Q#
RO 1

Question Description
ILT

Rev #	*Rev Date*	*Topic Area*
0	01/03/01	Main Turbine Generator Trip / 3

Diff

Q type	*Response Time*	*Max Point Value*	*Passing Point Value*	*Lesson #*
M/C		1		COR002-21-02

Objective #	*Reference*	*K/A #*	*10CFR 55 41/43/45*
COR001-14-01-4.b	OP-4.5, T. S. 3.3.1	295005 AA2.03	41(b)6, 41(b).7

K/A Text:

AA2.03 – Ability to determine and/or interpret the following as they apply to MAIN TURBINE GENERATOR TRIP: Trip Valve Position

Question:

Which of the following describes the arrangement of the turbine stop valve limit switches which input into the REACTOR PROTECTION SYSTEM (RPS).

a. There is one limit switch on each valve. The switches are connected such that one valve less than 90% open will cause a half scram.

b. There is one limit switch on each valve. The switches are connected such that one valve less than 10% open will cause a full scram.

c. There are two limit switches on each valve. The switches are connected such that one valve less than 90% open will cause a half scram.

d. There are two limit switches on each valve. The switches are connected such that one valve less than 10% open will cause a full scram.

Answer:

ANSWER: c.

REFERENCE: OP 4.5, T.S. 3.3.1

K/A System: 295005
K/A Number: AA2.03
K/A Value: 3.1
Cognitive Level: 1

Justification: There are two limit switches on each valve. The switches are connected such that one valve less than 90% open will cause a half scram.

Distracter a: There are two limit switches per valve

Distracter b: There are two limit switches per valve, positions are from full open not full closed.

Distracter d: Valve logic is from full open not full closed

SOURCE: Cooper Exam Bank

1.1.10 When a trip occurs in both trip systems, both scram header pilot valve solenoids are deenergized, the flow of instrument air is blocked past this point, and the header is vented, causing the scram header vent valves (two valves) and SDV drain valves to close. Moreover, whenever a reactor scram occurs, both backup scram valve solenoids (normally deenergized) are energized and instrument air is blocked and vented at this point. This backup action, by itself, would cause the insertion of the control rods and closure of the scram header vent valves and SDV drain valve. However, the backup scram valves take longer to bleed air from the header. Thus, scram times could be exceeded, if only backup scram valves caused the scram.

1.1.11 The Reactor Protection System is equipped with a seismically qualified, Class 1E Power Monitoring System. This system consists of eight Electrical Protection Assemblies (EPA) which isolate the power sources from the RPS if the input voltage or frequency are not within limits specified for safe system operation. Isolation of RPS power causes that RPS division to fail safe.

1.2 RPS SCRAM FUNCTION LOGIC

1.2.1 TURBINE STOP VALVES

The two turbine stop valves are equipped with local valve position switches which generate trips in the RPS when the turbine stop valves start to close. Partial closure ($\leq$ 10%) of both valves will initiate a reactor scram above 30% rated load. 30% of rated load equals 233 psig turbine first stage pressure.

1.2.2 TURBINE CONTROL VALVE FAST CLOSURE

This trip signal indicates loss of the turbine generator and resultant inadequate heat sink. Pressure switches set at $\geq$ 1018 psig turbine control valve fluid are connected into the RPS to scram the reactor on control valve fast closure above 30% rated load. Pressure Switches TGF-PS-63OPC1 and TGF-PS-63OPC3 go to RPS A. Pressure Switches TGF-PS-63OPC2 and TGF-PS-63OPC4 go to RPS B. This is a one of two taken twice logic. 30% of rated load equals 233 psig turbine first stage pressure.

Table 3.3.1.1-1 (page 3 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
7. Scram Discharge Volume Water Level - High					
a. Level Transmitter	1,2	2	G	SR 3.3.1.1.4 SR 3.3.1.1.9 SR 3.3.1.1.12 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 90 inches
	5[(a)]	2	H	SR 3.3.1.1.4 SR 3.3.1.1.9 SR 3.3.1.1.12 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 90 inches
b. Level Switch	1,2	2	G	SR 3.3.1.1.4 SR 3.3.1.1.9 SR 3.3.1.1.12 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 90 inches
	5[(a)]	2	H	SR 3.3.1.1.4 SR 3.3.1.1.9 SR 3.3.1.1.12 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 90 inches
8. Turbine Stop Valve — Closure	≥ 30% RTP	2	E	SR 3.3.1.1.4 SR 3.3.1.1.9 SR 3.3.1.1.12 SR 3.3.1.1.13 SR 3.3.1.1.14 SR 3.3.1.1.15	≤ 10% closed
9. Turbine Control Valve Fast Closure, DEH Trip Oil Pressure — Low	≥ 30% RTP	2	E	SR 3.3.1.1.4 SR 3.3.1.1.9 SR 3.3.1.1.12 SR 3.3.1.1.13 SR 3.3.1.1.14 SR 3.3.1.1.15	≥ 1018 psig
10. Reactor Mode Switch — Shutdown Position	1,2	1	G	SR 3.3.1.1.11 SR 3.3.1.1.13	NA
	5[(a)]	1	H	SR 3.3.1.1.11 SR 3.3.1.1.13	NA
11. Manual Scram	1,2	1	G	SR 3.3.1.1.9 SR 3.3.1.1.13	NA
	5[(a)]	1	H	SR 3.3.1.1.9 SR 3.3.1.1.13	NA

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

Q#	*Question Description*	*Rev #*	*Rev Date*	*Topic Area*	*Diff*
RO 2	ILT	0	01/03/01	SCRAM	

Q type	*Response Time*	*Max Point Value*	*Passing Point Value*	*Lesson #*
M/C		1		

Objective #	*Reference*	*K/A #*	*10CFR 55 41/43/45*
COR002-32-02-6.f	Reactor Vessel Level Control Text	295006 AK3.01	41(b).7

K/A Text:

AK3.01 – Knowledge of the reasons for the following responses as they apply to SCRAM: Reactor Water Level Response

Question:

The reactor is operating at 100% power and the Reactor Vessel Level Control (RVLC) System is maintaining RPV water level at 35 inches. Five (5) minutes after a reactor scram, which one of the following levels will the RVLC system maintain as indicated on Narrow Range instruments? **Assume NO Operator Action and Reactor Feed Pumps remain operating**.

a. 15 inches

b. 25 inches

c. 35 inches

d. 45 inches

Answer:

ANSWER: b. 25"

REFERENCE: Reactor Vessel Level Control Text

K/A System: 295006
K/A Number: AK3.01
K/A Value: 3.8
Cognitive Level: 1

Justification: Following a scram level control resets to control at 25 inches

Distracter a. All values selected ending in 5 and separated by 10"

Distracter c. All values selected ending in 5 and separated by 10"

Distracter d. All values selected ending in 5 and separated by 10"

SOURCE: Cooper Exam Bank

d. Place the Override Enable Switch in the OFF position.

e. At the M/A transfer station, slowly lower the setpoint while observing that the FW Demand indicator tracks downward on the Startup Station. Observe that the FEEDWATER IN SERVICE LED comes on when the FW Demand digital indicator is slightly lower than the Startup Demand. This indicates the M/A transfer station now has control.

f. Place MANUAL STARTUP RATE SELECTOR switch to rate "4".

g. Run out the Startup Station to the maximum level to provide the full range of control to the GEMAC controller.

h. Place controllers in AUTO.

LO-02d; 06f LO-09c,d

I. Response of Reactor Vessel Level Control System and Feedwater System to a Main Turbine Trip or Reactor Scram

LO-11a

1. When the Main Turbine is tripped, the following occurs:

a. Feed pump discharge valves shut (RF-29, 30 MV).

b. Startup flow control valves isolation valves open (RF-31, 32, 33, 34 MV).

c. The Startup Flow Controller, which is normally left in automatic and set at 25", should attempt to control vessel water level.

d. The actions in a. and b. above are interlocked for 3 minutes.

2. When the reactor scrams, the following occurs.

a. If the Main Turbine was in service, the action in V.H.1 will occur.

b. If the Main Turbine was not in service, the startup flow controller, which is normally left in automatic, should attempt to control vessel water level at the selected setpoint.

