

How is Reactor water level affected?

RPV water level…

a. stabilizes at 32 inches.

b. stabilizes at 37 inches.

c. rises causing the RFPs to trip.

d. lowers causing a reactor scram.

ANSWER: 44 19729

a. stabilizes at 32 inches.

Per 2.4RXLVL
If the signal from a steam flow instrument(s) is lost, while in 3 ELEMENT control at 100% power, RPV level will drop rapidly and then will stabilize at ~ 23". If reactor power is < 100%, level will stabilize at a proportionate value (e.g., 50% power, ~ 29").

12" drop at 100% power => 3" drop at 25% power.

Distractors:

b. is incorrect as the loss of a MSL flow will cause water level to lower.

c. is incorrect because water level lowers and the RFPs remain running.

d. is incorrect because the loss of one MSL flow monitor at 25% power will not reduce water level to anywhere near the scram setpoint.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
45	5827	01	08/06/2005	10/01/2005	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	COR0022802, The effect a loss of CCP-1B would have on SGT.

Related Lessons
COR0022802 OPS STANDBY GAS TREATMENT

Related Objectives	
COR0022802001100A	Predict the consequences of the following on the Standby Gas Treatment system: AC and DC power failures

Related References	
(B)(7)	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Related Skills (K/A)	
261000.K6.01	Knowledge of the effect that a loss or malfunction of the following will have on the STANDBY GAS TREATMENT SYSTEM: (CFR: 41.7 / 45.7) A.C. electrical distribution (2.9/3.0)

QUESTION: 45 5827 (1 point(s))

Given the following conditions:

- ➢ The plant is operating at 100% power
- ➢ An electrical fault de-energizes 120 VAC Panel CCP-1B

What is the effect on the "B" Standby Gas Treatment (SGT) train?

a. SGT Fan "1F" will automatically start if auto start signal is present.
SGT Train "B" Inlet **AND** Outlet Valves fail open.

b. SGT Fan "1F" will **NOT** automatically start if auto start signal is present.
SGT Train "B" Inlet **AND** Outlet Valves fail open.

c. SGT Fan "1F" will automatically start if auto start signal is present.
SGT Train "B" Inlet **AND** Outlet Valves fail closed.

d. SGT Fan "1F" will **NOT** automatically start if auto start signal is present.
SGT Train "B" Inlet **AND** Outlet Valves fail closed.

ANSWER: 45 5827

a. SGT Fan "1F" will automatically start if auto start signal is present.
SGT Train "B" Inlet **AND** Outlet Valves fail open.

The 120V AC critical Panel CCP-1B provides control power and solenoid operated valve power for SGT train B. The inlet and discharge valves will fail to their full open position (fail safe) on a loss of power or control air pressure, in order to allow for system operation should it be required.

Distractors:

b. is incorrect because SGT Fan "1F" WILL automatically start.

c. is incorrect because the SGT Train "B" Inlet **AND** Outlet Valves fail open.

d. is incorrect because SGT Fan "1F" WILL automatically start.

Reference: COR0022802 SBGT, BR 3038 and 3065

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
46	21327	00	08/06/2005	10/01/2005	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	INT0320131, Recognize abnormal indications that require entry into 5.3NBPP.

Related Lessons
INT0320131 CNS Abnormal Procedures (RO) Electrical

Related Objectives	
INT0320131S0S0100	Given plant condition(s), determine from memory the appropriate Abnormal/Emergency Procedure(s) to be utilized to mitigate the event(s).

Related References	
5.3NBPP	No Break Power Panel Failure
(B)(10)	Administrative, normal, abnormal, and emergency operating procedures for the facility.

Related Skills (K/A)	
2.4.4	Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures. (CFR: 41.10 / 43.2 / 45.6) (4.0/4.3)

LINK ONLY TO EOP/AOP LESSONS/QUESTIONS

QUESTION: 46 21327 (1 point(s))

With the NBPP initially in a normal lineup what set of annunciator/plant indications would, by themselves require entry into **5.3NBPP**?

a. Annunciator "No Break Inverter 1A Volt Failure /Panel C", C-4/E-7 and control room **NBPP voltage** indicates 110 VAC.

b. Annunciator "No Break PWR PNL Loss of Voltage Panel C", C-4 /D-7 and control room **NBPP voltage** indication is zero (0).

c. Annunciator "No Break Inverter 1A Volt Failure /Panel C", C-4/E-7 and a SO report that **NBPP inverter** voltage and frequency indicate zero (0).

d. Annunciator "No Break Sys Emergency AC Failure Panel C", C-4/F-7 and SO report that **emergency AC power** frequency and voltage indicate zero (0).

ANSWER: 46 21327

b. Annunciator "No Break PWR PNL Loss of Voltage Panel C", C-4 /D-7 and control room **NBPP voltage** indication is zero (0).

Entry into 5.3NBPP is required if a loss of NBPP voltage occurs. Annunciator "No Break PWR PNL Loss of Voltage Panel C", C-4 /D-7 and control room **NBPP voltage** indication is zero (0) are both indications of this condition.

Distractors:

a. is incorrect because this Annunciator "No Break Inverter 1A Volt Failure /Panel C", C-4/E-7 is only indicative of an inverter failure and would not combined with control room **NBPP voltage** indication of 110 VAC require entry into 5.3NBPP.

c. is incorrect because Annunciator "No Break Inverter 1A Volt Failure /Panel C", C-4/E-7 and a SO report that **NBPP inverter** voltage and frequency indicate zero (0) does not mean that NBPP is deenergized. The static switch would transfer loads to MCC-R.

d. is incorrect because Annunciator "No Break Sys Emergency AC Failure Panel C", C-4/F-7 and SO report that **emergency AC power** frequency and voltage indicate zero (0) is only indicative of a loss of power from MCC-R to the static switch.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
47	1108	03	07/16/2004	10/01/2005	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	COR0020802, LOOP, then place DG switch in STOP

Related Lessons
COR0020802 DIESEL GENERATORS COR0010102 AC Electrical Distribution

Related Objectives	
COR0020802001060B	Describe the interrelationship between Diesel Generators and the following: AC Electrical Distribution

Related References	
2.2.20	Standby AC Power System (Diesel Generator)

Related Skills (K/A)	
262001.A3.02	Ability to monitor automatic operations of the A.C. ELECTRICAL DISTRIBUTION including: Automatic bus transfer (CFR: 41.7 / 45.7)

QUESTION: 47 1108 (1 point(s))

With the plant at 100% power the following occurs:

- ALL Off-site power was lost.
- 1F and 1G are being powered from the diesels.
- NO operator actions have been taken on panel "C".
- Power has been restored to the emergency transformer

What is the effect (if any) of placing the control switch for the Diesel Generator (DG-1) on Panel C to "STOP" and then releasing it?

a. EG1 opens momentarily; then recloses to supply the bus.

b. EG1 remains closed and DG-1 continues to supply the bus.

c. EG1 opens; the bus becomes de-energized and remains de-energized.

d. EG1 opens and 1FS closes to supply the bus from the Emergency Transformer.

ANSWER: 47 1108

d. EG1 opens and 1FS closes to supply the bus from the Emergency Transformer.

EXPLANATION OF ANSWER: The diesel generator control switch in the control room is bypassed when an automatic start signal is present, but when the start signal clears (power restored), placing the control switch to STOP on Pnl C will break the seal-in circuitry to the 4EMX3 relay and shut down the diesel. Breaker EG1 will open on the undervolatge condition, which satisfies the conditions necessary for 1FS to auto close and supply the bus from the ESST.

Ref: EDG Student Text, CB G5-262-743; Cooper-Bessemer Cabinet Control, Sheet 1 2.2.20 p. 28, step 2.5 of Information section.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
48	21328	00	08/07/2005	10/01/2005	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
Abnormal/Emergency Procedures	INT0320131, Predict the effect of lowering voltage on NBPP and what actions mitigate the effect of this failure.

Related Lessons	
INT0320131	CNS Abnormal Procedures (RO) Electrical

Related Objectives	
INT0320131V0V0100	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	
(B)(5)	Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for

Related Skills (K/A)	
262002.A2.01	Ability to (a) predict the impacts of the following on the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of...: (CFR: 41.5 / 45.6) Under voltage (2.6/2.8)

QUESTION: 48 21328 (1 point(s))

The plant is operating at rated power with the NBPP supplied through its static switch from the AC source. The NBPP inverter is available with voltage. A Station Operator reports that NBPP AC supply voltage is rapidly lowering.

If this trend continues what will be the status of NBPP?

Based on the prediction above what action should be taken?

a. NBPP transfers to the inverter.
Open the breaker from the AC source to the inverter.

b. NBPP transfers to the inverter.
Place NBPP PWR TRANSFER switch to IVTR.

c. NBPP de-energizes.
Scram the reactor and then Place NBPP PWR TRANSFER switch to IVTR.

d. NBPP de-energizes.
Place NBPP PWR TRANSFER switch to IVTR and if NBPP remains deenergized then scram the reactor.

ANSWER: 48 21328

c. NBPP de-energizes.
Scram the reactor and then Place NBPP PWR TRANSFER switch to IVTR.

The NBPP has no automatic transfer back to the inverter. If the current trend continued NBPP would deenergize. If deenergized a reactor scram is required. After the reactor is scrammed then actions are taken to restore NBPP to service.

a. is incorrect because the NBPP does not automatically transfer to the inverter.

b. is incorrect because the NBPP does not automatically transfer to the inverter.

d. is incorrect because with NBPP deenergized a scram is required first followed by actions to reestablish NBPP.

Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
49	21329	00	08/15/2005	10/01/2005	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	COR0020702, Predict the impact of a loss of ventilation on the battery and based on that prediction determine actions to mitigate the impact.

Related Lessons
COR0020702 OPS DC ELECTRICAL DISTRIBUTION

Related Objectives
NONE

Related References	
(B)(7)	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
(B)(8)	Components, capacity, and functions of emergency systems.

Related Skills (K/A)	
263000.A2.02	Ability to (a) predict the impacts of the following on the D.C. ELECTRICAL DISTRIBUTION; and (b) based on those predictions, use procedures to correct, control, or mitigate the conseq...: (CFR: 41.5 / 45.6) Loss of ventilation during charging (2.6/2.9)

QUESTION: 49 21329 (1 point(s))

The plant is operating at 100% power with the following condition:

- ➢ Battery charge in progress following the replacement of a cell in the **Division II** 125/250 VDC station battery.
- ➢ EF-C-1C, BATT RM EXH FAN is out of service due to a motor failure.

MCC-LX is lost.

What impact does this have on the batteries?

What action is required?

a. Battery room ventilation is lost with possible hydrogen buildup in the battery room. Restore a battery room fan to service within 7 days.

b. Battery room ventilation is lost with possible hydrogen buildup in the battery room. Install and operate portable ventilation equipment for the battery rooms.

c. Power is lost to the Division II Battery Charger with eventual loss of battery voltage. Align the Division II battery charge to its alternate supply.

d. Power is lost to the Division II Battery Charger with eventual loss of battery voltage. Declare the Division II Battery inoperable.

ANSWER: 49 21329

b. Battery room ventilation is lost with possible hydrogen buildup in the battery room. Install and operate portable ventilation equipment for the battery rooms.

The loss of MCC-LX results in the loss of the only available Battery room ventilation fan.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
50	1507	02	03/31/2003	10/01/2005	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	COR0020802, DG Load Sequencing

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	
COR0020802	Diesel Generators
3.3.5.1	Emergency Core Cooling System (ECCS) Instrumentation
(B)(8)	Components, capacity, and functions of emergency systems.

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: 50 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 starts;
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: 50 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.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
51	2650	01	08/13/2005	10/01/2005	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	COR0012402, Cause Effect relationship between instrument air and cooling water to the compressors.

Related Lessons
COR0012402 OPS TURBINE EQUIPMENT COOLING SYSTEM

Related Objectives	
COR0012402001040D	Describe the TEC design feature(s) and /or interlock(s) that provide for the following: Air compressor Cooling water flow control
COR0012402001070D	Predict the consequences the following would have on the TEC system: Loss of power or air to the AOVS of the air compressors

Related References	
(B)(4)	Secondary coolant and auxiliary systems that affect the facility.

Related Skills (K/A)	
300000.K1.04	Knowledge of the connections and / or cause effect relationships between INSTRUMENT AIR SYSTEM and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) Cooling water to compressor (2.8/2.9)

QUESTION: 51 2650 (1 point(s))

Upon a loss of air to the AOVs controlling cooling water to the station air compressors, what will be the resultant cooling water lineup?

a. 1A and 1B air compressors will be lined up to REC, 1C air compressors will be lined up to TEC.

b. 1A and 1C air compressors will be lined up to REC, 1B air compressor will be lined up to TEC.

c. 1B and 1C air compressors will be lined up to REC, 1A air compressor will be lined up to TEC.

d. 1B and 1C air compressors will be lined up to TEC, 1A air compressor will be lined up to REC.

ANSWER: 51 2650

a. 1A and 1B air compressors will be lined up to REC, 1C air compressor will be lined up to TEC.

Upon loss of air or power, the TEC/REC isolation valves will fail such that A and B compressor will align to REC and C compressor will align to TEC.

Distractors:

b. is incorrect because 1C air compressors will be lined up to TEC not REC and 1B air compressor will be lined up to REC not TEC.

c. is incorrect because 1C air compressors will be lined up to TEC not REC, 1A air compressor will be lined up to REC not TEC.

d. is incorrect because 1C air compressors will be lined up to TEC not REC.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
52	21127	00	09/01/2004	10/01/2005	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0021902, Loss of REC Effect on RHR pump Operation

Related Lessons
COR0021902 REACTOR EQUIPMENT COOLING

Related Objectives	
COR0021902001060I	Given a specific REC malfunction, determine the effect on any of the following: RHR pumps

Related References	
5.2REC	Loss Of REC
(B)(10)	Administrative, normal, abnormal, and emergency operating procedures for the facility.