LO-06h LO-09c, d

J. Response of Reactor Vessel Control System to a Loss of Power

1. Loss of Inverter "A" (AA2)

If INV-A is lost, steam flow "B", feed flow "B", and reactor water level "B" output signals are lost. The following actions would occur:

a. The "B" relay of the high water level trip logic would deenergize. This provides one of the two necessary high level signals to the Main Turbine and RFP turbine trip circuits.

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO3	ILT	0	01/03/01	HIGH REACTOR PRESSURE	

Q type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C		1		

Objective #	Reference	K/A #	10CFR 55 41/43/45
COR002-14-02-7.a	COR002-14-02, Page 22, IV.C.5.b	295007 AA1.05	41(b).5, 41(b).7

K/A Text:

AA1.05 – Ability to operate and monitor the following as they apply to HIGH REACTOR PRESSURE: Reactor/turbine pressure regulating system.

Question:

The plant is operating at 100% power when the in-service DEH pressure controller fails such that controller output INCREASES slowly. Which one of the following describes the plant response? **Assume NO operator action**.

a. The reactor will scram when reactor pressure rises to approximately 1050 psig.

b. The MSIV will isolate when reactor pressure lowers to approximately 835 psig.

c. Turbine throttle pressure will be controlled approximately 4 psig lower than before the failure.

d. Turbine throttle pressure will be controlled approximately 4 psig higher than before the failure.

Answer:

ANSWER: b. the MSIV will isolate when reactor pressure lowers to approximately 835 psig.

REFERENCE: COR002-14-02, Page 22, IV.C.5.b, rev. 12

K/A System: 295007
K/A Number: AA1.05
K/A Value: 3.7
Cognitive Level: 2

Justification: the MSIV will isolate when reactor pressure lowers to approximately 835 psig.

Distracter a: Reactor pressure will lower as controller output signals the TCVs to OPEN.

Distracter c: The backup pressure regulator is set for a pressure 4 psi higher.

Distracter d: Reactor pressure will lower as controller output signals the TCVs to OPEN.

SOURCE: Cooper Exam Bank

Fig 6

The pneumatic supply to the MSIVs is used to provide the motive force to the main valve and to reposition the pilot valve. Loss of pneumatic pressure to the pilot valve will, at ≈80 psig decreasing, cause the pilot valve spring to reposition the pilot valve to close the main valve. The main valve is closed by a spring and pneumatic pressure in the accumulator. Since the control solenoids will not necessarily be repositioned by an automatic action, if pneumatic pressure is subsequently restored the pilot valve will be repositioned and the MSIV will reopen.

On receipt of alarm 9-3-1/C-2 (DRYWELL PNEUMATIC HEADER LOW PRESSURE) any MSIVs that go closed must have their control switches placed to the closed position. If pneumatic supply pressure drops below 73 psig, close the inboard MSIVs (commitment SOER 88-1).

LO-05d, 07j

4. Loss of Electrical Power

a. Loss of 125VDC

1) Loss of DC power will cause the DC solenoids to reposition to close the MSIV. Because the AC solenoids remain energized in the open position, the MSIV will remain open. Loss of AA2 will affect the inboard MSIVs and loss of BB2 will affect the outboard MSIVs.

2) Loss of DC power will also prevent the electrical operation of MS-MO-77. If the loss is because of loss of power to the Rx Bldg Starter Rack, power may be able to be transferred to the other DC bus.

b. Loss of RPSPP

Loss of AC power will cause the AC solenoids to reposition to close the MSIV. As with the loss of DC power, the MSIV will remain open. Loss of RPSPP1A will affect only the solenoids for the inboard MSIVs and will close MO-74. Loss of RPSPP1B will affect only the solenoids for the outboard MSIVs and will close MO-77. Loss of power to the Group I isolation logic will also signal the drain valves MO-74, 77 to close.

c. Loss of MCC R

Loss of MCC R will prevent the electrical operation of MS-MO-74, MS-MO-78 and MS-MO-79. MCC R may be transferred to an alternate source of power.

LO-07a

5. Pressure Regulator Malfunction

a. Controller output decreases

If the in service pressure controller fails such that controller output decreases, control will automatically be shifted to the backup pressure controller. Because of the bias applied to the backup controller, its output is lower and the Pressure Control signal is less. This will cause the governor valves to start closing.

As the GV's close, pressure begins to increase because steam generation level has not changed. This causes an increase in the pressure error and a return of the Pressure Control signal to its original value. The GV's reopen to slightly less than their original position. Throttle pressure will level off about 4 psi higher than before due to the bias applied to the backup controller.

LO-07c

b. Output increases slowly

On a DEH output rising slowly the pressure control signal rises up to the flow limit setting. The GV's open and pressure lowers. With no operator action, pressure will lower to the Group I isolation setpoint, where the MSIVs will close to stop the pressure drop.

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO 4	ILT	0	2/2001	REACTOR WATER LEVEL CONTROL	

Q Type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C				COR002-32-02

Objective #	Reference	K/A #	10CFR 55 41/43/45
5, 7	2.4.5.1, Section 4.4, 6.3	295009, AK2.02	41(b)(7)

K/A Text:

AK2.02 – Knowledge of the interrelations between LOW REACTOR WATER LEVEL and the following: Reactor water level control

Question:

The plant is operating at power with the following reactor vessel level control alignment:

- RFC-LC-83, MASTER LEVEL CONTROLLER in balance
- RFC-MA-84A, FW CONTROLLER STATION A in balance
- RFC-MA-84B, FW CONTROLLER STATION B in balance

Feedwater flow is approximately $9.6x10^6$ lbm/hr. Steam flow is approximately $9.6x10^6$ lbm/hr. RPV water level is +35 inches.

The Master Controller OUTPUT slowly fails downscale. RPV water level lowers to +27 inches when the operator places the "A" and "B" RFP controllers to MANUAL.

Assuming **NO** additional action is taken by the operator, which one of the following describes the response of Feedwater Flow and RPV water level?

Feedwater flow will ...

a. rise to $9.6x10^6$ lbm/hr. Level will rise to +42 inches.

b. rise to $9.6x10^6$ lbm/hr. Level will remain at +27 inches.

c. rise above $9.6x10^6$ lbm/hr. Level will rise to +42 inches.

d. remain below $9.6x10^6$ lbm/hr. Level will continue to lower.

Answer:

ANSWER: b.

REFERENCE: 2.4.5.1, Section 4.4, 6.3

Tier: 1
Group: 1
K/A System: 295009
K/A Number: AK2.02
K/A Value: 3.9
Cognitive Level: 2
Bank/Mod/New: Bank

Distracter a: Level will not rise.
Distracter c: Feed flow will not rise above $9.6x10^6$ lbm/hr. Level will not rise.
Distracter d: Feed flow rises to $9.6x10^6$ lbm/hr. Level does not lower.

Proposed references to be provided to applicants during the examination: NONE

6. DISCUSSION

6.1 Reactor water level is maintained by the Feedwater Control System. This is accomplished automatically during plant operation. The Feedwater Control System performs its automatic function by use of three element control (reactor vessel level, steam flow, and feedwater flow). These are suitably combined to satisfy the reactor water level requirements according to operational mode and power requirements.

6.2 High water level results in water carry-over with the steam, which if allowed to occur, could cause turbine blading damage. The water level must be kept high enough on the steam separator skirts so as to eliminate carry-under of the steam with the water returning back to the downcomer annulus. This carry-under of steam could cause cavitation of the recirculation pumps. Thus, maintaining proper water level in the reactor is very important. Therefore, in the event of loss of the automatic function of the level control system, the Reactor Operator must be prepared to take manual control of the reactor feed pumps for level control.

6.3 This procedure covers malfunctions in the control system or indication. It is important in this procedure to determine quickly whether the controls or the level indicators are in error. Comparison of multiple instruments is important to ensure correct action. Since controllers are maintained balanced, taking manual control during the initial stages of the transient will usually stop the transient and allow the operator additional time to determine the failure. With level controls in manual, constant surveillance of vessel level is required.

6.4 All the Control Room A side level instruments share a common reference leg. This is also true of the B side level instruments. A reference line break or leak on the controlling instrument could cause all the level instruments connected to that line to indicate a higher level than actual reactor water level. This will cause the Feedwater Level Control System to reduce the speed of the feed pumps and actual level will drop. The level instruments connected to the affected reference leg will indicate an increasing level while the instruments connected to the unaffected line will indicate the actual decreasing level. The Operator must then compare the A side and B side level indications to the Steam Nozzle indication on Panel 9-3. The Steam Nozzle indication has an independent reference leg from the A and B side level instruments and will trend actual level. The Operator can then determine which level instruments are reading correctly by determining which instruments are trending with the Steam Nozzle indicator.©

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO 5	ILT	0	2/2001	LOW REACTOR WATER LEVEL	

Q Type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C				INT008-06-18, 2

Objective #	Reference	K/A #	10CFR 55 41/43/45
2	EOP-1A	295009, AA2.01	41(b)(7) 41(b)(10)

K/A Text:

AA2.01 – Ability to determine and/or interpret the following as they apply to LOW REACTOR WATER LEVEL: Reactor Water Level

Question:

During conduct of the EOPs, the following parameters exist:

- Reactor pressure 20 psig
- Drywell pressure 8 psig

- Drywell temperature 300°F
- Torus temperature 105°F
- Rx Building temperature 150°F

If actual reactor water level is at the top of active fuel (TAF) and **NO** instrument run boiling is observed, which one of the following describes the RPV level instrumentation that can be used to confirm reactor water level?

a. All level instruments are unavailable.

b. Fuel Zone level instruments can be used.

c. Wide Range level instruments can be used.

d. Narrow Range level instruments can be used.

Answer:

ANSWER: b.
Caution 1. Although in the unsafe region of Graph 1, instrument can be used as long as no boiling is observed.

REFERENCE: EOP-1A

Tier: 1
Group: 1
K/A System: 295009
K/A Number: AA2.01
K/A Value: 4.2
Cognitive Level: 2
Bank/Mod/New: Bank

Distracter a: Although in the unsafe region of Graph 1, instrument can be used as long as no boiling is observed.
Distracter c: Below minimum indicated level.
Distracter d: Below minimum indicated level.

Proposed references to be provided to applicants during the examination: All EOP graphs.

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO6	ILT	0	01/03/01	HIGH DRYWELL PRESSURE	

Q type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C		1		

Objective #	Reference	K/A #	10CFR 55 41/43/45
	EOP Bases EOP/SAG Graphs, Graphs 7 & 10	295010 AK1.01	41(b).5, 41(b).8

K/A Text:

AK1.01 – Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL PRESSURE: Downcomer Submergence: Mark I & II

Question:

After a transient, the following parameter values are noted:

- Drywell pressure 4.5 psig rising
- Drywell air temperature 140°F rising
- Torus pressure 4.5 psig rising
- Torus water temperature 82°F stable

Which one of the following is causing this response?

a. A safety relief valve has opened and its tailpipe vacuum breaker is open

b. The containment is functioning normally following a water break LOCA

c. A high energy discharge into the drywell with torus-to-drywell vacuum breakers closed.

d. High energy discharge into the drywell and PC water level has lowered to less than 9.6 feet

Answer:

ANSWER: d. PC water level has lowered to less than 9.6 feet

REFERENCE: EOP Bases EOP/SAG Graphs, Graphs 7 and 10

K/A System: 295010
K/A Number: AK1.01
K/A Value: 3.0
Cognitive Level: 2

Justification: At water levels of less than 9.6 feet drywell atmosphere will pass through the downcomers directly into the torus free air space.

Distracter a: This would pressurize the drywell and eventually the torus but torus pressure would be lower than drywell pressure because steam would be condensed in the torus and the downcomers would maintain a drywell torus d/p.

Distracter b: For normal function during a LOCA, the downcomers would maintain a d/p between the drywell and torus.

Distracter c: This is normal and drywell pressure would be higher than torus pressure.

SOURCE: New

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO 7	ILT	0	2/2001	FEEDWATER	

Q Type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C				

Objective #	Reference	K/A #	10CFR 55 41/43/45
COR002-02-02-9.a	2.4.9.4.7	295014, AK2.06	41(b)(10)

K/A Text:

AK2.06 – Knowledge of the interrelations between INADVERTENT REACTIVITY ADDITION and the following: Moderator temperature.

Question:

Following a loss of feedwater heating, the LOSS OF FW HEATING REGION of the Loss of Feedwater Heating Curve (attached) is entered.

Per 2.4.9.4.7, "Loss of Feedwater Heating," which one of the following describes the required action?

a. Lower reactor power below 25% RTP within 4 hours.

b. Restore operation to the normal region within 2 hours.

c. Immediately place the reactor mode switch to SHUTDOWN.

d. Immediately perform an emergency power reduction per 2.1.5.

Answer:

ANSWER: b.

REFERENCE: 2.4.9.4.7

Tier: 1
Group: 1
K/A System: 295014
K/A Number: AK2.06
K/A Value: 3.4
Cognitive Level: 1
Bank/Mod/New: New

Distracter a: Misinterpretation of the requirement for lowering reactor power. If performed, it must be completed within 2 hours.
Distracter c: This condition requires restoring feedwater temperature to the normal feedwater heating range, not a reactor scram.
Distracter d: This condition requires restoring feedwater temperature to the normal feedwater heating range, not a reactor scram.

Proposed references to be provided to applicants during the examination:
2.4.9.4.7, Attachment 1 (LOSS OF FEEDWATER HEATING CURVE)

1. SYMPTOMS

1.1 Annunciator A-2/C-6, HEATER HIGH LEVEL TRIP, alarms.

1.2 Annunciator A-2/C-5, HEATER HIGH LEVEL, alarms.

1.3 Annunciator A-2/C-4, HEATER LOW LEVEL, alarms.

1.4 Condensate flow recorder erratic (Panel A, MC-FR-17).

1.5 Decrease in feedwater temperature (Panel A, RF-TI-1, and Panel 9-4, RR-TR-165).

1.6 Alarm typer prints out abnormal temperature on affected heater.

2. AUTOMATIC ACTIONS

2.1 Heater non-return check valve closes on high-high heater water level or turbine trip.

2.2 Individual heater and extraction dump valves open or close, as required, for level control.

3. IMMEDIATE OPERATOR ACTION

3.1 None.

4. SUBSEQUENT OPERATOR ACTION

4.1 Monitor Panel 9-5 for power changes due to reactivity addition from increased subcooling from Feedwater System.

4.2 If feedwater temperature is lowering, observe following restrictions:

4.2.1 Maintain rod line < 120%.

4.2.2 If rod line is > 80%, maintain core flow > 45% (33.1 Mlbs/hr).

4.3 If feedwater temperature is in UNANALYZED REGION of Attachment 1, perform emergency shutdown from power per Procedure 2.1.5.

4.4 If feedwater temperature is in LOSS OF FW HEATING REGION of Attachment 1, restore feedwater temperature to NORMAL FEEDWATER HEATING REGION within 2 hours <u>or</u> reduce and maintain reactor power < 25% RTP.

<u>CAUTION</u> - High radiation levels are present in heater bay area.

4.5 Check for possible cause.

4.5.1 Insufficient venting.

4.5.2 High or low shell liquid level (controller failure).

4.6 Monitor turbine vibration and temperatures closely. Water induction can cause severe damage and must be stopped as quickly as possible.

4.7 If it becomes necessary to isolate or bypass feedwater heater(s) for more than 2 hours, reduce and maintain reactor power < 25% RTP.