Related Skills (K/A)	
400000.K3.01	Knowledge of the effect that a loss or malfunction of the CCWS will have on the following: (CFR: 41.7 / 45.6) Loads cooled by CCWS (2.9/3.3)

QUESTION: 52 21127 (1 point(s))

A loss of all REC occurs and cannot be recovered.

- ➢ No forced air cooling can be brought online in any reactor building quad.

What is the **MAXIMUM** number of RHR pumps can be used **continuously** for longer than **20 minutes** if conditions do not change?

a. One pump running in one RHR loop.

b. One pump running in each RHR loop.

c. One pump running in one loop and two pumps running in the other loop.

d. Two pumps running in each loop.

ANSWER: 52 21127

b. One pump running in each RHR loop.

The loss of REC cooling prevents the long time (20 minutes) concurrent operation of both pumps in a loop. The maximum number of pumps operating would be one in each RHR loop.

Distractors:

a. is incorrect because one pump in each loop could be operated this in not the maximum asked for in the question.

c. is incorrect because this would be two pump running in one loop this is not allowed with a loss of REC.

d. is incorrect because this would be two pump running in one loop this is not allowed with a loss of REC.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
53	21331	00	08/06/2005	10/01/2005	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	COR0012402, Monitor TEC system indications available in the Control Room.

Related Lessons
COR0012402 OPS TURBINE EQUIPMENT COOLING SYSTEM

Related Objectives	
COR0012402001020G	Describe the interrelationships between the TEC system and the following: Service Water system
COR0012402001040A	Describe the TEC design feature(s) and /or interlock(s) that provide for the following: TEC automatic temperature control.

Related References	
(B)(14)	Principles of heat transfer thermodynamics and fluid mechanics.

Related Skills (K/A)	
400000.A4.01	Ability to manually operate and / or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) CCW indications and control (3.1/3.0)

QUESTION: 53 21331 (1 point(s))

Following heavy rains service water temperature rapidly lowers by 10°F.

How does this change in Service Water temperature affect the TEC system?

TEC system flow...

a. rises and system pressure rises.

b. rises and system pressure lowers.

c. lowers and system pressure rises.

d. lowers and system pressure lowers.

ANSWER: 53 21331

c. lowers and system pressure rises.

The reduced SW temperatures will result in lower TEC system temperatures and therefore greater cooling. The Temperature control valves are in-line not bypasses, so as temperature lowers the TCVs close resulting in a lower system flow and a higher system pressure.

Distractors:

a. is incorrect because system flow lowers.

b. is incorrect because system flow lowers and system pressure rises this would be picked if the candidate held the misconception that temperature control valves controlled bypass flow.

d. is incorrect because system pressure rises.

Source: Modified 1333

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
54	21332	00	08/06/2005	10/01/2005	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	COR0022002, Predict changes in CRD drive water flow for RMCS manipulations.

Related Lessons
COR0022002 OPS REACTOR MANUAL CONTROL SYSTEM

Related Objectives	
COR0022002001090A	Describe the interrelationships between RMCS and/or RPIS, and the following: CRDH

Related References	
(B)(6)	Design, components, and functions of reactivity control mechanisms and instrumentation.
(B)(14)	Principles of heat transfer thermodynamics and fluid mechanics.

Related Skills (K/A)	
201002.A1.01	Ability to predict and/or monitor changes in parameters associated with operating the REACTOR MANUAL CONTROL SYSTEM controls including: (CFR: 41.5 / 45.5) CRD drive water flow (2.8/2.8)

QUESTION: 54 21332 (1 point(s))

Compare the expected drive water flow indicated on panel 9-5 on CRD-FI-305 CRD Drive Water Flow meter for driving and stall flows for inserting and withdrawal of a control rod.

	Driving	**Stall Flow**
a.	Insert > withdraw	Insert ≈ Withdraw
b.	Insert < withdraw	Insert ≈ Withdraw
c.	Insert ≈ Withdraw	Insert >Withdraw
d.	Insert ≈ Withdraw	Insert <Withdraw

ANSWER: 54 21332

a.	Insert > withdraw	Insert ≈ Withdraw

The drive flow instrument measures actual flow going to the drive. Insert flow is greater than withdraw flow because of the size of the under piston area. The stall flow should be approximately the same. This value a function of what flow is driven past the seals.

Distractors:

b. is incorrect because insert flow indicated on the 9-5 flow instrument is greater than withdraw.

c. is incorrect because insert flow indicated on the 9-5 flow instrument is greater than withdraw flow and insert and withdraw flows are approximately equal.

d. is incorrect because insert flow indicated on the 9-5 flow instrument is greater than withdraw flow and insert and withdraw flows are approximately equal.

Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
55	21263	00	10/17/2004	10/01/2005	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
4	3	1	5	Multiple Choice	

Topic Area	Description
Systems	COR0022602, Will the system bypass with steam flow greater than LPAP and feed flow less than LPAP?

Related Lessons
COR0022602 OPS ROD WORTH MINIMIZER

Related Objectives	
COR0022602001060E	Briefly describe the following concepts as they apply to the RWM: Steam flow indication.

Related References	
(B)(6)	Design, components, and functions of reactivity control mechanisms and instrumentation.
(B)(7)	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Related Skills (K/A)	
201006.K6.05	Knowledge of the effect that a loss or malfunction of the following will have on the ROD WORTH MINIMIZER SYSTEM (RWM) (PLANT SPECIFIC): (CFR: 41.7 / 45.7) Steam flow input: P-Spec (Not-BWR6) (2.7/2.7)

QUESTION: 55 21263 (1 point(s))

The plant is operating at 25% power when the "A" Steam Flow transmitter fails **upscale.**

What effect does this failure have on the Rod Worth Minimizer (RWM)?

The RWM . . .

a. is automatically bypassed.

b. enforces the rod sequence with insert and withdraw blocks.

c. generates a RPIS/RWM hardware failure **AND** blocks all control rod movement.

d. indicates out-of-sequence rods but does **NOT** enforce insert and withdraw blocks.

ANSWER: 55 21263

a. is automatically bypassed.

When Reactor power is in the Transition Zone sensed steam flow above 35% auto bypasses the RWM. The failure upscale of one steam flow instrument upscale would cause total sensed steam flow to go to approximately 46%. This is greater than the 35% setpoint therefore the RWM is auto bypassed.

Distractors:

b. is incorrect because the RWM is bypassed. The examinee that held the misconception that feed flow is required to be greater than 35% to bypass the RWM may choose this answer.

c. is incorrect because the RWM is bypassed. This answer would be correct if the initial conditions were just below the LPSP.

d. is incorrect because the RWM is bypassed. The examinee that held the misconception that feed flow is required to be greater than 35% to bypass the RWM may choose this answer.

Modified 21098

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
56	1017	01	01/12/2000	10/01/2005	Licensed Operator	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	COR0012002, RWCU response to initiation of the SLC Pump "A"

Related Lessons
COR0012002 OPS Reactor Water Cleanup

Related Objectives	
COR0012002001140A	Given plant conditions, determine if: RWCU pumps should have tripped

Related References	
(B)(9)	Shielding, isolation, and containment design features, including access limitations.

Related Skills (K/A)	
204000.K4.04	Knowledge of REACTOR WATER CLEANUP SYSTEM design feature (s) and/or interlocks which provide for the following: (CFR: 41.7) System isolation upon-receipt of isolation signals (3.5/3.6)
204000.K1.08	Knowledge of the physical connections and/or cause-effect relationships between REACTOR WATER CLEANUP SYSTEM and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) SBLC (3.7/3.8)

QUESTION: 56 1017 (1 point(s))

Given the following conditions:

- ➢ The Plant was operating normally at 100% power.
- ➢ A transient occurs and ATWS conditions now exist.
- ➢ Standby Liquid Control (SLC) Pump "B" is out of service.

What will be the Reactor Water Cleanup System (RWCU) response to initiation of the SLC Pump "A"?

a. Both RWCU pumps receive a trip signal due to RWCU-MO-15 closing.

b. Both RWCU pumps receive a trip signal due to RWCU-MO-18 closing.

c. Both RWCU pumps receive a trip signal when RWCU flow drops below 95 gpm.

d. "A" RWCU pump will trip when SLC is initiated, "B" RWCU pump must be tripped manually.

ANSWER: 56 1017

a. Both RWCU pumps receive a trip signal due to RWCU-MO-15 closing.

Starting SLC pump "A" initiates closure of RWCU-MO-15. Closure of this valve generates a trip of both RWCU pumps.

Distractors:

b. is incorrect because RWCU-MO-18 does not close with a SLC "A" initiation.

c. is incorrect as the pumps trip on the closure of RWCU-MO-15.

d. is incorrect because both pumps trip.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
57	21333	00	08/06/2005	10/01/2005	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	COR0021502, Implications of a leak in the core d/p monitoring instruments.

Related Lessons
COR0021502 NUCLEAR BOILER INSTRUMENTATION

Related Objectives	
COR0021502001040D	Briefly describe the following concepts as they apply to NBI: Vessel DP measurement
COR0021502001050C	Predict the consequences of the following items on the NBI: Instrument line leakage
COR0021502001060L	Given a specific NBI malfunction, determine effect on any of the following: Core DP monitoring

Related References	
(B)(2)	General design features of the core, including core structure, fuel elements, control rods, core instrumentation, and coolant flow.

Related Skills (K/A)	
216000.K5.04	Knowledge of the operational implications of the following concepts as they apply to NUCLEAR BOILER INSTRUMENTATION: (CFR: 41.5 / 45.3) Vessel differential pressure measurement (2.8/2.9)

QUESTION: 57 21333 (1 point(s))

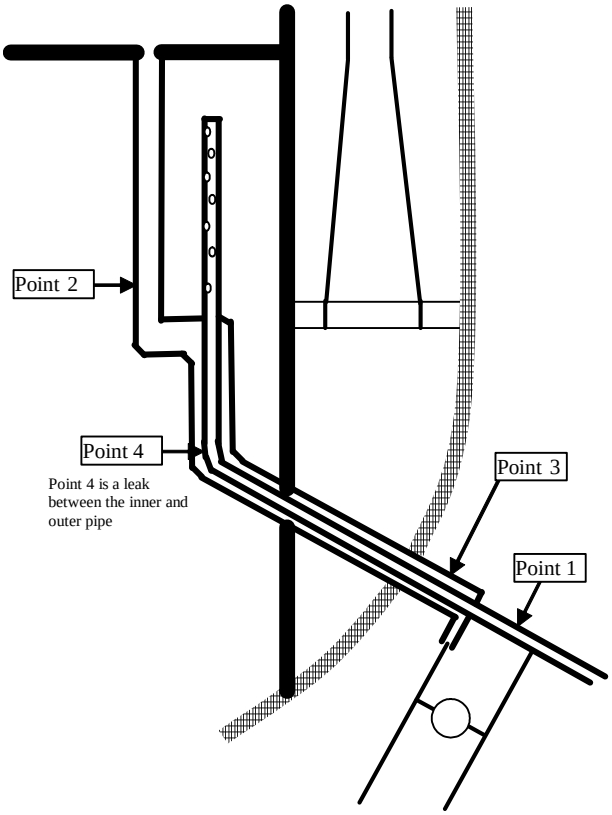
Refer to the drawing of the core plate instrument lines. An instrument line failure at what point would result in a higher than actual indicated core plate differential pressure. (When considering these failure points consider a large failure at each of the locations.)

a. Point 1

b. Point 2

c. Point 3

d. Point 4



ANSWER: 57 21333

c. Point 3

A leak at point 3 would result in less pressure on the low side of the core plate D/P cell. Since this would not affect the high side total sensed D/P would increase.

Distractors:

a. is incorrect because a leak at point 1 would lower the sensed high side pressure and result in a lower core plate D/P.

b. is incorrect because a leak at point 2 would lower the sensed high side pressure and result in a lower core plate D/P.

d. is incorrect because a leak at point 4 would lower the sensed high side pressure and result in a lower core plate D/P.

K/A 216000.K5.04

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
58	3145	02	10/05/2005	12/31/2003	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	INT0070507, T.S. 3.6, What actions are required when SP Temp exceeds 110

Related Lessons
INT0070507 CNS Tech. Spec. 3.6, Containment Systems

Related Objectives	
INT00705070010500	From memory in MODES 1,2, and 3, state the actions required in less than one hour if suppression pool average temperature > 110 degrees F but is either less than or equal to 120 degrees F (LCO 3.6.2.1)

Related References	
3.6.2.1	Suppression pool average temperature

Related Skills (K/A)	
2.2.22	Knowledge of limiting conditions for operations and safety limits. (CFR: 43.2 / 45.2) (3.4/4.1) **EXAM USE ONLY**

QUESTION: 58 3145 (point(s))

Given the following conditions:

- ➢ The plant is operating at 100% power
- ➢ Reactor Core Isolation Cooling (RCIC) is in service for it's quarterly test
- ➢ Initial Suppression Pool Temperature is 92°F
- ➢ Annunciator 9-3-1/C-1, SAFETY/RELIEF VALVE LEAKING alarms
- ➢ The discharge temperature of Safety Relief Valve (SRV) D is determined to be 322°F

What Technical Specifications actions are required?

When Suppression Pool temperature *exceeds* . . .

a. 95°F; immediately shut down RCIC.

b. 105°F; within one (1) hour place the Reactor Mode Switch in SHUTDOWN.

c. 110°F; immediately place the Reactor Mode Switch in SHUTDOWN.

d. 120°F; immediately initiate Boron injection.

ANSWER: 1 3145

c. 110°F; immediately place the Reactor Mode Switch in SHUTDOWN.