4.8 Observe following Main Turbine Load limits due to heater removal:

4.8.1 One feedwater heater, no load reduction.

4.8.2 Two feedwater heaters, 5% load reduction from maximum.

4.8.3 Each additional heater drop load by 5%, maximum reduction 50%.

4.9 Reduce level in heater A5/B5 as follows:

4.9.1 Open the following valves by lowering pressure output of LIC-60, A-5 HEATER LEVEL CONTROLLER, on IR-1A/LIC-65, B-5 HEATER LEVEL CONTROLLER, on IR-1B (Turbine Building Control Corridor) or manually in case of controller failure:

4.9.1.1 CD-LCV-60A, HEATER 1A5 TO HEATER 1A4 LEVEL CONTROL.

4.9.1.2 CD-LCV-60B, HEATER 1A5 TO CONDENSER 1A LEVEL CONTROL.

4.9.1.3 CD-LCV-65A, HEATER 1B5 TO HEATER 1B4 LEVEL CONTROL.

4.9.1.4 CD-LCV-65B, HEATER 1B5 TO CONDENSER 1B LEVEL CONTROL.

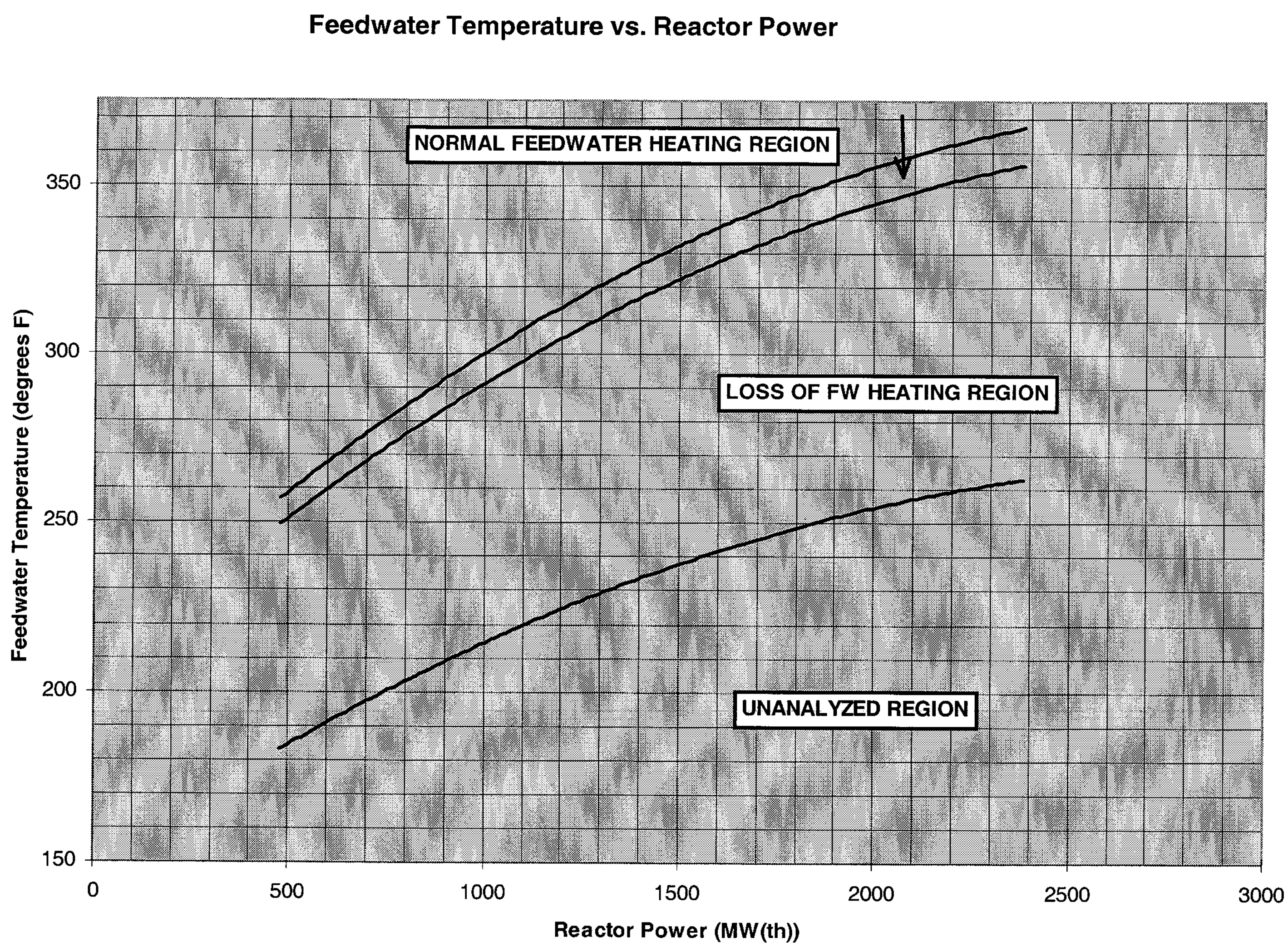

Feedwater Temperature vs. Reactor Power
NORMAL FEEDWATER HEATING REGION
LOSS OF FW HEATING REGION
UNANALYZED REGION
Feedwater Temperature (degrees F)
350
300
250
200
150
0
500
1000
1500
2000
2500
3000
Reactor Power (MW(th))

Figure 1

Q#
RO8

Question Description
ILT

Rev #	*Rev Date*	*Topic Area*
0	02/15/01	INCOMPLETE SCRAM

Diff

Q type	*Response Time*	*Max Point Value*	*Passing Point Value*	*Lesson #*
M/C		1		

Objective #	*Reference*	*K/A #*	*10CFR 55 41/43/45*
	EOP-6A	295015 AA1.07	41.6, 41.7, 41.10

K/A Text:

AA1.07 – Ability to operate and/or monitor the following as they apply to INCOMPLETE SCRAM: Neutron Monitoring System.

Question:

The plant has scrammed and all the control rods have NOT fully inserted. EOP-6A has been entered.

- The main turbine has tripped
- RPV pressure control is on the turbine bypass valves
- Feedwater control is maintaining RPV water level
- Drywell pressure is 0.4 psig
- APRM downscale lights are turned OFF

Which one of the following actions is required at this time?

a. Initiate boron injection.

b. Trip the recirculation pumps.

c. Prevent injection from ECCS systems NOT required for core cooling.

d. Exit EOP-6A and enter 2.1.5, "Emergency Shutdown and Scram Response".

Answer:

ANSWER: b. Trip the recirculation pumps

REFERENCE: EOP-6A

K/A System: 295015
K/A Number: AA1.07
K/A Value: 3.6
Cognitive Level: 2

Justification: At power levels above 3% the recirc pumps must be tripped, APRM downscale are ON below 3% and power levels above 3% are possible on Range 8 of the IRMs

Distracter a: Not required, heat is not being added to the torus

Distracter c: Pressure is above 350 psig, this is not required

Distracter d: Entry conditions are still met for EOP-6A

SOURCE: NEW

Q#	*Question Description*	*Rev #*	*Rev Date*	*Topic Area*	*Diff*
RO9	ILT	0	02/15/01	HIGH DRYWELL PRESSURE	

Q type	*Response Time*	*Max Point Value*	*Passing Point Value*	*Lesson #*
M/C		1		

Objective #	*Reference*	*K/A #*	*10CFR 55 41/43/45*
	EOP-3A and EOP-Graph 10	295024 2.4.6	41(b) 10

K/A Text:

2.4.6 – Knowledge of symptom-based EOP mitigation strategies.

Question:

Following a Loss of Coolant Accident the following conditions exist:

• Torus Pressure	25 psig
• Containment Water Level	12.0 feet
• Torus Water Temperature	145°F
• Drywell Pressure	27 psig
• Drywell Temperature	245°F and stable
• Reactor Pressure	50 psig
• RPV Water Level (Wide Range)	-34 inches

Which one of the following actions is required at this time?

a. Perform RPV Flooding.

b. Vent the Torus using SBGT..

c. Anticipate Emergency Depressurization

a. Conduct an Emergency RPV Depressurization.

Answer:

ANSWER: d. Conduct an Emergency RPV Depressurization

REFERENCE: EOP-3A, PC/P-4, Figure 9, EOP/SAG Graphs, Graph 10

K/A System: 295024
K/A Number: 2.4.6
K/A Value: 3.1
Cognitive Level: 2

Justification: PSP has been exceeded Emergency Depressurization is required.

Distracter a: RPV Flooding is not required water level instruments are operable with these conditions.

Distracter b: Drywell Pressure is not high enough to require emergency venting and Emergency Depressurization is required.

Distracter c: Emergency Depressurization is required because PSP has been exceeded. Anticipation of Emergency Depressurization is incorrect when the requirements to ED have already been met.

SOURCE: New

Q#	Question Description
RO10	ILT

Rev #	Rev Date	Topic Area	Diff
	01/03/01	HIGH REACTOR PRESSURE	

Q type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C		1		

Objective #	Reference	K/A #	10CFR 55 41/43/45
COR002-16-02-1.a	COR002-16-02	295025 EK3.09	41(b).7

K/A Text:

EK3.09 – Knowledge of the reasons for the following as they apply to HIGH REACTOR PRESSURE: Low low set initiation.

Question:

Which one of the following is the basis for the LOW LOW SET SRV logic?

a. Mitigate thrust loading on SRV discharge piping.

b. Eliminate cyclic stresses on the reactor pressure vessel.

c. Limit the reactivity effects caused by large pressure variations.

d. Prevent relief valve actuation if reactor water level is below the top of active fuel.

Answer:

ANSWER: a.

REFERENCE: COR002-16-02,

K/A System: 295025
K/A Number: EK3.09
K/A Value: 3.7
Cognitive Level: 1

Justification: Reduce high frequency loadings on the containment caused by SRV cycling.

Distracter b: Not a valid concern

Distracter c: Not a valid concern

Distracter d: Not a valid concern

SOURCE: Cooper Exam Bank

I. SYSTEM BRIEF DESCRIPTION

A. System Purpose

LO-01a
SO-01

1. The NUCLEAR PRESSURE RELIEF (**NPR**) system operates to prevent over-pressurization of the reactor system to prevent failure of the process barrier.

SO-02a

2. The AUTOMATIC DEPRESSURIZATION SYSTEM (**ADS**), utilizing six SAFETY/RELIEF VALVES (**SRV**s) in the Nuclear Pressure Relief system, operates as a backup to the HIGH PRESSURE COOLANT INJECTION (**HPCI**) system in the event of a small break loss of coolant accident.

SO-02b

3. The LOW-LOW SET (**LLS**) relief logic, utilizing two SRVs, reduces the number of SRV actuations during reactor isolation events in order to reduce the dynamic loads on the containment.

LO-10

B. Design Bases

1. Power Generation Design Bases

   The Pressure Relief system is designed to limit any over-pressure condition which would occur during an abnormal operational transient.

   a. The relief valves shall prevent the opening of the spring-loaded safety valves during normal plant isolations and load rejections.

   b. The relief valves shall discharge to the Primary Containment Suppression Pool.

   c. The relief valves shall properly re-close following a plant isolation or load rejection so that normal operation can be resumed as soon as possible.

   d. The Pressure Relief system is designed to be used to remove decay heat and depressurize the reactor to achieve safe shutdown in the special event of a fire.

LO-08j

2. Safety Design Bases

   The Pressure Relief system is designed to prevent over-pressurization of the nuclear system; thus protecting the nuclear system process barrier from failure which could result in the uncontrolled release of fission products. In addition, the Automatic Depressurization feature of the Pressure Relief system acts in conjunction with the CORE STANDBY COOLING SYSTEMS (**CSCS**) for reflooding the core, to maintain adequate core cooling, following small breaks in the nuclear system process barrier. This protects the reactor fuel barrier (cladding) from failure due to overheating.

LO-01d

   a. The Pressure Relief system shall prevent over-pressurization of the nuclear system in order to prevent failure of the nuclear system process barrier.

- Reactor water level must be low enough to close a -113 in. level switch.

- The ADS Logic A Timer Reset pushbutton (S2A) must not be depressed.

- The ADS A Inhibit Switch (S3A) must be in the "AUTO" position.

- Any one RHR or CS pump must be developing a discharge pressure of greater than 108 psig (AV $\geq$ 108 psig and $\leq$ 160 psig; (K12A contact).

- The reactor water level switch will be sealed in if all four of the above conditions are met.

LO-05a

b. ADS may be secured, or prevented from actuating, at any time by placing the ADS A and B Inhibit switches to the "INHIBIT" position. Depressing the ADS Logic A and B Timer Reset pushbuttons will secure the ADS blowdown, if in progress, and will reset the timers to zero. If the initiating conditions still exist, the timer will begin timing down again immediately after it has been reset. Securing all low pressure injection pumps (CS and LPCI) will also cause the ADS to stop a blowdown in progress.

c. ADS logic is powered from 125V DC, with Channel A powered from Panel AA2 and Channel B from Panel BB2. Channel B will automatically transfer to Panel AA2 on loss of power. Channel A does not have an automatic power supply backup.

E. Low-Low Set (LLS)

1. Upon initiation, LLS will lower the opening and closing setpoints of 2 SRVs by using RPV pressure switches to energize/de-energize the solenoid control valve to the pneumatic actuator of the SRV. The LLS pressure setpoints are:

	Valve	Open	Close	Blowdown
a.	RV-71 D	1015 psig ±20 psig	875 psig ±20 psig	approximately 140 psig
b.	RV-71 F	1025 psig ±20 psig	875 psig ±20 psig	approximately 150 psig

LO-01c, 05c

2. The LLS will mitigate SRV subsequent actuation induced loads. The opening and closing setpoints for Low-Low Set relief are spread farther apart than for normal relief. This allows for more steam (energy) to be released each time an SRV opens. More energy will be required for repressurization before an SRV reopens and the number of SRV cycles are reduced.

LO-05e

a. LLS will prevent excessive water-clearing thrust loadings on the SRV discharge piping, by allowing sufficient time for the vacuum breaker to reduce the water leg to a normal level.

LO-5d

b. LLS will also reduce the high frequency loadings on the containment caused by the inability of the suppression water to condense the air trapped in the discharge piping upon actuation.

Fig 6

3. LLS Logic

There are two LLS logics (A and B), associated with the two safety/relief valves (RV-71 D and RV-71 F) actuated by LLS.

a. For a relief valve to open from a LLS signal, two contacts in the valve logic must be closed, K20 and K21. The two relays causing closure of these contacts are part of the LLS logic.

LO-03j

1) To energize the K20 relay (arming relay), the following conditions must be met:

- Reactor pressure must be above the scram setpoint of 1050 psig.

- Any one of the eight SRVs must be open causing the 30 psig pressure switch (ITS Limits ≥25 psig to ≤55 psig) (K14 contacts) on the discharge piping to be closed.

2) When the K20 relay is energized, the LLS is "ARMED" and a seal in for arming LLS is achieved.

3) The K21 relay will energize when Reactor pressure reaches 1015 psig (± 20) and de-energize at 875 psig (± 20) for RV-71D, and between 1025 psig (± 20) and 875 psig (± 20) for RV-71F.

b. The only method of disarming LLS once it is armed is to depress the LLS reset pushbuttons (S6A and S6B), on Panel 9-3 in the Control Room, while either no SRVs are blowing down or reactor pressure is below the 1050 psig scram setpoint.

c. 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. On a loss of normal power, both channels will automatically transfer to the alternate power supply.

III. INSTRUMENTATION AND CONTROLS

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO 11	ILT	0	2/2001	REACTOR LOW WATER LEVEL	

Q Type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C				INT008-06-18

Objective #	Reference	K/A #	10CFR 55 41/43/45
8	EOP-1A, 6A	295031, EA1.08	41(b)(8) 41(b)(10)

K/A Text:

EA1.08 – Ability to operate and/or monitor the following as they apply to REACTOR LOW WATER LEVEL: Alternate Injection Systems

Question:

Note: All RPV levels are as INDICATED on the Fuel Zone instruments.

Which one of the following conditions assures adequate core cooling?

a. All control rods are fully inserted, Reactor Pressure 128 psig, RPV level -40 inches, **NO** SRVs open, the only available injection is ECCS pressure maintenance.

b. All control rods are fully inserted, Reactor Pressure 200 psig, RPV level -50 inches, **NO** SRVs open, the only available injection is one (1) Core Spray pump.

c. ATWS with reactor power at 5%, Reactor Pressure 60 psig, RPV level -20 inches, Three (3) SRVs open, the only available injection is one (1) RHR pump.

d. ATWS with reactor power at 14%, Reactor Pressure 385 psig, RPV level -50 inches, One (1) SRV open, the only available injection is (1) Condensate pump.