REFERENCE: ITS 3.6.2.1

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
59	21335	00	08/06/2005	10/01/2005	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	COR0022302, Knowledge of the power supplies required for containment sprays.

Related Lessons
COR0022302 RESIDUAL HEAT REMOVAL

Related Objectives	
COR0022302001020A	State the electrical power supplies to the following: RHR pump motors
COR0022302001080C	Predict the consequences a malfunction of the following will have on the RHR system: Emergency Diesel Generator
COR0022302001080A	Predict the consequences a malfunction of the following will have on the RHR system: A.C. electrical power (including RPS)

Related References	
(B)(7)	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Related Skills (K/A)	
226001.K2.02	Knowledge of electrical power supplies to the following: (CFR: 41.7) Pumps (2.9*/2.9*)

QUESTION: 59 21335 (1 point(s))

A loss of coolant accident occurred concurrent with a loss all Off Site Power:

- ➢ DG 1 and DG 2 automatically started and loaded their busses.
- ➢ Drywell and torus pressure rose and the crew started RHR loop A in drywell sprays.
- ➢ RHR loop B is placed in Tours spray and suppression pool cooling.
- ➢ Reactor water level is 30" (NR) with HPCI injecting.
- ➢ DG 2 trips and cannot be restarted.

What RHR **pumps** remain available for drywell and/or torus sprays at this time?

a. RHR pumps A and B

b. RHR pumps C and D

c. RHR pumps B and C

d. RHR pumps A and D

ANSWER: 59 21335

a. RHR pumps A and B

RHR pumps A and B are powered by 4160 VAC 1F. This is the only critical switchgear that remains energized following the loss of all offsite power. Therefore the only pumps available to spray containment are RHR pumps A and B.

Distractors:

b. is incorrect because neither C nor D pumps have power.

c. is incorrect because C pump does not have power.

d. is incorrect because D pump den not have power.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
60	21336	00	08/06/2005	10/01/2005	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	COR0010602, Monitor automatic FPC pump trip due to leak in the FPC 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
COR0010602001090C	Describe the FPC design features and/or interlocks that provide for the following: Pump protection
COR0010602001110A	Given plant conditions, determine if: FPC pumps should have tripped

Related References	
(B)(4)	Secondary coolant and auxiliary systems that affect the facility.

Related Skills (K/A)	
233000.A3.02	Ability to monitor automatic operations of the FUEL POOL COOLING AND CLEAN-UP including: (CFR: 41.7 / 45.7) Pump trip (s) (2.6/2.6)

QUESTION: 60 21336 (1 point(s))

A leak develops in the Fuel Pool Cooling (FPC) System. Level is in the skimmer surge tank is lowering at 0.5 ft/minute. If no operator action occurs, what is the expected response of the FPC system?

The operating fuel pool cooling pump...

a. trips on low suction pressure.

b. trips due to low skimmer surge tank level.

c. trips on low flow when the pump loses suction.

d. becomes air bound when the skimmer surge tanks are empty but continues to operate.

ANSWER: 60 21336

b. trips due to low skimmer surge tank level.

Fuel Pool Cooling pump protection is provided by a surge tank low level trip set at elevation 978' 6" (50 ft^3) and a low suction pressure trip of -10 ft. of water. The skimmer surge tank level will occur first resulting in a pump trip.

Distractors:

a. is incorrect because the skimmer surge tanks are located above the pumps and the setpoint of -10 ft of water could never be reached before the pumps trip on low level in the skimmer surge tank.

c. is incorrect because the pumps are not provided with low flow trips and the skimmer surge tank would trip the pump before low flow could.

d. is incorrect becomes the pump would trip on low skimmer surge tank level before the pumps could become air bound.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
61	20660	00	04/15/2004	10/01/2005	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
Systems	COR0020902, DEH Failure Reactor Depressurization

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	
(B)(5)	Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for

Related Skills (K/A)	
241000.K3.02	Knowledge of the effect that a loss or malfunction of the REACTOR/TURBINE PRESSURE REGULATING SYSTEM will have on following: (CFR: 41.7 / 45.4) Reactor pressure (4.2*/4.3*)

QUESTION: 61 20660 (1 point(s))

The plant was operating at 100% power when the **throttle pressure input signal** to the "A" DEH pressure controller (currently the in service controller) fails. This failure resulted in a signal that **slowly and continuously rises.**

What terminates the resulting reactor pressure transient?

The resulting reactor pressure transient is terminated when...

a. a Group 1 isolation occurs as reactor pressure reaches approximately 850 psig.

b. the DEH flow limiter limits maximum flow and stabilizes reactor pressure above 850 psig.

c. the DEH pressure controller "B" assumes control of DEH and controls reactor pressure 5 psig above starting pressure.

d. the DEH load reference limits valve position when load reaches approximately 840 Mw, stabilizing reactor pressure above 850 psig.

ANSWER: 61 20660

a. a Group 1 isolation occurs as reactor pressure reaches approximately 850 psig.

The increase in the throttle pressure input to DEH will result in further opening of the turbine governor valves. This increases steam flow above reactor power resulting in the decreasing pressure. Since the pressure signal has failed DEH gets no feedback from the decreasing reactor pressure. This is further aggravated by the pressure reduction increasing the reactor void content resulting in a power reduction. Pressure decreases until the Group 1 occurs.

Distractors:

b. is incorrect although the flow limiter may limit flow in this case, the flow limiter is set above the current level for reactor power resulting in a mismatch and eventual depressurization below 850 psig.

c. is incorrect because pressure will decrease and controller "B" will not control pressure.

d. depending on the rate of the failure the DEH load reference may be reached but since reactor power and steam flow are mismatched and there is no feedback to DEH, the plant depressurizes.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
62	21337	00	08/06/2005	10/01/2005	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	COR0020202, Cause effect relationship between the condensate system and the feedwater system.

Related Lessons
COR0020202 OPS CONDENSATE AND FEED

Related Objectives	
COR0020202001110A	Briefly describe Condensate and Feedwater operation under the following conditions: Pump Trips
COR0020202001090O	Given a specific Condensate and/or Feedwater malfunction, determine the effect on any of the following: Condensate and/or Feedwater
COR0020202001120B	Given plant conditions, determine if: Reactor Feed Pumps should have tripped

Related References	
(B)(4)	Secondary coolant and auxiliary systems that affect the facility.

Related Skills (K/A)	
259001.K1.05	Knowledge of the physical connections and/or cause-effect relationships between REACTOR FEEDWATER SYSTEM and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) Condensate system (3.2/3.2)

QUESTION: 62 21337 (1 point(s))

The plant was operating with the following conditions:

- ➢ A and B RFP are in service.
- ➢ A and C CBP are running.
- ➢ A and B Condensate pumps are running.
- ➢ Both RR pump Scoop tubes are locked.
- ➢ Reactor power is 68% power

CBP A Trips.

How do the reactor feed pumps respond?

a. Both RFPs continue to run.

b. Both RFPs trip immediately.

c. RFP A trips after about 10 seconds and RFP B continues to run.

d. RFP A trips after about 10 seconds followed by RFP B at 15 seconds.

ANSWER: 62 21337

d. RFP A trips after about 10 seconds followed by RFP B at 15 seconds.

Low RFP suction pressure (260 psig following a 10 and 15 sec. time delay for A and B RFPs respectively). Protects against pump cavitation. The loss of the CBP would cause suction pressure to the feed pumps to lower to below the setpoint for RFP trip. RFP A would trip followed by RFP B.

Distractors:

a. Is incorrect because the RFPs have inadequate suction pressure and they trip.

b. Is incorrect because the there is a 10 second time delay for A RFP and 15 second delay for B RFP.

c. Is incorrect because RFP B would also trip. Even though the loss of RFP A may temporarily raise pressure to B RFP, the loss of A RFP would cause an almost instant rise in RFP B flow.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
63	6028	00	07/28/1999	10/01/2005	Licensed Operator	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	COR0011902 Radwaste/Water Treatment

Related Lessons
COR0011902 Radwaste/Water Treatment System

Related Objectives	
COR0011902001090F	Identify the relationships (physical and/or cause effect) between Radwaste and Water Treatment systems: Residual Heat Removal (RHR) system

Related References	
(B)(4)	Secondary coolant and auxiliary systems that affect the facility.

Related Skills (K/A)	
268000.K1.09	Knowledge of the physical connections and/or cause- effect relationships between RADWASTE and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) ECCS systems (2.6/2.8)

QUESTION: 63 6028 (1 point(s))

Which Radwaste tanks **CAN** the Residual Heat Removal (RHR) system be directed to via the RHR-MO-57 and RHR-MO-67 valves (RHR Discharge to Radwaste)?

a. Waste Surge **Only**

b. Waste Collector and Floor Drain Collector **Only**

c. Floor Drain Collector, Waste Collector and Waste Surge

d. Floor Drain Collector, Floor Drain Sample and Waste Collector

ANSWER: 63 6028

c. Floor Drain Collector, Waste Collector and Waste Surge

The RHR system via RHR-MO-57 and MO-67 can be aligned to the Waste Surge Tank, the Waste Collector Tank or the Floor Drain Collector tank.

Distractors:

a. is incorrect as the Waste Collector Tank and the Floor Drain Collector tank can also receive RHR water.

b. is incorrect as the as the Waste Surge can also accept RHR water via this line.

d. is incorrect because RHR water via MO-57 and 67 cannot be directed to the Floor Drain Sample tank.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
64	21339	00	08/27/2005	10/01/2005	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	COR0011802, Ability to manually trip the RB Vent Rad Monitors to cause an isolation.

Related Lessons
COR0011802 OPS Radiation Monitoring

Related Objectives	
COR001802001050R	Describe the interrelationship between the RM system and the following: Reactor building ventilation system
COR001802001100D	Given a control manipulation, predict and explain the changes to the following Radiation Monitoring systems: Reactor Building Vent Exhaust Plenum radiation monitoring system
COR001802001120E	Given plant conditions related to the Radiation Monitoring system, determine if any of the following should occur: Reactor Building Ventilation Isolation

Related References	
(B)(11)	Purpose and operation of radiation monitoring systems, including alarms and survey equipment.

Related Skills (K/A)	
272000.A4.06	Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) ?Manually trip process radiation monitor logic (2.9/3.2)

QUESTION: 64 21339 (1 point(s))

How does Procedure 2.1.22 direct the manual initiation of a group 6 isolation?

A full Group 6 isolation is initiated manually by placing the Mode switches for RX BLDG VENT RAD MON...

a. CH A to TRIP TEST and then back to OPERATE.

b. CH B to TRIP TEST and then back to OPERATE.

c. CH A and CH B to TRIP TEST and then back to OPERATE.

d, CH B and CH D to TRIP TEST and then back to OPERATE.

ANSWER: 64 21339

c. CH A and CH B to TRIP TEST and then back to OPERATE.

Procedure 2.1.22 states: "To cause full Group 6 Isolation, perform following:

Place Mode switch for RMP-RM-452A, RX BLDG VENT RAD MON CH A, to TRIP TEST

Place Mode switch for RMP-RM-452A to OPERATE.

Place Mode switch for RMP-RM-452B, RX BLDG VENT RAD MON CH B, to TRIP TEST.

Place Mode switch for RMP-RM-452B to OPERATE.

Distractors:

a. is incorrect as this would only cause a half a group 6 isolation.

b. is incorrect as this would only cause a half a group 6 isolation.

d, is incorrect as this would only cause a half an isolation.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
65	21340	00	08/13/2005	10/01/2005	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
Systems	COR0010502, Impact of inadvertent initiation of deluge system on transformer and actions to reset the deluge system.

Related Lessons
COR0010102 AC Electrical Distribution

Related Objectives
NONE

Related References	
(B)(8)	Components, capacity, and functions of emergency systems.
(B)(10)	Administrative, normal, abnormal, and emergency operating procedures for the facility.

Related Skills (K/A)	
286000.A2.07	Ability to (a) predict the impacts of the following on the FIRE PROTECTION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences...: (CFR: 41.5 / 45.6) Inadvertent system initiation (2.9/2.9)

QUESTION: 65 21340 (1 point(s))

Supervisory air to the C Main Transformer deluge system is lost which results in the inadvertent initiation of the deluge system.

What effect does this have on C Main Transformer?

What action(s) is/are required to reset the deluge system?

a. The cooling fans and oil pumps trip.
Restore the Supervisory air, then isolate, drain and reset the deluge system.

b. The cooling fans and oil pumps trip.
Restore the deluge system supervisory air only.

c. Only the cooling fans trip.
Restore the Supervisory air, then Isolate, drain and reset the deluge system.

d. Only the cooling fans trip.
Restore the deluge system supervisory air only.

ANSWER: 65 21340

a. The cooling fans and oil pumps trip.
Restore the Supervisory air, then isolate, drain and reset the deluge system.

The initiation of the deluge system causes the trip of the oil pumps and fans on the Main Transformer. In order to reset the deluge valve it must be isolated to stop the water flow, the system must be drained, supervisory air restored, and then the deluge valve may be reset.

b. is incorrect because once the valve is tripped it must be isolated to stop flow before it can be reset.

c. is incorrect because the oil pumps also trip.

d. is incorrect because the oil pumps also trip and once the valve is tripped it must be isolated to stop flow before it can be reset.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
66	3133	02	07/09/2004	10/01/2005	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	INT0070510, < 1 hour action for inop accumulator in MODE 5 with rod withdrawn

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

Related Objectives	
INT00705100010800	From memory, in MODES 5, state the actions required in less than one hour if one or more withdrawn control rods inoperable (LCO 3.9.5).
INT00705100010100	Given a set of plant conditions, recognize non-compliance with a Section 3.9 LCO.
INT00705100010300	Given a set of plant conditions that constitutes non-compliance with a Section 3.9 LCO determine the ACTIONS that are required.