Answer:

ANSWER: c.
Level is above -30 inches for adequate steam cooling and 3 SRVs are open with Minimum Alternate RPV Flooding Pressure met.

REFERENCE: EOP-1A, 6A

Tier: 1
Group: 1
K/A System: 295031
K/A Number: EA1.08
K/A Value: 3.8
Cognitive Level: 3
Bank/Mod/New: Bank

Distracter a: -40 inches is too low for adequate steam cooling RC/L-16.
Distracter b: -50 inches corrected is below minimum steam cooling level.
Distracter d: -50 inches corrected is below minimum steam cooling level but above old minimum steam cooling level.

Proposed references to be provided to applicants during the examination: All the EOP graphs, EOPS 1A, 6A, and 7A.

Q#	*Question Description*	*Rev #*	*Rev Date*	*Topic Area*	*Diff*
RO 12	ILT	0	2/2001	ATWS	

Q Type	*Response Time*	*Max Point Value*	*Passing Point Value*	*Lesson #*
M/C				INT008-06-10

Objective #	*Reference*	*K/A #*	*10CFR 55 41/43/45*
3	EOP-7A	295037, 2.4.20	41(b)(10)

K/A Text:

2.4.20 – Knowledge of operational implications of EOP warnings/cautions and notes.

Question:

While performing EOP-7A, "RPV Level/Failure to Scram," with power below 3%, which one of the following CAUTIONS applies as reactor water level is lowered?

Lowering RPV water level to ...

a. -42 inches will result in an ADS initiation if ADS is **NOT** inhibited.

b. -110 will result in low pressure ECCS injection unless it is stopped and prevented.

c. -110 inches will result in an MSIV isolation and loss of the main condenser as a heat sink.

d. -42 inches will result in injection from low pressure ECCS systems **NOT** required for RPV level control.

Answer:

ANSWER: c.

REFERENCE: EOP-7A

Tier: 1
Group: 1
K/A System: 295037
K/A Number: 2.4.20
K/A Value: 3.3
Cognitive Level: 1
Bank/Mod/New: Bank

Distracter a: Caution does not exist in EOP-7A.
Distracter b: Caution does not exist in EOP-7A.
Distracter d: Caution does not exist in EOP-7A.

Proposed references to be provided to applicants during the examination: EOP 7A with all CAUTIONS blanked out.

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO 13	ILT	0	2/2001	CONTAINMENT	

Q Type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C				INT008-06-13 COR002-03-02

Objective #	Reference	K/A #	10CFR 55 41/43/45
INT008-06-13, 4 COR002-03-02, 14e	EOP-3A	500000, EA2.03	41(b)(7)

K/A Text:

EA2.03 – Ability to determine and/or interpret the following concepts as they apply to HIGH CONTAINMENT HYDROGEN CONCENTRATIONS: Combustible Limits for Drywell

Question:

A LOCA has occurred and the following conditions exist:

- Drywell H2 concentration is 7%
- Torus H2 concentration is 4%

- Drywell O2 concentration is 4%
- Torus O2 concentration is 6%

In accordance with the EOPs, which one of the following describes the Primary Containment H2/O2 combustible limit status (above or below the combustible limit) and the required actions?

The Primary Containment H2/O2 concentration is ...

a. **below** the combustible limit. Reactor scram and emergency depressurization is required.

b. **below** the combustible limit. Reactor scram and emergency depressurization is **NOT** required.

c. **above** the combustible limit. Reactor scram and emergency depressurization is required.

d. **above** the combustible limit. Reactor scram and emergency depressurization is **NOT** required.

Answer:

ANSWER: c.
The limits, 6%, H2 and 5%, O2 in either torus or drywell are the limits for the primary containment. Combustible limit exceeded requires a reactor scram and emergency depressurization.

REFERENCE: EOP-3A

Tier: 1
Group: 1
K/A System: 500000
K/A Number: EA2.03
K/A Value: 3.3
Cognitive Level: 2
Bank/Mod/New: Bank

Distracter a: See justification above.
Distracter b: See justification above.
Distracter d: See justification above.

Proposed references to be provided to applicants during the examination: **EOP-1A & EOP-3A.**

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO14	ILT		01/03/01	LOSS OF CORE FLOW	

Q type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C		1		

Objective #	Reference	K/A #	10CFR 55 41/43/45
COR002-22-02-6.e	2.4.2.2.1, T.S. 3.0 Bases C.1	295001 AK3.04	41(b).1, 41(b).10

K/A Text:

EK3.09 – Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Reactor SCRAM.

Question:

A plant startup is in progress with RPV level in the normal band. The reactor operator has just placed the Reactor Mode Switch in RUN per the startup procedure when both Reactor Recirculation Pumps TRIP.

Assuming **NO** other plant transient occurs, which one of the following actions is required and why?

a. RPV level must be raised to insure adequate core flow prior to restarting the first recirculation pump.

b. The reactor must be scrammed before a recirculation pump can be started to prevent a reactivity insertion accident.

c. Control rods must be inserted and the reactor cooled down before either of the recirculation pumps may be started to prevent thermal shock to the reactor vessel.

d. Bottom vessel drain, recirculation loop and the vessel saturation temperature must be within limits to prevent a cold water accident prior to starting each recirculation pump.

Answer:

ANSWER: b.

REFERENCE: 2.4.2.2.1, Trip of Reactor Recirculation Pumps, Sect. 3.0, Tech. Spec. Bases C.1.

K/A System: 295001
K/A Number: aK3.04
K/A Value: 3.4
Cognitive Level: 1

Justification: The reactor must be scrammed before a recirculation pump can be started to prevent a reactivity insertion accident.

Distracter a: Recirculation pumps may not be started, a scram is required first.

Distracter c: There is no requirement to cooldown prior to starting the recirculation pumps and the bases for the scram is reactivity NOT thermal shock.

Distracter d: Recirculation pumps may not be started, a scram is required first.

SOURCE: Modified Cooper Exam Bank

1. SYMPTOMS

[ ] 1.1 Annunciator 9-4-3/A-1, RRMG A BKR 1CS TRIP, alarms.

[ ] 1.2 Annunciator 9-4-3/A-2, RRMG A BKR 1CN TRIP, alarms.

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

[ ] 1.4 Annunciator 9-4-3/A-6, RRMG B BKR 1DN TRIP, alarms.

[ ] 1.5 Annunciator 9-5-2/C-8, ARI & ATWS RPT CHAN A/B PRESS TRIP, alarms.

[ ] 1.6 Annunciator 9-5-2/D-7, ATWS RPT CHAN A/B LEVEL TRIP, alarms.

[ ] 1.7 RRMG set generator volt, amp and power indicators read zero.

[ ] 1.8 Reactor power decrease on loss of recirculation flow.

[ ] 1.9 Reduction in RR loop flow as indicated on RR-FR-163 at Panel 9-4.

2. AUTOMATIC ACTIONS

[ ] 2.1 None.

3. IMMEDIATE OPERATOR ACTIONS

[ ] <u>**NOTE**</u> - Multiple indications should be used to determine if pumps have tripped (i.e., generator amps, volts and power, loop flows, etc.).

[ ] 3.1 If <u>both</u> RR pumps are tripped and reactor power > 1% rated thermal power, scram reactor.

[ ] 3.2 If <u>one</u> RR pump has tripped, perform following:

[ ] 3.2.1 Monitor power-to-flow map and core stability for entry conditions into Procedure 2.4.1.6.

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO15	ILT		01/03/01	LOSS OF CONDENSER VACUUM	

Q type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C		1		

Objective #	Reference	K/A #	10CFR 55 41/43/45
	2.4.9.3.5,	295002 AK2.07	41(b).5, 41(b).13

K/A Text:

AK2.07 – Knowledge of the interrelations between LOSS OF MAIN CONDENSER VACUUM and the following: Offgas system

Question:

The plant has been operating at 100% power for several days.