Related References	
3.9.5	Control rod operability - refueling
10CFR55.43 (2)	Facility operating limitations in the technical specifications and their bases.

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) **EXAM USE ONLY**
2.1.33	Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications. (CFR: 43.2 / 43.3 / 45.3) (3.4/4.0) **EXAM USE ONLY**
2.1.11	Knowledge of less than one hour technical specification action statements for systems. (CFR: 43.2 / 45.13) (3.0/3.8) **EXAM USE ONLY**

QUESTION: 66 3133 (1 point(s))

Given the following conditions:

- ➢ The reactor is in MODE 5.
- ➢ The Reactor Mode Switch is in REFUEL.
- ➢ Rod 06-15 is being withdrawn to collect data for evaluation of excessive friction.
- ➢ Annunciator 9-5-1/G-4, CRD ACCUM LOW PRESS **OR** HIGH LEVEL alarms.
- ➢ The ACCUM light on the Full Core Display for rod 06-15 illuminates.
- ➢ Accumulator pressure for rod 06-15 is 930 psig.

If conditions do not change, what Technical Specifications actions are required ***in less than one hour***?

a. Insert rod 06-15 **only**.

b. Verify shutdown margin is at least 0.38% $\Delta k/k$ **only**.

c. Insert rod 06-15 **AND** place the Reactor Mode Switch in SHUTDOWN.

d. Verify shutdown margin is at least 0.38% $\Delta k/k$ **OR** place the Reactor Mode Switch in SHUTDOWN.

ANSWER: 66 3133

a. Insert rod 06-15 **only**.

LCO 3.1.5 does not apply in MODE 5, 3.9.5 does. No mode switch in shutdown in 3.9.5.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
67	5482	01	07/17/2002	10/01/2005	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	COR0021402, Ability to perform integrated plant procedures.

Related Lessons
COR0021402 OPS MAIN STEAM

Related Objectives	
COR0021402001040H	Describe the Main Steam system design features and/or interlocks that provide for the following: Equalization of pressure across the MSIV's before opening
COR0021402001030H	Describe the interrelationships between the Main Steam System and the following: PCIS

Related References	
COR0021402	Main Steam
2.1.22	Recovering From A Group Isolation
2.2.56	Main Steam System
(B)(10)	Administrative, normal, abnormal, and emergency operating procedures for the facility.

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)
(DO NOT LINK RO OR SRO MATERIAL [other than 2.1.1 and 2.1.4 material] TO THIS K/A. More specific K/As are available.)	

QUESTION: 67 5482 (1 point(s))

A Group 1 isolation has occurred at power. Reactor pressure is 825 psig when it is decided to open the MSIVs.

AFTER the Group 1 isolation is reset, which is the **NEXT** step that must be performed?

a. Open the Inboard MSIVs.

b. Open the Outboard MSIVs.

c. Open Main Steam Drain Valves.

d. Place ALL MSIV control switches in the CLOSE position.

ANSWER: 67 5482

b. Open the Outboard MSIVs.

EXPLANATION OF ANSWER: b. Correct. Open the outboard MSIVs to prevent creating a high dp when equalizing. a. The differential pressure across the Inboard MSIVs must be equalized before opening. d. The stem stated that the Group 1 isolation had been reset so the MSIV control switches would already be in the close position per 2.1.22. c. Opening the drain valves before the outboard isolation valves would cause a large differential pressure across the outboard MSIVs.

REFERENCES: COR0021402, PR 2.2.56, PR 2.1.22

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
68	21342	00	08/13/2005	10/01/2005	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	INT0320136, Where are the controls to scram the reactor when the control room is abandoned?

Related Lessons
INT0320136 CNS Abnormal Procedures (RO) Miscellaneous

Related Objectives	
INT0320136N0N0100	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	
(B)(10)	Administrative, normal, abnormal, and emergency operating procedures for the facility.

Related Skills (K/A)	
2.1.30	Ability to locate and operate components / including local controls. (CFR: 41.7 / 45.7) (3.9/3.4) **NRC EXAM ONLY**

QUESTION: 68 21342 (1 point(s))

The Control Room is abandoned due to toxic gas. The crew did not scram the reactor before leaving the Control Room.

Where and how is the reactor scrammed?

a. From the Cable Spreading Room by de-energizing APRMs.

b. From the RPS MG set rooms by tripping the MG set EPA breakers.

c. From MCC-LX and MCC-TX by opening breaker RPS MG sets supply breakers.

d. From the RPS MG set rooms by tripping the alternate power supply breakers to RPS and the RPS MG set supply breaker.

ANSWER: 68 21342

d. From the RPS MG set rooms by tripping the alternate power supply breakers to RPS and the RPS MG set supply breaker.

5.1ASD directs the operator to the RPS MG set rooms to deenergize RPS by opening the alternate feed to RPS and the MG set supply breaker.

Distractors:

a. is incorrect although this would result in a reactor scram this is not what is directed by the procedure.

b. is incorrect as this is not the action directed by the procedure.

c. is incorrect although this would scram the reactor this is not the method used ion 5.1ASD.

Source New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
69	20517	00	08/06/2005	10/01/2005	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
Administrative	Knowledge of the refueling process.

Related Lessons
NONE

Related Objectives
NONE

Related References	
10.25	Refueling - Core Unload, Reload, and Shuffle
(B)(1)	Fundamentals of reactor theory, including fission process, neutron multiplication, source effects, control rod effects, criticality indications, reactivity coefficients, and poison effects.

Related Skills (K/A)	
2.2.26	Knowledge of refueling administrative requirements. (CFR: 43.5 / 45.13) (2.5/3.7)
2.2.27	Knowledge of the refueling process. (CFR: 43.7 / 45.13) (2.6/3.5)

QUESTION: 69 20517 (1 point(s))

The reactor has been shutdown for 7 days with core fuel shuffling activities in progress. During the shuffle two (2) bundles are maintained around each operable SRM. In addition to the bundles around the SRMs, additional fuel is maintained in the core during the shuffle.

What purpose does this additional fuel serve during the shuffle?

a. Ensure the cooling capacity of the FPC system is not exceeded.

b. Ensures maintenance of the shutdown K_{eff} of the Spent Fuel Storage Pool (SFSP).

c. Provide neutronic coupling of the core to ensure indication of neutron population at the SRM detectors.

d. Limits the I-131 inventory in the Spent Fuel Storage Pool (SFSP) so that the limits of 10CFR100 would not be exceeded during a long duration loss of Fuel Pool Cooling.

ANSWER: 69 20517

c. Provide neutronic coupling of the core to ensure indication of neutron population at the SRM detectors.

Reloading sequences are designed to ensure proper neutron flux monitoring. During core shuffle activities, at least two fuel bundles shall remain around each operable SRM with sufficient fuel remaining in the core at proper locations to ensure adequate core coupling.

Distractors:

a. is incorrect because sufficient cooling capacity exists with RHR to cool the pool if the core were discharged.

b. is incorrect as the spent fuel pool would have remain shutdown with the entire core offloaded.

d. is incorrect as this is not a consideration for leaving fuel in the vessel.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
70	20532	0	04/07/2004	12/31/2003	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
Administrative	Delegation of Authority to Authorize the Performance of Surveillances.

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

Related Objectives	
INT032010100G010I	Discuss the following as described in Administrative Procedure 0.26, Surveillance Program: Surveillance test authorization

Related References	
0.26	Surveillance Program

Related Skills (K/A)
2.2.12 Knowledge of surveillance procedures. (CFR: 41.10 / 45.13) (3.0/3.4) **NRC EXAM and 0.26 lessons ONLY**

QUESTION: 70 20532 (1 point(s))

Who may the Shift Manger delegate, if anyone, the authority to authorize performance of surveillance tests?

a. The CRS **only**.

b. WCC SRO **only**.

c. The WCC SRO **or** the CRS.

d. Authority to authorize performance **cannot** be delegated.

ANSWER: 3 20532

c. The WCC SRO **or** the CRS.

Procedure 0.26 states that the authority to authorize performance of the test may be delegated to the CRS or the WCC SRO as the SM deems necessary.

a. is incorrect because the WCC SRO can also be delegated.

b. is incorrect because the CRS can also be delegated.

d. is incorrect because the authority to authorize performance can be delegated to the WCC SRO or the CRS.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
71	20073	00	12/10/2003	10/01/2005	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
Administrative	INT0320115, Admin Rad Proc.-Planned Special Exposure allowed

Related Lessons	
INT0320115	OPS CNS Administrative Procedures Radiation Protection and Chemistry Procedures (Formal Classroom/Pre-OJT Training)

Related Objectives	
INT0320115C0C010A	Discuss the following as described in Administrative Procedure 0.ALARA.7, Planned Special Exposure: Precautions and limitations

Related References	
0.ALARA.7	Planned Special Exposure
9.ALARA.1	Personnel Dosimetry and Occupational Radiation Exposure Program
(B)(12)	Radiological safety principles and procedures.

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)
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: 71 20073 (1 point(s))

During the current calendar year a Station Operator had received a **Planned Special Exposure** (PSE) of 3500 mRem (TEDE) and an **Occupational Exposure** of 850 mRem (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. 100 mRem

b. 150 mRem

c. 250 mRem

d. 500 mRem

ANSWER: 71 20073

b. 150 mRem

Per 0.ALARA.7, step 1.8 on p. 9, "The dose from a PSE is not to be considered in controlling future occupational dose of the affected individual(s) under the limits listed in 10CFR20.1201; however, it shall be included in evaluations of future proposed PSEs."

Per 9.ALARA.1, step 6.2.1.1 on p. 13, "Authorization to exceed 1,000 mrem on-site requires written approval of the individual's Department Supervisor, the ALARA Supervisor, and shall be documented on the CNS RP-9."

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
72	16473	01	07/08/2003	10/01/2005	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
Administrative	INT0320115, Ability to guard against excessive personnel exposure.

Related Lessons	
INT0320115	OPS CNS Administrative Procedures Radiation Protection and Chemistry Procedures (Formal Classroom/Pre-OJT Training)

Related Objectives	
INT0320115E0E0100	Discuss the precautions and limitations associated with Radiation Work Permits (RWP's) as described in Radiological Protection Procedure 9.ALARA.4, Radiation Work Permits.

Related References	
0.ALARA.1	CNS ALARA Program
3.14	Temporary Shielding
9.ALARA.1	Personnel Dosimetry and Occupational Radiation Exposure Program
9.ALARA.5	ALARA Planning and Controls
(B)(12)	Radiological safety principles and procedures.

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: 72 16473 (1 point(s))

A job is being planned in a high radiation area. Consideration is being given to installing temporary shielding and to using a special tool that will allow the job to be performed further from the source, but will increase the time required to complete the job. The temporary shielding will require a total dose of 1R to install and remove.

The following estimates are available:

- OPTION 1: The job can be performed in a field of 2.4 R/hr in one hour without temporary shielding and without the special tool.
- OPTION 2: The job can be performed in a field of 500 mR/hr in one hour with temporary shielding and without the special tool.
- OPTION 3: The job can be performed in a field of 600 mR/hr in two hours without temporary shielding and with the special tool.
- OPTION 4: The job can be performed in a field of 120 mR/hr in two hours with temporary shielding and with the special tool.

In order to maintain the station's dose as low as reasonably achievable (ALARA), which OPTION should be performed?

a. OPTION 1

b. OPTION 2

c. OPTION 3

d. OPTION 4

ANSWER: 72 16473

c. OPTION 3

0.ALARA.1: 2.2, 8.2.1 (requires application of generic fundamentals for radiation protection to determine which option is ALARA). Choice "c" results in 1.2 REM.

Distractors:
a. Total dose is 2.4R.
b. Total dose is 1.5 R (.5R to do the work and 1R to install and remove the shielding)
d. Total dose is 1.24 R (.24 R to do the work and 1R to install and remove the shielding)

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
73	19322	01	08/07/2003	10/01/2005	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	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.
INT00705070010100	Given a set of plant condtions, recognize non-compliance with a Chapter 3.6 LCO.

Related References	
(B)(13)	Procedures and equipment available for handling and disposal of radioactive materials and effluents.

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: 73 19322 (1 point(s))

The plant is in Mode 3, following a shutdown due to increased drywell leakage. The Drywell is being de-inerted to facilitate entry for inspection.

What Technical Specification requirements exist regarding the usage of the 24 inch vent valves, PC-MO-230, TORUS EXH INBD ISOL VLV and PC-MO-231, DW EXH INBD ISOL VLV for this operation?

a. **Neither** 24 inch vent valve may be used at this time to ensure primary containment integrity is maintained.

b. **Both** 24 inch vent valves may be used to facilitate a quick de-inerting of the drywell and torus.

c. **Only** one 24 inch vent valve may be used with one SGT train to minimize the offsite release rates during de-inerting.

d. **Only** one 24 inch vent valve may be used with one SGT train to ensure at least one SGT system is operable if a LOCA occurred.

ANSWER: 73 19322

d. **Only** one 24 inch vent valve may be used with one SGT train to ensure at least one SGT system is operable if a LOCA occurred.

With the plant in MODE 3, only one 24 inch vent path may be aligned to a single SGT train. This provides the other train for use if a DBA LOCA occurred causing damage to the operating train.

a. is incorrect. PC Technical Specifications and 2.2.60 allow the use of a single 24 inch vent path for de-inerting.

b. is incorrect. A note in 2.2.60 states that using the 24 inch vent valves will facilitate a quick de-inerting of the drywell and torus; however, operation is restricted to a single line in MODE 3.

c. is incorrect. Though operating a single train will reduce the release rates this is not the reason for restriction of a single line for venting.

References:2.2.60 Caution at Step 10.3 and Tech Spec Bases 3.6.1.3.1

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
74	12431	00	03/06/2001	10/01/2005	Licensed Operator	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
Integrated Plant	INT0090101, Lowest Total Hydrogen generation during accident

Related Lessons
INT0090101 DEGRADED CORE

Related Objectives	
INT0090101001030E	Describe the characteristics of hydrogen generation due to in-vessel metal-water reaction including: Effects of core reflood rates on the amount of hydrogen produced.

Related References	
10CFR55.41	Written examinations: Operators
(B)(8)	Components, capacity, and functions of emergency systems.
(B)(10)	Administrative, normal, abnormal, and emergency operating procedures for the facility.

Related Skills (K/A)	
2.4.18	Knowledge of the specific bases for EOPs. (CFR: 41.10 / 45.13) (2.7/3.6)

QUESTION: 74 12431 (1 point(s))

The plant had been operating for an extended period of time at power when an accident occurred that resulted in the loss of **ALL** injection capability to the RPV and the eventual decrease in reactor water level to below the bottom of the fuel.

Which of the following would produce the **lowest *TOTAL*** hydrogen generation?

Restoration and injection from . . .

a. SLC at a flow rate of 43 gpm.

b. CRD at a flow rate of 150 gpm.

c. Condensate at a flow rate of 2500 gpm.

d. LPCI at a flow rate of 7000 gpm.

ANSWER: 74 12431

d. LPCI at a flow rate of 7000 gpm.

The total amount of hydrogen generated from the higher injection flow rate is much less than the amount generated at lower flow rates due to the earlier quenching of the core material. For a brief period, the higher flow rate could result in a higher RATE of production but TOTAL will be the least for the higher flow rate and earlier quenching.

DISTRACTORS:
a, b, c - all lower flow than "d" which will result in higher total hydrogen generation.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
75	19121	02	03/15/2002	10/01/2005	Licensed Operator	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	INT0320134, CNS Abnormal Procedures (RO) - Fire

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

Related Objectives	
INT0320134G0G0100	Given plant condition(s), determine from memory if a Main Turbine trip is required due to the event(s).

Related References	
5.4FIRE-SD (B)(10)	Fire Induced Shutdown From Outside Control Room Administrative, normal, abnormal, and emergency operating procedures for the facility.

Related Skills (K/A)	
2.4.27	Knowledge of fire in the plant procedure. (CFR: 41.10 / 43.5 / 45.13) (3.0/3.5)

QUESTION: 75 19121 (1 point(s))

Given the following conditions:

- ➢ The Plant is operating at 100% power.
- ➢ A fire in the Cable Spreading Room has been reported **AND** the Fire Brigade is on the scene.
- ➢ The actions of 5.4FIRE, "General Fire Procedure" **AND** 5.4POST-FIRE, "Post-Fire Operational Information" are being performed.
- ➢ The Control Room Supervisor enters 5.4FIRE-SD, "Fire Induced Shutdown From Outside Control Room" as directed by procedure.

Based on these conditions, which choice below describes a condition that will dictate a Turbine Trip be initiated per 5.4FIRE-SD, "Fire Induced Shutdown From Outside Control Room"?

A Turbine Trip should be performed . . .

a. prior to leaving the Control Room, **IF** possible.

b. **ONLY** for a confirmed fire in the Control Room.

c. **ONLY IF** the fire in the Cable Spreading Room is confirmed.

d. **ONLY IF** unexpected DEH/Turbine related actuations occur.

ANSWER: 75 19121

a. prior to leaving the Control Room, **IF** possible.

REFERENCE: PR 5.4FIRE-SD, "Fire Induced Shutdown From Outside Control Room"

Foils:

b. Required prior to leaving if possible, irrespective of fire location.
c. Not a reason to trip the turbine.
d. Not a reason to trip the turbine.

THIS QUESTION WAS USED ON THE 2002 CNS NRC EXAMINATION AS RECORD # 3175.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
76	13705	02	08/20/2005	10/01/2005	NRC Style Question	RO: SRO: NLO:	N Y N

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

Topic Area	Description
Abnormal/Emergency Procedures	INT0320131, Assessment of electric plant conditions to determine lineup and selection of appropriate procedure.

Related Lessons
INT0320131 CNS Abnormal Procedures (RO) Electrical

Related Objectives	
INT0320131S0S0100	Given plant condition(s), determine from memory the appropriate Abnormal/Emergency Procedure(s) to be utilized to mitigate the event(s).

Related References	
5.3SBO	Station Blackout
10CFR55.43 (5)	Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency sanitations.
5.3EMPWR	Emergency Power
5.3NBPP	No Break Power Panel Failure

Related Skills (K/A)	
295003.AA2.04	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) System lineups (3.5/3.7)

QUESTION: 76 13705 (1 point(s))

The plant is operating at 100% power. Diesel Generator #2 is tagged out for maintenance to repair a faulty fuel oil line.

Severe weather causes the loss of the Emergency Transformer AND the Startup Transformer. One (1) minute later, a reactor scram occurs.

What procedure(s) should be entered?

a. 5.3SBO **ONLY**.

b. 5.3EMPWR **ONLY**.

c. 5.3SBO **AND** 5.3NBPP.

d. 5.3EMPWR **AND** 5.3NBPP.

ANSWER: 76 13705

b. 5.3EMPWR **ONLY**.

EXPLANATION: DG #1 will pick-up 4160V Bus F. Entry condition of 5.3SBO requires ALL 4160V Buses de-energized. NBPP remains energized from 250 VDC Bus A therefore, 5.3NBPP is not required to be entered.

Source: Direct

SRO Justification: 10CFR55.43(b)(5) SRO assessment of facility conditions and selection of appropriate procedures.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
77	21344	00	08/20/2005	10/01/2005	NRC Style Question	RO: SRO: NLO:	N Y N

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

Topic Area	Description
Administrative	INT0320203, Asses DC failure and determine notification requirements

Related Lessons	
INT0320103	CNS Administrative Procedures Conduct of Operations and General Alarm Procedures (Formal Classroom/Pre-OJT Training)
INT0320203	CONDUCT OF OPERATIONS PROCEDURES (SRO)
GEN0030107	NOTIFICATION

Related Objectives
NONE

Related References	
10CFR55.43 (5)	Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situation.

Related Skills (K/A)	
2.4.30	Knowledge of which events related to system operations/status should be reported to outside agencies. (CFR 43.5 / 45.11) (2.2/3.6)

QUESTION: 77 21344 (1 point(s))

The plant is operating at power when a failure causes the loss of 250 VDC SR "B". The bus is not expected to be recovered for 10 days.

What is the **most restrictive** report that is required?

a. 1 hour report

b. 4 hour report

c. 8 hour report

d. 24 hour report

ANSWER: 77 21344

c. 8 hour report

The loss of 250 VDC SR B would prevent HPCI from injecting. Failure of a single train safety system such that it cannot perform its safety function is reportable. This includes HPCI and CREFS.

Distractors:

a. is incorrect as no criteria for a 1 hour report are present.

b. is incorrect as no criteria are met for a 4 hour report.

d. is incorrect because the most restrictive report is the 8 hour report.

Source: New

SRO Only Justification: Only SRO personnel make reportability determination. 10CFR55.43(b)(5)

Provide the Candidate with Procedure 2.0.5.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
78	21345	00	10/26/2005	10/01/2005	NRC Style Question	RO: SRO: NLO:	N Y N

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

Topic Area	Description
Abnormal/Emergency Procedures	INT0320123, Asses plant conditions and determine required actions for FW temperature following a Turbine trip

Related Lessons
INT0320123 CNS Abnormal Procedures (RO) Reactivity

Related Objectives

Related References	
10CFR55.43 (5)	Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Related Skills (K/A)	
295005.AA2.06	Ability to determine and/or interpret the following as they apply to MAIN TURBINE GENERATOR TRIP: (CFR: 41.10 / 43.5 / 45.13) Feedwater temperature. (2.6/2.7)

QUESTION: 78 21345 (1 point(s))

A plant startup is in progress with power at 24% when an inadvertent turbine trip occurs. After the turbine trip the Reactor Operator took action to lower reactor power. When conditions stabilized the following plant conditions were noted:

- ➢ Reactor power is 27% and slowly rising.
- ➢ Feedwater temperature is 200°F and slowly falling.

What is required?

a. Enter 2.4RXPWR and scram the reactor.

b. Enter 2.4RXPWR and lower power as required to terminate the power rise.

c. Enter 2.4EX-STM and direct power be reduced to 24% per Procedure 2.1.10.

d. Enter 2.4EX-STM and restore feed water temperature to normal within 4 hours.

ANSWER: 78 21345

c. Enter 2.4EX-STM and direct power be reduced to 24% per Procedure 2.1.10.

Entry into 2.4EX-STM is required due to the lowering feed water temperature. The procedure requires power be lowered per 2.1.10 to where prior to the reduction in FW temperature.

Distractors:

a. is incorrect because entry into 2.4RXPWR is not required because the cause of the power rise is known and a scram is not required.

b. incorrect because entry into 2.4RXPWR is not required because the cause of the power rise is known.

d. is incorrect because feedwater temperature restoration is required within 2 hours.

Source: New
SRO Only Justification: 10CFR55.43(b)(5)

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
79	21347	00	08/20/2005	10/01/2005	NRC Style Question	RO: SRO: NLO:	N Y N

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

Topic Area	Description
Abnormal/Emergency Procedures	COR0011702 , What is the status of the safety related systems needed for level and pressure control after the loss of all air compressors.

Related Lessons
INT0320136 CNS Abnormal Procedures (RO) Miscellaneous COR0011702 Plant Air

Related Objectives	
COR0011702001030G	Describe the interrelationships between the Plant Air system and the following: Main Steam
COR0011702001030I	Describe the interrelationships between the Plant Air system and the following: Reactor Feedwater
COR0011702001030B	Describe the interrelationships between the Plant Air system and the following: Instrument Air

Related References	
10CFR55.43 (5)	Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Related Skills (K/A)	
295019.AA2.02	Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: (CFR: 41.10 / 43.5 / 45.13) Status of safety-related instrument air system loads (see AK2.1 - AK2.19) (3.6/3.7)

QUESTION: 79 21347 (1 point(s))

The plant is operating at rated power a rupture results in an immediate and complete loss of instrument air.

What action will be required?

a. Enter EOP-1A and use HPCI and RCIC for level control.

b. Enter EOP-1A and use Condensate and Feed with RFP minimum flow valves manually closed for level control

c. Enter 2.1.4 and use HPCI and RCIC for level control.

d. Enter 2.1.4 and use Condensate and Feed with RFP minimum flow valves manually closed for level control.

ANSWER: 79 21347

a. Enter EOP-1A and use HPCI and RCIC for level control.

Entry into EOP-1 will be required following the MSIV closure (high pressure) or due to the loss of feed (low level). The loss of instrument air results in the loss of control of normal plant level control and pressure control systems. The effects of a loss of instrument air are very broad. Although HPCI and RCIC have components that use air they still function.

Distractors:
b. is incorrect because the feed pumps no longer have a steam supply due to the MSIV closure.
c. is incorrect because the loss of air will eventually lead to a condition that requires a scram and therefore entry into EOP-1A.
d. is incorrect because entry into EOP-1A will be required and the feed pumps no longer have supply steam due to the MSIV closure.

Source: Modified INPO 25895
SRO Justification: Assessment of plant conditions, selection of the appropriate procedures based on that assessment and direction of appropriate mitigating actions. 10CFR55.43

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
80	19335	01	07/31/2003	10/01/2005	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 Plan	GEN0030401, SRO classify a dropped bundle per 5.7.1

Related Lessons
GEN0030401 Emergency Plan for Licensed Operators MCR0010101 On-Shift Emergency Director

Related Objectives	
GEN0030401C0C050E	Concerning event classification: Given a copy of EPIP 5.7.1 and hypothetical abnormal plant symptoms, indications, or events, determine any and all EALs which have been exceeded and specify the appropriate emergency classification.

Related References	
10CFR55.43 (4)	Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

Related Skills (K/A)	
295023.AA2.05	Ability to determine and/or interpret the following as they apply to REFUELING ACCIDENTS: (CFR: 41.10 / 43.5 / 45.13) ?Entry conditions of emergency plan (3.2/4.6*)

QUESTION: 80 19335 (1 point(s))

The following plant conditions exist:

- ➢ Refueling activities are in progress.
- ➢ An irradiated fuel bundle is dropped in the cattle chute.
- ➢ REFUEL AREA HIGH RAD, 9-3-1/A-10 is in alarm (both Ronan 1448 and 1449).

Which one of the following describes the **MINIMUM** required Emergency Classification for this event?

a. Unusual Event

b. Alert

c. Site Area Emergency

d. General Emergency

ANSWER: 80 19335

b. Alert

An ALERT should be declared per EAL 3.2.1.

a. is incorrect. Conditions are met for an ALERT.

c. and d. are incorrect. Only a single bundle has been dropped. Elevation to a higher level above an ALERT would require major fuel damage defined as more than ten irradiated fuel bundles.

Reference: 5.7.1 EAL 3.2.1

SRO Only Justification: Only SRO personnel make EP classifications.

<u>Provide Procedure 5.7.1 to the candidate.</u>

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
81	21806	00	10/05/2005	12/31/2003	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
Administrative	INT0320103, What is the type of report that is required for injection of ECCS?

Related Lessons	
INT0320103	CNS Administrative Procedures Conduct of Operations and General Alarm Procedures (Formal Classroom/Pre-OJT Training)

Related Objectives	
INT032010300F010A	Procedure 2.0.5, "Reports to NRC Operations Center" - Discuss the following as described in Procedure 2.0.5, "Reports to NRC Operations Center: Purpose of procedure along with the differences between Immediate Emergency and Non-Emergency notifications.
INT032010300F010D	Procedure 2.0.5, "Reports to NRC Operations Center" - Discuss the following as described in Procedure 2.0.5, "Reports to NRC Operations Center: Instruction section of procedure.