Over the last several hours Main Condenser Backpressure has risen from 2.5" Hg to 3.5" Hg and Offgas flow on AR-FR-47 has risen from 18 scfm to 30 scfm. There have been **NO** alarms during this time.

Which one of the following is the cause for these indications?

a. Circulating water temperature has risen.

b. A feedwater heater drain valve has failed open.

c. A detonation has occurred in the offgas system.

d. Air leakage into the main condenser has increased.

Answer:

ANSWER: d.

REFERENCE: 2.4.9.3.5, Loss of Condenser Vacuum, Section 4.8

K/A System: 295002
K/A Number: AK2.07
K/A Value: 3.1
Cognitive Level: 2

Justification: Air leakage into the main condenser has increased off gas flow

Distracter a: This would not cause offgas flow to rise

Distracter b: This would cause an alarm and not cause a loss of vacuum because the feedwater heaters are vented to the main condenser.

Distracter c: This would cause an alarm and not cause these changes in parameters, ie offgas would isolate and if not recovered backpressure would rise more significantly.

SOURCE: New question

1. SYMPTOMS

1.1 Annunciator B-1/A-3, TG LOW VACUUM TRIP, alarms.

1.2 Annunciator B-1/B-3, TG LOW VACUUM PRE-TRIP, alarms.

1.3 Annunciator 9-5-2/B-2, MSIV NOT FULL OPEN TRIP, alarms.

1.4 Low pressure turbine absolute Recorders MS-PR-73A, A ABSOLUTE PRESS, and MS-PR-73B, B ABSOLUTE PRESS, increasing on Panel B.

1.5 Low pressure turbine condenser vacuum Indicators MS-PI-72A, A VACUUM, and MS-PI-72B, B VACUUM, on Panel B indicate decreasing vacuum.

1.6 SJAE STEAM supply pressure low as indicated by MS-PI-77A, A INLET PRESS, and MS-PI-77B, B INLET PRESS, on Panel B.

1.7 AR-FR-47, SJAE AIR FLOW, high/low on Panel B.

2. AUTOMATIC ACTIONS

2.1 Gradual loss.

2.1.1 None.

2.2 Sudden and/or total loss.

2.2.1 Main Turbine trips.

2.2.2 Reactor scram, from stop valve closure, if reactor power is above 25% of rated first stage pressure.

NOTE - Group 1 low vacuum isolation is bypassed if all of following conditions are met:

2.2.2.1 REACTOR Mode switch is not in RUN.

2.2.2.2 Turbine stop valves are closed.

2.2.2.3 All four CONDENSER LOW VACUUM LOGIC TEST switches on Panels 9-15 and 9-17 are in BYPASS.

4.5.4 Locally in SJAE Room perform following:

NOTE - Venting feedwater heater flash sections too long can cause flooding of SJAE inner condenser.

4.5.4.1 Check levels in SJAEs inner and after condensers for proper level.

a. If level is high, ensure following valves on Panel B are open:

1. CD-IV-8A, SJAE A AFTER CNDR DRN TRAP INLET ISOL.

2. CD-IV-9A, SJAE A AFTER CNDR DRN TRAP OUTLET ISOL.

3. CD-IV-8B, SJAE B AFTER CNDR DRN TRAP INLET ISOL.

4. CD-IV-9B, SJAE B AFTER CNDR DRN TRAP OUTLET ISOL.

b. If valves are open, perform following in an attempt to lower SJAE inner and after condenser level:

1. Open CD-BV-10A, SJAE A AFTER CNDR DRN TRAP BYPASS, if SJAE A condenser levels are high.

2. Open CD-BV-10B, SJAE B AFTER CNDR DRN TRAP BYPASS, if SJAE B condenser levels are high.

c. Check SJAE valve line-up is correct per Procedure 2.2.55.

4.5.5 Check mechanical vacuum pumps for proper operation if they are required to be running per Procedure 2.2.55.

4.6 If loss of condenser vacuum was due to high off-gas activity and closure of isolation valves, refer to Procedure 2.4.7.1.

4.7 If loss of condenser vacuum was due to a high temperature or high pressure in off-gas line, refer to Procedure 2.4.7.2 or 2.4.7.3.

4.8 Check off-gas flow on AR-FR-47, SJAE AIR FLOW (Panel B), for indications of higher air in-leakage and perform following if readings have gone up:

4.8.1 Check operation of Gland Sealing System per Procedure 2.2.75.

4.8.2 Ensure AR-MO-150, VACUUM BREAKER (Panel B), is closed.

4.8.3 Send an Operator locally to vacuum breaker (Heater Bay 903 east side) and ensure vacuum breaker is sealed by pouring water into vent line which extends from top of valve.

4.8.4 Check RWCU blowdown line-up to ensure RWCU-MO-56 and RWCU-MO-57 (Panel 9-3) are not open at same time.

4.8.5 Walk down main condenser and attached piping and check for air leaks.

5. PROBABLE CAUSE

5.1 Circulating water pump(s) trip.

5.2 Condenser expansion joint leak or rupture.

5.3 SJAE/mechanical vacuum pump malfunction.

5.4 Loss of sealing steam.

6. DISCUSSION

6.1 This procedure covers loss of main condenser vacuum which could be due to SJAE failure, circulating water pump failure, condenser air leaks, high radiation in the off-gas pipe and resulting isolation of off-gas system, steam sealing system failure, or closure of the MSIVs. The main intent of this procedure is to maintain the reactor in a safe condition and prevent the release of radioactive material to the environment.

6.2 The following limits are to protect the last row of low pressure turbine blading during high back pressure operation. Last row blade and/or disc attachment fatigue damage can occur during relatively brief periods under high back pressure, low load conditions. The damage is cumulative and irreversible. If operating data indicate a trend of increasing exhaust pressure, it is advisable to identify and correct the causes as soon as possible.

6.2.1 At unit loads > 30% of rated capacity, the back pressure is to be maintained at 5.5" HgA or lower.

6.2.2 At unit loads < 30% of rated capacity, the back pressure is to be maintained at 3.5" HgA or lower.

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO16	ILT	0	01/03/01	Partial or Complete Loss of AC Power	

Q type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C		1		

Objective #	Reference	K/A #	10CFR 55 41/43/45
COR002-23-02-8.a	COR002-23-02, COR001-01-01	295003 AK1.03	41(b).7

K/A Text:

AK1.03 – Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER. Under voltage/degraded voltage effects on electrical loads.

Question:

Given the following conditions:

- The Plant is in cold shutdown
- The C Residual Heat Removal (RHR) Pump is running in shutdown cooling
- ALL plant systems respond as designed

A fault on the electrical system lowers 1G 4160 VAC Switchgear Bus voltage to 1000 volts for one (1) second then recovers

Which one of the following statements below describes the response of the C RHR Pump to these conditions?

The C RHR Pump will...

a. trip on undervoltage, **BUT** will automatically restart.

b. trip on undervoltage, **BUT** will NOT automatically restart.

c. continue to run because bus voltage recovers within 3 seconds.

d. continue to run because the bus fast transfers in less than 0.25 seconds.

Answer:

ANSWER: b ~~a.~~ ~~trip on undervoltage, BUT will automatically restart~~

REFERENCE: COR002-23-02, Residual Heat Removal, page 14, section II.C, rev. 13
COR001-01-01, AC Electrical Distribution, page 45, section IV.F, rev. 11

K/A System: 295003
K/A Number: AK1.03
K/A Value: 2.9
Cognitive Level: 2

Justification: A low voltage on the 1G Buss (less than 2300 V) will trip the C RHR Pump breaker. When voltage is restored the UV relays automatically reset. The Anti-Pump feature is only active if the breaker trips with a sustained START signal present. Since no start signal was present at the time of the transient, the Anti-Pump feature does not need to be reset, and any subsequent LPCI signal will automatically start the pump.

Distracters b: The pump breaker will reclose attempting a re-start because a LPCI signal was not present.

Distracter c& d: Bus voltage will drop below 2300 V, tripping all pump breakers

SOURCE: Modified Cooper Question

a. With normal power available, pumps A and D start immediately and pumps B and C start after a 5 second time delay. This time delay prevents voltage dips on the 4160V emergency buses due to ECCS initiation.