Related References	
2.0.5	Reports to NRC Operations Center

Related Skills (K/A)	
2.4.30	Knowledge of which events related to system operations/status should be reported to outside agencies. (CFR 43.5 / 45.11) (2.2/3.6)

QUESTION: 81 21806 (1 point(s))

A Reactor Feedwater loss has caused RPV water level to drop to -45" WR and the required automatic actions occurred as designed to return water level into the normal band. Based on this event what type of notification is required, if any?

a. None.

b. A four hour per 10CFR50.72 Non-Emergency Report.

c. An eight hour per 10CFR50.72 Non-Emergency Report.

d. An LER per 10CRF50.73, Licensee Event Report System.

ANSWER: 81 21806

b. A four hour per 10CFR50.72 Non-Emergency Report.

Explanation of answer:

Procedure 2.0.5, REPORTS TO NRC OPERATIONS CENTER Attachment 2 states that a 4 Hour Report per 50.72(b)(2)(iv)(A) Any event that results or should have resulted in ECCS discharge into the Reactor Coolant System as a result of a valid signal. This includes manual action to mitigate the consequences of an event, i.e., starting HPCI in response to a loss of feedwater, is required.

Distractors

a. At -42 HPCI and RCIC should have started and injected into the vessel to restore water level, which requires a Notification to the NRC iaw 10CFR50.72

c. The correct time limit for the notification for a HPCI initiation is 4 hours

d. 10CFR50.72 governs this type of event and a four hour report would be made.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
82	21807	00	10/05/2005	12/31/2003	NRC Style Question	RO: SRO: NLO:	N Y N

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

Topic Area	Description
Emergency Operating Procedures	INT0080611, What do you do when RPV Level Instruments are restored?

Related Lessons	
INT0080612	OPS EOP FLOWCHART 7B - RPV FLOODING (FAILURE-TO-SCRAM)

Related Objectives	
INT00806120010400	Given plant conditions and EOP Flowchart 7B, RPV FLOODING (FAILURE TO SCRAM) and REACTOR POWER (FAILURE TO SCRAM), determine required actions.
INT00806120010500	Given plant conditions and EOP flowchart 7B, RPV FLOODING (FAILURE TO SCRAM) and REACTOR POWER (FAILURE TO SCRAM), state the reasons for the actions contained in the steps.

Related References	
EOP 7B	RPV Flooding Failure To Scram

Related Skills (K/A)	
2.4.6	Knowledge symptom based EOP mitigation strategies. (CFR: 41.10 / 43.5 / 45.13) (3.1/4.0)

QUESTION: 82 21807 (point(s))

During the execution of EOPs during an ATWS, water level indication is lost and flooding is entered. Maintenance restores water level indication and the TSC notifies the control room that water level indication is restored.

What action is required?

a. Exit EOP-7B, enter EOP-1A and restore water level +3' to +54" per level leg and maintain the reactor depressurized per the pressure leg.

b. Enter EOP-7A and control level -25' to +54' with outside shroud injection and continue actions in EOP-7B to maintain the reactor depressurized.

c. Exit EOP-7B, Enter EOP-7A and control level -25' to +54' with outside shroud injection and Enter EOP-6A and maintain the reactor depressurized.

d. Enter EOP-1A and maintain the reactor depressurized in the pressure leg and Enter EOP-7A and control level -25' to +54' with outside shroud injection

ANSWER: 82 21807

c. Exit EOP-7B, Enter EOP-7A and control level -25' to +54' with outside shroud injection and Enter EOP-6A and maintain the reactor depressurized.

The 4th override contained in EOP-7B FS/F-1 states that if water level can be determined and the it has not been determined that the reactor remain shutdown under all conditions without boron then exit and go to EOP-6A and 7A for pressure and level control actions.

Distractors:
a. is incorrect because EOP-1A actions are not taken when the reactor is Not shutdown under all conditions without boron.
b. is incorrect as EOP-7B is exited and EOP-7B actions are no longer taken.
d. is incorrect because EOP-1A actions are not taken when the reactor is Not shutdown under all conditions without boron.

Source: NEW

SRO Only Justification Assessment of plant conditions and selection of appropriate procedures and actions based on that assessment.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
83	21348	00	08/20/2005	10/01/2005	NRC Style Question	RO: SRO: NLO:	N Y N

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

Topic Area	Description
Technical Specifications, ODAM, TRM	INT0070501, Determine if a Safety Limit was violated during a control rod drop accident.

Related Lessons
INT0070501 OPS Introduction to Technical Specifications

Related Objectives	
INT00705010010900	State the actions which must be performed should a Safety Limit violation occur at CNS.

Related References	
10CFR55.43 (2)	Facility operating limitations in the technical specifications and their bases.
2.1	SLs
2.1.1	Reactor SL's

Related Skills (K/A)	
295014.AA2.05	Ability to determine and/or interpret the following as they apply to INADVERTENT REACTIVITY ADDITION: (CFR: 41.10 / 43.5 / 45.13) ?Violation of safety limits (4.2*/4.6*)

QUESTION: 83 21348 (1 point(s))

A plant startup and heatup is in progress with the following conditions:

- ➢ Reactor Power is 4% and steady.
- ➢ Reactor water level is 35".
- ➢ Reactor Pressure is 700 psig.

An inadvertent positive reactivity addition occurs resulting in a reactor scram. The following parameters were observed during the accident:

- ➢ Reactor power peaked at 28% power.
- ➢ Reactor level remained at 35"
- ➢ Reactor pressure peaked at 725 psig.

What is the HIGHEST level of authority required in order commence a reactor startup?

a. NRC

b. SORC

c. Operations Manager

d. Operations Supervisor

ANSWER: 83 21348

a. NRC

During the accident reactor power increased above 25%. With steam dome pressure less than 785 psig the reactor core safety limit is violated. Restart following a safety limit violation is only after NRC review and authorization.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
84	8863	00	08/17/2000	10/01/2005	Licensed Operator	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	INT0080612, FLOWCHART 7B - RPV FLOODING FAILURE-TO-SCRAM

Related Lessons	
INT0080612	OPS EOP FLOWCHART 7B - RPV FLOODING (FAILURE-TO-SCRAM)

Related Objectives	
INT00806120010400	Given plant conditions and EOP Flowchart 7B, RPV FLOODING (FAILURE TO SCRAM) and REACTOR POWER (FAILURE TO SCRAM), determine required actions.

Related References	
10CFR55.43 (5)	Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Related Skills (K/A)	
2.4.6	Knowledge symptom based EOP mitigation strategies. (CFR: 41.10 / 43.5 / 45.13) (3.1/4.0)

QUESTION: 84 8863 (1 point(s))

EOP Flowcharts 6A, "RPV Pressure and Reactor Power (Failure to Scram)" and 7B, "RPV Flooding (Failure To Scram)" are being performed. Boron injection is in progress and 35% of the SLC tank has been injected. The crew has inserted all but 3 control rods which are at notch 24. The Reactor Engineer determines that the Reactor will remain shutdown under all conditions without boron.

What actions are required?

a. Flowchart 7B must be exited and flowchart 1A and 2B are to be entered from the top.

b. Flowchart 7B must be exited and flowchart 2B is to be entered from the top.

c. Continue to perform the actions of flowchart 7B until the steam lines are flooded.

d. Continue to perform the actions of flowchart 7B until all control rods have been inserted.

ANSWER: 84 8863

b. Flowchart 7B must be exited and flowchart 2B is to be entered from the top.

The first override of FS/F-1 directs that 2B be entered when it has been determined that the reactor will remain shutdown under all conditions without boron.

Distractors:

a. is incorrect because 1A cannot be entered until water level is determined.

c. is incorrect even though actions to flood the steam lines would continue they would be directed by 2B not 7B.

d. is incorrect because under this condition 7B is exited before all rods are inserted.

Source: Direct

SRO Only Justification: 10CFR55.43(b)(5)

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
85	21349	00	08/20/2005	10/01/2005	NRC Style Question	RO: SRO: NLO:	N Y N

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

Topic Area	Description
Emergency Operating Procedures	INT0080617, Determine the source of the radiation release to determine required actions.

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	
10CFR55.43 (5)	Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Related Skills (K/A)	
295034.EA2.02	Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT VENTILATION HIGH RADIATION: (CFR: 41.10 / 43.5 / 45.13) Cause of high radiation levels (3.7/4.2*)

QUESTION: 85 21349 (1 point(s))

The plant is operating at near rated power with operations in progress to place the spent fuel cask in the fuel pool. A crane failure resulted in the fuel shipping cask being dropped into the fuel pool damaging several bundles of fuel. In addition to the fuel pool damage some Reactor Building siding was punctured by crane debris.

Reactor building and downwind surveys indicate the following:

- ➢ Offsite dose is projected to be 1.1 Rem TEDE and 5.5 Rem CDE (thyroid).
- ➢ Fuel Pool area is 2000 mrem/hr.
- ➢ Reactor Building 976' level near the SCL tank is 1500 mrem/hr.
- ➢ Reactor Building 958' level near the RWCU precoat area is 1200 mrem/hr.

What action is required?

a. Shutdown per 2.1.4.

b. Shutdown per 2.1.5.

c. Rapid Shutdown and cooldown per 2.1.4.1.

d. Entry into EOP1A and emergency depressurize the reactor.

ANSWER: 85 21349

b. Shutdown per 2.1.5.

Even though offsite release rates are high and RB rad levels are greater than max safe no primary system is discharging outside containment. Therefore step SC-15 requires a shutdown per 2.1.5.

Distractors:

a. is incorrect because a 2.1.5 shutdown is required.

c. is incorrect because a 2.1.5 shutdown is required.

d. is incorrect because emergency depressurization of the reactor is not required since there is no primary system discharging outside primary containment.

<u>Provide EOP-5A with the entry conditions and cautions removed.</u>
<u>Provide EP-5.7.1.</u>

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
86	19346	01	08/11/2003	10/01/2005	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	INT0070509, DG Failure effect on RHR Operability

Related Lessons	
INT0070509	OPS Tech. Spec. 3.8, Electrical Power Systems
INT0070506	OPS Tech. Spec. 3.5, Emergency Core Cooling (ECCS) and Reactor Core Isolation Cooling (RCIC) System
INT0070501	OPS Introduction to Technical Specifications

Related Objectives	
INT00705010010200	Given plant conditions and a Specification, apply the rules of Section 3.0 to determine appropriate actions.
INT00705060010100	Given a set of plant conditions, recognize non-compliance with a Section 3.5 LCO.
INT00705060010300	Given a set of plant conditions that constitutes non-compliance with a Section 3.5 LCO, determine the ACTIONS that are required.
INT00705090010100	Given a set of plant conditions, recognize non-compliance with a Section 3.8 LCO.
INT00705090010300	Given a set of plant conditions that constitutes non-compliance with a Section 3.8 LCO, determine the ACTIONS that are required.

Related References	
3.5.1	ECCS Operating
3.8.1	AC Sources - Operating
10CFR55.43 (2)	Facility operating limitations in the technical specifications and their b

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) **EXAM USE ONLY**
203000.A2.06	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...: (CFR: 41.5 / 45.6) Emergency generator failure (3.8/3.9)

QUESTION: 86 19346 (1 point(s))

Given the following conditions:

- ➢ The plant is operating at rated power.
- ➢ 1/15 at 1600, "C" RHR Pump motor failed.
- ➢ 1/17 at 1400, a bearing failure on DG1 occurs.

IF these problems are NOT corrected, when is the plant required to be in **MODE 3**?

a. 1/18 at 0700

b. 1/19 at 0700

c. 1/20 at 0900

d. 1/21 at 0900

ANSWER: 86 19346

a. 1/18 at 0700

3.8.1.B.2 Declare LPCI pumps supported by #1 DG inop when redundant "C" RHR pump is inop within 4 hours (1800 on Jan 17).

3.5.1.H Enter LCO 3.0.3 immediately (1800 on Jan 17)

3.0.3 Action shall be initiated within one hour to place the unit, as applicable, in:

MODE 2 within 7 hours; (0100 on Jan 18)
MODE 3 within 13 hours; and (0700 on Jan 18)
MODE 4 within 37 hours. (0700 on Jan 19)

Provide TS3.5.1 and TS3.8.1 to the candidate.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
87	21808	00	10/05/2005	12/31/2003	NRC Style Question	RO: SRO: NLO:	N Y N

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

Topic Area	Description
Technical Specifications, ODAM, TRM	INT0070506, What is the basis for HPCI and ADS inoperable?

Related Lessons	
INT0070506	OPS Tech. Spec. 3.5, Emergency Core Cooling (ECCS) and Reactor Core Isolation Cooling (RCIC) System

Related Objectives	
INT00705060010200	Discuss the applicable Safety Analysis in the Bases associated with each Section 3.5 Specification.

Related References	
3.5.1	ECCS Operating

Related Skills (K/A)	
2.2.25	Knowledge of bases in technical specifications for limiting conditions for operations and safety limits. (CFR: 43.2) (2.5/3.7)

LINK ONLY TO TECH SPEC LESSONS & QUESTIONS

QUESTION: 87 21808 (point(s))

What is the basis behind the requirement to enter T.S. 3.0.3 when HPCI and two or more ADS valves are inoperable?

a. The plant is in a condition outside of the accident analyses.

b. A single component failure concurrent with a design basis LOCA could result in the minimum required ECCS equipment not being available.

c. A single active component failure concurrent with a design basis LOCA results in a potential for not having the minimum required ECCS equipment available.

d. A single failure in one of the remaining operable subsystems concurrent with a design basis LOCA may result in the ECCS not being able to perform its intended safety function.

ANSWER: 87 21808

a. The plant is in a condition outside of the accident analyses.

Explanation:

In accordance with Tech Spec Bases 3.5.1 H.1 When multiple ECCS subsystems are inoperable, as stated in Conditions H, the plant is in a condition outside of the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

Distractors:

b. is incorrect because this is the basis behind one ADS valve and one low pressure system being inoperable.

c. is incorrect because this is the basis behind HPCI and RCIC being inoperable at the same time.

d. is incorrect because this is the basis behind one ADS Valve and one Low Pressure ECCS System being inop

Source: New

SRO Only Justification; 10CFR55.43 (b) 2 Facility operating limitations in the technical specifications and their bases.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
88	21190	00	09/25/2004	10/01/2005	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Technical Specifications, ODAM, TRM	INT007-05-01, **Technical Specification Completion Times

Related Lessons
INT0070501 OPS Introduction to Technical Specifications

Related Objectives	
INT00705010010600	Explain the rules for Completion Times and apply these rules to determine the time allowed to complete Required Actions.

Related References	
3.1.7	Standby liquid control (SLC) system

Related Skills (K/A)	
2.2.22	Knowledge of limiting conditions for operations and safety limits. (CFR: 43.2 / 45.2) (3.4/4.1) **EXAM USE ONLY**

QUESTION: 88 21190 (1 point(s))

With the plant operating at 100% power the following events occurred:

- ➢ A SLC pump declared inoperable at 0900 on 10/17.
- ➢ B SLC pump declared inoperable at 1100 on 10/19.
- ➢ A SLC pump is returned to operable at 1500 on 10/19.

Which one of the following is the **maximum time** allowed by Technical Specifications to return the B SLC pump to operable?

a. 0900 on 10/24

b. 0900 on 10/25

c. 1100 on 10/26

d. 1100 on 10/27

ANSWER: 88 21190

b. 0900 on 10/25

Seven days (T.S. 3.1.7) from the initial inoperability plus 24 hours.
a. This answer does not include the 24 hour extension
c. d. These times are based on the inoperability of the B SLC pump.
If situations are discovered that require entry into more than one Condition at a time within a single LCO (multiple Conditions), the Required Actions for each Condition must be performed within the associated Completion Time. When in multiple Conditions, separate Completion Times are tracked for each Condition starting from the time of discovery of the situation that required entry into the Condition.

However, when a subsequent division, subsystem, component, or variable expressed in the Condition is discovered to be inoperable the Completion Time(s) may be extended. To apply this Completion Time extension, two criteria must first be met. The subsequent inoperability:

a. Must exist concurrent with the first inoperability; and
b. Must remain inoperable or not within limits after the first inoperability is resolved.
The total Completion Time allowed shall be limited to the more restrictive of either:

a. The stated Completion Time, as measured from the initial entry into the Condition, plus an additional 24 hours; or
b. The stated Completion Time as measured from discovery of the subsequent inoperability.

Provide TS3.1.7 to the candidate.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
89	21351	00	08/26/2005	10/01/2005	NRC Style Question	RO: SRO: NLO:	N Y N

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

Topic Area	Description
Systems	INT0080606, Predict the effect of high reactor pressure on LLS and SRV valves and based on those predictions determine the action required by EOPs.

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

Related Objectives	
INT00806060011100	Given plant conditions and the EOP Flowchart 6A, RPV PRESSURE/POWER FAILURE-TO-SCRAM, determine required actions.

Related References	
10CFR55.43 (5)	Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Related Skills (K/A)	
239002.A2.06	Ability to (a) predict the impacts of the following on the RELIEF/SAFETY VALVES; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those...: (CFR: 41.5 / 45.6) Reactor high pressure (4.1/4.3*)

QUESTION: 89 21351 (1 point(s))

The plant is at 100% power when an inadvertent group 1 isolation occurs resulting in a rapid rise in reactor pressure. Very little rod motion occurs on the subsequent scram. After the initial pressure transient is over the following are noted:

- ➢ LLS is armed.
- ➢ Reactor power is cycling between 18% and 25% power.

How will the non LLS SRVs respond?

What action, if any is required?

a. All **non LLS** SRVs remain closed.
Enter EOP-6A and 7A and direct SRVs manually opened until pressure drops to 940.

b. All **non LLS** SRVs remain closed.
Enter EOP-1A and verify operation of LLS valves.

c. One or more **non LLS** SRVs cycle on mechanical setpoints.
Enter EOP-6A and 7A and direct SRVs manually opened until pressure drops to 940.

d. One or more **non LLS** SRVs cycle on mechanical setpoints.
Enter EOP-1A and verify operation of LLS valves.

ANSWER: 89 21351

c. One or more **non LLS** SRVs cycle on mechanical setpoints.
Enter EOP-6A and 7A and direct SRVs manually opened until pressure drops to 940.

The reactor power specified is greater than the combined capacities of the LLS valves which results in a high reactor pressure and the cycling of the non-LLS valves. So a prediction can be made that one or two non-LLS valves will cycle. EOP-6A directs the opening of SRVs to reduce pressure to 940 psig if SRVs are cycling.

Distractors:

a. is incorrect because non LLS SRVs will cycle.
b. is incorrect because non LLS SRVs will cycle and action to reduce pressure to 940 psig is required as is entry into EOP6A and 7A.
d. is incorrect because action to reduce pressure to 940 psig is required as is entry into EOP6A and 7A.

Source: New
SRO Only Justification: 10CFR55.43(b)(5)

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
90	21352	00	08/27/2005	10/01/2005	NRC Style Question	RO: SRO: NLO:	N Y N

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

Topic Area	Description
Administrative	INT0320140, Notification requirements for IA system pressure out of the green band.

Related Lessons
INT0320140 CNS Administrative Procedures Operations Instructions

Related Objectives
NONE

Related References	
10CFR55.43 (5)	Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Related Skills (K/A)	
2.1.14	Knowledge of system status criteria which require the notification of plant personnel (such as Reactivity Management Events). (CFR: 43.5 / 45.12) (2.5/3.3)

QUESTION: 90 21352 (1 point(s))

The plant is operating at rated power when conditions require entry into 5.2AIR. The crew took appropriate actions and restored the system to normal and exited 5.2AIR. The plant remained at rated power during the entire event.

What **minimum** notifications are required?

a. Operations Manager, Operations Supervisor

b. Operations Manager, Operations Supervisor and the Plant Manager

c. Plant Manager, Operations Manager and the NRC Resident Inspector

d. NRC resident inspector, Operations Manager and Operations Supervisor

ANSWER: 90 21352

d. NRC resident inspector, Operations Manager and Operations Supervisor

With instrument air pressure out of the green band the crew would enter 5.2AIR and take the appropriate actions. When the crew restored air and an emergency no longer existed OI-12 requires the notification of the Operations Manager, Operations Supervisor and the NRC Resident Inspector.

Distractors:

a. is incorrect because Notification of the NRC Resident Inspector is also required whenever an abnormal or emergency procedure is entered.

b. is incorrect because notification of the Plant Manager is not required and Notification of the NRC Resident is required.

c. is incorrect because notification of the operation Supervisor is required.

SRO Only Justification: SRO personnel determine and make the notifications. 10CFR55.43(b)(5)

Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
91	21353	00	08/20/2005	10/01/2005	NRC Style Question	RO: SRO: NLO:	N Y N

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

Topic Area	Description
Systems	INT0070505, Predict the impacts of a mismatch of RR pump speed and determine corrective actions based on the mismatch.

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	
10CFR55.43 (2)	Facility operating limitations in the technial specifications and their bases.

Related Skills (K/A)	
202001.A2.08	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...: (CFR: 41.5 / 45.6) Recirculation flow mismatch: Plant-specific (3.1/3.4)

QUESTION: 91 21353 (1 point(s))

The plant was operating at 100% power with core flow at 73.5Mlbm/hr.

- ➢ RR pump "A" controller fails and speed runs back to 45%.
- ➢ Core Flow is 42%
- ➢ Operation briefly entered the stability exclusion region which was exited by inserting control rods.

How would this failure affect RR pump flows?

What action is required with regards to thermal limits?

a. RR pump flow mismatch would exceed 10%.
Modify **both** APLHGR and MCPR limits.

b. RR pump flow mismatch be 10% or less.
Modify **only** APLHGR limit is required to be modified.

c. RR pump flow mismatch would exceed 10%.
Modify **only** APLHGR limit is required to be modified.

d. RR pump flow mismatch be 10% or less.
Modify **both** APLHGR and MCPR limits.

ANSWER: 91 21353

a. RR pump flow mismatch would exceed 10%.
Modify **both** APLHGR and MCPR limits.

The flows for the two loops would be mismatched by greater than 10%. With the flows mismatched only one RR loop is considered to be in operation by Technical Specifications. With only one recirculation loop considered in operation, modifications to the APLHGR limits and the MCPR limit is required.

SRO Justification: 10CFR55.43(b)(2)

Provide TS3.4.1 to the candidate.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
92	21354	00	08/20/2005	10/01/2005	NRC Style Question	RO: SRO: NLO:	N Y N

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

Topic Area	Description
Systems	INT0320135, Recognize parameters and parameter trends that are entry conditions for abnormal procedure.

Related Lessons
INT0320135 CNS Abnormal Procedures (RO) - Condensate/Feedwater

Related Objectives	
INT0320135G0G0100	Given plant condition(s), determine from memory the appropriate Abnormal/Emergency Procedure(s) to be utilized to mitigate the event(s).

Related References	
10CFR55.43 (5)	Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Related Skills (K/A)	
2.4.4	Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures. (CFR: 41.10 / 43.2 / 45.6) (4.0/4.3)
LINK ONLY TO EOP/AOP LESSONS/QUESTIONS	

QUESTION: 92 21354 (1 point(s))

Reactor power is at 30% when a power reduction is commenced. As power is lowered to less than 30% reactor water level starts to rise. The power reduction is stopped and the following conditions are noted:

- ➢ Reactor water level is 38" and **slowly rising**.
- ➢ Reactor power is 27% and **steady.**

What procedure is required to be entered?

a. 2.1.5, REACTOR SCRAM

b. 2.4CSCS, INADVERTENT CSCS INITIATION

c. 2.4RXLVL, RPV WATER LEVEL CONTROL TROUBLE

d. 2.4MC-RF, CONDENSATE AND FEEDWATER ABNORMAL

ANSWER: 92 21354

c. 2.4RXLVL RPV WATER LEVEL CONTROL TROUBLE

Entry into 2.4RXLVL is required if reactor water level does not follow power changes.

a. is incorrect as no scram condition exists.

b. is incorrect as no CSCS initiation is present. If RCIC or HPCI had initiated power would not be constant.

d. is incorrect as there is no indication of a pump trip or steam line break.

SRO Only Justification: 10CFR55.43(b)(5)

Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
93	21297	00	07/15/2005	10/01/2005	NRC Style Question	RO: SRO: NLO:	N Y N

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

Topic Area	Description
Technical Specifications, ODAM, TRM	INT0070508, Bases for the CREF operability during Mode 1 operation

Related Lessons
INT0070508 CNS Tech. Spec. 3.7, Plant Systems

Related Objectives	
INT00705080010200	Discuss the applicable Safety Analysis in the Bases associated with each Chapter 3.7 Specification.

Related References	
TS	Tech Specs, Sec 3.7/4.7
10CFR55.43 (2)	Facility operating limitations in the technical specifications and their bases.
3.7.4	Control room emergency filter (CREF) system
ALL BASES	Improved Standard Technical Specification Bases

Related Skills (K/A)	
2.2.25	Knowledge of bases in technical specifications for limiting conditions for operations and safety limits. (CFR: 43.2) (2.5/3.7)
LINK ONLY TO TECH SPEC LESSONS & QUESTIONS	
290003.K5.01	Knowledge of the operational implications of the following concepts as they apply to CONTROL ROOM HVAC (CFR: 41.5 / 45.3) Airborne contamination (e.g., radiological, toxic gas, smoke) control (3.2/3/5)

QUESTION: 93 21297 (1 point(s))

What is the bases of for maintaining the CREF system operable during Mode 1 operations?

<u>Maintains the control room environment following a DBA for a…</u>

a. 200 man day continuous occupancy without exceeding 5 rem whole body dose or its equivalent to any part of the body.

b. 200 man day continuous occupancy without exceeding the limits specified in 10CFR100 for accidents.

c. 365 man day continuous occupancy without exceeding 5 rem whole body dose or its equivalent to any part of the body.

d. 365 man day continuous occupancy without exceeding the limits specified in 10CFR100 for accidents.

ANSWER: 93 21297

a. 200 man day continuous occupancy without exceeding 5 rem whole body dose or its equivalent to any part of the body.

The CREF System is designed to maintain the control room environment for a 200 man day continuous occupancy after a DBA without exceeding 5 rem whole body dose or its equivalent to any part of the body. The CREF System will pressurize the control room to $\geq$ 0.1 inches water gauge to prevent infiltration of air from surrounding buildings and the outside atmosphere.

SRO Justification: 10CFR55.43(b)(2)

Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
94	19325	00	08/20/2005	10/01/2005	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	INT032010400E0200, Facility License limitations; SRO ONLY

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

Related Objectives	
INT032010400E0200	Discuss Precautions and Limitations associated with Procedure 2.1.10, Station Power Changes.
INT032010400E030A	Discuss the following as described in Procedure 2.1.10, Station Power Changes: General Guidelines for Station Power Changes.

Related References	
2.1.10	Station Power Changes
10CFR55.43 (1)	Conditions and limitations of the facility license.

Related Skills (K/A)	
2.1.10	Knowledge of conditions and limitations in the facility license. (CFR: 43.1 / 45.13) (2.7/3.9)
2.1.10	Knowledge of conditions and limitations in the facility license. (CFR: 43.1 / 45.13) (2.7/3.9)

QUESTION: 94 19325 (1 point(s))

The reactor has been operating at 100% power. The current 8 hour core thermal average power is 2381 MWth. Reactor engineering delivers a Notification documenting an error in the core thermal power heat balance program. Due to the error, calculated thermal power is 15 MWth lower than actual thermal power.

What action is required?

a. initiate a reactor shutdown within one hour because Technical Specification 3.0.3 must be entered.

b. immediately restore reactor power to ≤ 2381 MWth **actual** since the operating license has been violated.

c. restore reactor power to ≤ 2381 MWth **actual** within two hours since a thermal limit has been exceeded.

d. restore reactor power to ≤ 2381 MWth **actual** and insert all insertable control rods within two hours since a safety limit has been exceeded.

ANSWER: 94 19325

b. is correct. The operating license requires thermal power to be ≤2381 MWth and power must be restored less than this. 2.1.10 Step 9.2.3 requires a immediate reduction of power is a limit is exceeded and power has exceeded the 8 hour average allowed.

a. is incorrect. Tech Spec 3.0.3 is not applicable to this condition since an LCO has not been entered that cannot be completed.

c. is incorrect. This is the action for violation of thermal limit and is not applicable.

d. is incorrect. This is the action for violation of a safety limit and is not applicable.

Reference: Operating License and 2.1.10 Section 9 and Precaution 2.1.6.

SRO Only Justification: 10CFR55.43(b)(1)

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
95	12483	01	09/10/2003	10/01/2005	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 Requirements Manual	INT0070601, TRM 3.4.1 High Rx conductivity at 9% pwr

Related Lessons
INT0070601 TRM - Overview, Reactor Power Distribution, Reactor Coolant and Refueling

Related Objectives	
INT0070601001040B	Given plant conditions and the TRM, determine ACTIONS required per the following TLCOs: T.3.4.1 RCS Chemistry

Related References	
T3.4.1	RCS Chemistry
10CFR55.43 (2)	Facility operating limitations in the technical specifications and their bases.

Related Skills (K/A)	
204000.A1.09	Ability to predict and/or monitor changes in parameters associated with operating the REACTOR WATER CLEANUP SYSTEM controls including: (CFR: 41.5 / 45.5) Reactor water conductivity (3.0/3.2)
2.1.32	Ability to explain and apply system limits and precautions. (CFR: 41.10 / 43.2 / 45.12) (3.4/3.8)
NRC EXAM ONLY	
2.1.34	Ability to maintain primary and secondary plant chemistry within allowable limits. (CFR: 41.10 / 43.5 / 45.12) (2.3/2.9)
2.1.33	Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications. (CFR: 43.2 / 43.3 / 45.3) (3.4/4.0) **EXAM USE ONLY**

QUESTION: 95 12483 (1 point(s))

The plant is at 9% power and 926 psig during a plant startup. Chemistry sample results indicate Reactor Coolant System (RCS) conductivity (corrected to 25°C) is 2.3 µmho/cm (stable). Noble Chem injection is NOT in progress.

What action is required?

Restore conductivity to within limits within . . .

a. 48 hours or be in MODE 2 within the following 8 hours.

b. 48 hours or be in MODE 3 within the following 12 hours.

c. 72 hours or be in MODE 2 within the following 8 hours.

d. 72 hours or be in MODE 3 within the following 12 hours.

ANSWER: 95 12483

b. 48 hours or be in MODE 3 within the following 12 hours.

Power is less than 10% and temperature is greater than 212°F which is condition 2 of table T3.4.1-1. Conductivity is outside the limits specified for that condition. T3.4.1.A requires restoration to within the limit within 48 hours. And T3.4.1.F requires operation to be in Mode 3 within the following 12 hours.

a. and d. combine the actions for power greater than and less than 10%.
c. If power were greater than 10%.

<u>Provide TRM 3.4.1 to the candidate.</u>

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
96	20539	00	04/09/2004	10/01/2005	NRC Style Question	RO: SRO: NLO:	N Y N

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

Topic Area	Description
Administrative	Requirements for 10CFR50.59 Review

Related Lessons
SKL0100102 Initial License Self-Study Program

Related Objectives	
SKL0100102001120A	0.4/0.4A, Procedure Change Process/Procedure Change Process Supplement: Discuss the following as described in Administrative Procedures 0.4, Procedure Change Process and 0.4A, Procedure Change Process Supplement: 1) Administrative Hold 2) Non-Intent Change 3) Pen-And-Ink Criteria 4) Periodic Procedure Review 5) SRO Reviews

Related References	
0.4	Procedure Change Process
10CFR55.43 (3)	Facility licensee procedures required to obtain authority for design and operating changes in the facility

Related Skills (K/A)	
2.2.8	Knowledge of the process for determining if the proposed change / test / or experiment involves an unreviewed safety question. (CFR: 43.3 / 45.13) (1.8/3.3)

QUESTION: 96 20539 (1 point(s))

A non-intent screen of 6.1CS.201, CS MOTOR OPERATED VALVE OPERABILITY TEST (IST) (DIV 1) has been initiated.

Which of the following requested procedure changes **would be rejected** during the screening as a non-intent change?

a. Correct a **step number** referenced in the procedure body.

b. Add a drawing to **clarify the location** of a manipulated component.

c. Change the approval authority **title** from Shift Supervisor to Shift Manager.

d. Change CS-MO-12A stroke time **Acceptance Criteria** from ≤14 sec to ≤18 sec.

ANSWER: 96 20539
d. Change CS-MO-12A stroke time Acceptance Criteria from#14 sec to #18 sec.

Procedure 0.4 ATTACHMENT 4 non-intent screen would flag this as an intent change. This clearly alters the technical acceptance criteria for the valve stroke time.

Distractors:

a. is incorrect because this is a non-intent change.

b. is incorrect because this is a non-intent change.

c. is incorrect because this is a non-intent change.

SRO Only Justification: 10CFR55.43(b)(3)

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
97	21363	00	09/09/2005	10/01/2005	NRC Style Question	RO: SRO: NLO:	N Y N

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

Topic Area	Description
Administrative	INT0320109, What constitutes a Temporary Configuration Change.

Related Lessons
INT0320109 ENGINEERING PROCEDURES (RO)

Related Objectives
NONE

Related References	
10CFR55.43 (3)	Facility licensee procedures required to obtain authority for design and operating changes in the facility

Related Skills (K/A)	
2.2.15	Ability to identify and utilize as-built design and configuration change documentation to ascertain expected current plant configuration and operate the plant. (CFR: 43.3 / 45.13) (2.2/2.9)

QUESTION: 97 21363 (1 point(s))

What condition constitutes a Temporary Configuration Change (TCC) in accordance with Procedure 3.4.4?

a. Instrument calibration that is performed promptly and the system returned to normal upon completion.

b. Repair of equipment declared inoperable by Operations, where the original configuration is restored following the activity.

c. A pipe patch placed on a vital system that is not performed as a Work Order activity and is scheduled for maintenance in the near future.

d. Equipment with lifted power leads for repair and tagged out per Procedure 0.9, that is restored to its approved design configuration prior to release of the Clearance Order.

ANSWER: 97 21363

c. A pipe patch placed on a vital system that is not performed as a Work Order activity and is scheduled for maintenance in the near future.

A Temporary Configuration Change that directly relates to and is necessary to support maintenance is considered a Temporary Alteration in Support of Maintenance. A Temporary Alteration in Support of Maintenance is controlled by the work control process and is restored to the authorized design upon completion of the maintenance activity.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
98	20527	0	04/05/2004	10/01/2005	NRC Style Question	RO: SRO: NLO:	N Y N

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

Topic Area	Description
Administrative	Approving Radioactive Releases

Related Lessons	
INT0320115	OPS CNS Administrative Procedures Radiation Protection and Chemistry Procedures (Formal Classroom/Pre-OJT Training)

Related Objectives	
INT0320115B0B0300	State the number of Circulating Water Pumps required to be in service during liquid radioactive discharges.

Related References	
8.8.11	Liquid Radioactive Waste Discharge Authorization

Related Skills (K/A)	
2.3.6	Knowledge of the requirements for reviewing and approving release permits. (CFR: 43.4 / 45.10) (2.1/3.1)

QUESTION: 98 20527 (1 point(s))
A plant startup is in progress with one CW pump running and de-icing in progress. A liquid radioactive discharge is needed. The liquid radwaste monitor is inoperable.

Can this liquid radioactive release be approved/authorized?

Why or why not?

a. No, insufficient dilution flow is currently available.

b. No, a liquid radioactive release is prohibited with the monitor inoperable.

c. Yes, as long as a second individual performs an independent verification of work/calculations supporting the discharge.

d. Yes, as long as two additional grab samples are obtained and analyzed to ensure the limits of 10CFR20, Appendix B, Table 2 are not exceeded.

ANSWER: 98 20527

a. No, insufficient dilution flow is currently available.

With only one CW pump running and de-icing in progress insufficient dilution flow exists for the discharge.

Distractors:

a. is incorrect because a discharge is allowed with the monitor inoperable.

c. is incorrect because the discharge is not allowed.

d. is incorrect because the discharge is not allowed.

SRO Justification: This is an SRO only item because in accordance with procedure 8.8.11 only the duty Shift Manager can approve liquid radioactive releases.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
99	21810	00	10/05/2005	12/31/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	6	Multiple Choice	

Topic Area	Description
Abnormal/Emergency Procedures	INT0320130, What is to be done with Rx Bldg sumps during high rad conditions

Related Lessons
INT0320130 CNS Abnormal Procedures (RO) High Radiation

Related Objectives	
INT0320130F0F0100	Given plant condition(s), and the applicable Abnormal/Emergency Procedure, determine the correct subsequent actions required to mitigate the event(s).

Related References	
5.1RAD	Building Radiation Trouble
5.2FUEL	Fuel Failure

Related Skills (K/A)	
2.4.8	Knowledge of how the event-based emergency/abnormal operating procedures are used in conjunction with the symptom-based EOPs. (CFR: 41.10 / 43.5 / 45.13) (3.0/3.7)

QUESTION: 99 21810 (point(s))

An accident occurred with the following conditions:

- ➢ MSL HI HI Radiation Alarm.
- ➢ DW High range radiation monitors are at 5E5 Rem/hour.
- ➢ Radiation protection personnel exiting the Reactor Building report general radiation levels near the DW at 2000 Rem/hour.
- ➢ Reactor Building Sumps are OFF per 5.2 Fuel..
- ➢ An unisolable RWCU leak has occurred and cannot be isolated.

What action is required?

a. Enter EOP-5A only and leave the reactor building sumps off.

b. Enter EOP-5A only and operate the reactor building sumps only as required to maintain building water levels below maximum safe level.

c. Enter EOP-5A and EOP-1A and leave the reactor building sumps off.

d. Enter EOP-5A and EOP-1A only and operate the reactor building sumps only as required to maintain building water levels below maximum safe level.

ANSWER: 99 21810

d. Enter EOP-5A and EOP-1A only and operate the reactor building sumps only as required to maintain building water levels below maximum safe level.

Considering procedure hierarchy it would appear that the Operator would ignore Procedure 5 .2FUEL and just follow the EOPs; however, the intent is that if in 5 .2FUEL, the designated sump pumps should be considered to be unavailable in the context of Flowchart 5A; therefore, the sump pumps would only be used to maintain the area below the max safe operating level. Both EOP-5A and EOP1A require entry. EOP-5A due to the high building radiation levels and EOP-1A because a scram is required.

a. is incorrect because EOP-1A requires entry and the pumps are operated to maintain levels below maximum safe water level.
b. is incorrect because EOP-1A requires entry.
c. is incorrect because the pumps are operated to maintain levels below maximum safe water level.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
100	21811	00	10/05/2005	12/31/2003	NRC Style Question	RO: SRO: NLO:	N Y N

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

Topic Area	Description
Abnormal/Emergency Procedures	SKL0540101, Based on plant conditions determine who must be authorized KI

Related Lessons
SKL0540101 EP Training Scenarios

Related Objectives
NONE

Related References	
5.7.12	Emergency Radiation Exposure Control
5.7.14	Stable Iodine Thyroid Blocking (KI)
5.7.2	Shift Supervisor EPIP

Related Skills (K/A)	
2.4.38	Ability to take actions called for in the facility emergency plan / including (if required)supporting or acting as emergency coordinator. (CFR: 43.5 / 45.11) (2.2/4.0)
2.4.29	Knowledge of the emergency plan. (CFR: 43.5 / 45.11) (2.6/4.0)
LINK TO GEN003-04-01 AND QUESTIONS ONLY	

QUESTION: 100 21811 (point(s))

A refueling accident occurred and the reactor building was evacuated. The reactor building Station Operator is injured and remains in the building. The injured operator is in communication with the Control Room. Many Area Radiation Monitor alarms, including refuel floor #2 are in alarm. Annunciator 9-4-1/E-4 RX BLDG VENT HI-HI RAD is in alarm. The estimated CDE is 30 Rem (thyroid) for the rescue team and projected CDE for the employees in the Control Building Envelope is < 1 Rem (thyroid).

Who should be authorized potassium iodide (KI)?

a. ONLY the injured operator.

b. ONLY the rescue team members

c. ONLY the rescue team and the injured operator.

d. Rescue team, injured operator and the Control Room staff.

ANSWER: 100 21811

c. ONLY the rescue team and the injured operator.

Emergency Director (non-delegable) shall authorize KI for Emergency Workers when a calculated dose of 10 Rem (0.1 Sv) to the thyroid (CDE) is likely to be received. Both the rescue team and the injured operator are likely to exceed this dose.

a. is incorrect because the rescue team members should be authorized.

b. is incorrect because the injured operator should be authorized.

d. is incorrect because the Control Room staff should not be authorized.