LO-03f; 06i; 08a

b. With a loss of off-site power, pumps A and D start immediately upon restoration of power, and pumps B and C start 5 seconds after the restoration of power. This time delay prevents overloading the emergency diesels due to ECCS initiation.

NOTE - If the pump control switch is taken to the STOP position and released (AUTO after STOP) with an initiation signal present, the pump will stop and an amber PUMP STOP SIG SEALED-IN light above the control switch will illuminate. This light remains on as long as the LPCI signal is present. It resets automatically as soon as the initiation signal clears. The amber light will also illuminate, without an initiation signal present, any time the control switch is in STOP or placed in PULL-TO-LOCK. The light will go out once the switch is removed from these positions.

LO-08a
LO-15c

6. The RHR pump motor supply breaker will trip on the following:

a. Electrical fault (overcurrent, ground, etc.).
b. Low voltage on critical bus (≈ 2300V).
c. MO-17, or MO-18, or the associated MO-15 valve not full open AND the associated MO-13 valve not full open.

Each RHR pump 4160V breaker has anti-pump circuitry to prevent the breaker from cycling after an electrical trip. The breakers have auto-close signals (LPCI initiation) which must be interrupted to remove the anti-pump seal in logic. First, the condition that caused the electrical trip must be cleared. The anti-pump seal in logic can be interrupted by either operating the 4160V breaker shutter to the left and back to the right or taking the RHR pump control switch to STOP.

LO-03d,i, 04c

7. Check valves are located on the discharge of each pump in order to prevent back flow through an idle pump and to aid in maintaining that leg of piping filled with water. The discharge piping is maintained filled with water, up to the LPCI injection valves, by the Pressure Maintenance system for two reasons:

LO-05e

a. Prevents water hammer on pump starts and the possible pipe and valve damage that may result.

b. Prevents a pump runout condition from occurring while discharging to an empty pipe on a start up. (Refer to the Condensate and Feedwater text for further information on the Pressure Maintenance system.)

LO-13e

8. Minimum Flow Valves (RHR-MO-16(#)A and B)

The pump minimum flow valves, one in each loop, provide the necessary flow through the pump in order to prevent pump overheating. The RHR pump

IV. OPERATIONAL SUMMARY

A. 345 kV Switchyard Breakers

LO-10b

The 345 kV switchyard circuit breakers, except for PCB-3310 and PCB-3312, can be opened or closed from the switchyard Control House or by the supervisory control from the Load Dispatchers office or locally at the breaker. Circuit breakers PCB-3310 and PCB-3312, also called the Main Generator output circuit breakers, are normally operated from Control Room Panel C. These breakers can also be operated from the switchyard control house or locally at the breaker. These breakers will automatically trip in such a manner as to isolate any fault on the 345 kV grid or outgoing line from the Main Generator.

Fig 4

B. 4160V

LO-10b

LO-09a,b,d
SO-13d

LO-13e,i,j; 14f
SO-13a,b

1. All 4160V breakers are individually controlled normally from the Control Room with their Trip-Close control switches. In emergencies, the breakers could be tripped using the pushbutton switch locally on the breaker. The position of the breakers are indicated by red (closed) and green (open) indicating lights and a position indicating window which are provided locally at the breaker cubicle and on the control panels. The 4160V breakers are furnished with individual protective relays which would initiate an automatic signal to trip the breaker during fault or abnormal conditions. The breakers are tripped by undervoltage, overcurrent, ground sensing, or differential type relays in order to disconnect and isolate the electrical fault and protect the electrical equipment while maintaining continuity of service on the remaining systems. Selective tripping is the protection scheme designed to trip the closest device to the fault source to safely interrupt the fault current. This minimizes the number of circuits that could lose power or become damaged. Following a trip, a breaker cannot be reclosed unless the abnormality has been corrected and the protective relays have been reset manually or automatically.

LO-13k

There are two levels of undervoltage protection at CNS. The first level is a loss of voltage protection which is designed to actuate at conditions indicative of a grid voltage rapidly collapsing to zero volts (i.e., bus voltages < 2870 volts). The relays which actuate are a time undervoltage relay with inverse time characteristics (i.e., the lower the voltage, the faster the actuation).

The second level of undervoltage protection is for sustained degraded (low) voltage conditions. This system is designed to respond to a static low voltage condition and will actuate whenever the bus voltage drops below 3880 ± 52 volts for a time period of 7.5 ± 0.8 seconds.

LO-09j

Some equipment powered from 4160V buses has anti-pump circuitry to prevent the breaker from cycling after an electrical trip. These breakers have auto-close signals which must be interrupted to remove the anti-pump seal in logic.

NOTE: The electrical trip signal must be cleared before proceeding, record and reset all flags on the breaker.

a. For all 4160V breakers this is accomplished by operating the breaker

Q#	*Question Description*	*Rev #*	*Rev Date*	*Topic Area*	*Diff*
RO 17	ILT	0	2/2001	DC DISTRIBUTION	

Q Type	*Response Time*	*Max Point Value*	*Passing Point Value*	*Lesson #*
M/C				SKL012-42-03

Objective #	*Reference*	*K/A #*	*10CFR 55 41/43/45*
02	2.6.1, section 6.1.1 2.2.25, section 2.2.6	295004, AA1.02	41(b)(6) 41(b)(7)

K/A Text:

AA1.02 – Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF DC POWER: Systems necessary to assure safe plant shutdown.

Question:

Given the following conditions:

- **ALL** 4160 volt busses are de-energized
- VBD-H Manual Transfer switch is in ALTERNATE
- **ALL** Division I DC power sources are <u>un</u>available

Which one of the following describes the indicators available to assess Suppression Pool Temperature **WITHOUT** reliance on other indications?

a. PMIS/SPDS only.

b. Alternate Shutdown Panel instruments only.

c. PMIS/SPDS and one (1) of the Suppression Chamber Water Temperature recorders.

d. Alternate Shutdown Panel instruments and one (1) of the Suppression Chamber Water Temperature recorders.

Answer:

ANSWER: b.

REFERENCE: 2.6.1, section 6.1.1; 2.2.25, section 2.2.6, 2,4,6,9

Tier: 1
Group: 2
K/A System: 295004
K/A Number: AA1.02
K/A Value: 3.8
Cognitive Level: 2
Bank/Mod/New: Bank

Distracter a: PMIS cannot be used as a sole source.
Distracter c: NBPP is not available as DIV I DC is de-energized and no AC power is available to the temperature recorder, PMIS cannot be used as a sole source.
Distracter d: NBPP is not available as DIV I DC is de-energized and no AC power is available.

Proposed references to be provided to applicants during the examination: None.

1. SYMPTOMS

1.1 Annunciator C-1/A-1, 250 VDC SWGR BUS 1A BLOWN FUSE, alarms.

1.2 Annunciator C-4/A-6, 250 VDC SWGR BUS 1B BLOWN FUSE, alarms.

1.3 Annunciator C-1/C-1, 250 VDC BATT CHARGER 1A TROUBLE, alarms.

1.4 Annunciator C-4/C-6, 250 VDC BATT CHARGER 1B TROUBLE, alarms.

1.5 Annunciator C-1/B-1, 250 VDC BUS 1A GROUND, alarms.

1.6 Annunciator C-4/B-6, 250 VDC BUS 1B GROUND, alarms.

2. AUTOMATIC ACTIONS

2.1 NBPP will transfer automatically to MCC-R, if 250 VDC System A or Static Inverter 1A is lost.

3. IMMEDIATE OPERATOR ACTIONS

3.1 At Panel C, check NO-BREAK POWER PANEL VOLTS meter is indicating ~ 240 volts.

3.1.1 If voltage indicated is low, transfer NO-BREAK POWER PANEL to MCC-R by placing switch on Panel C to MCC-R.

4. SUBSEQUENT OPERATOR ACTIONS

4.1 If NBPP is <u>not</u> energized, take action per Procedure 2.4.6.7 in conjunction with this procedure.

4.2 Monitor affected 250 VDC bus voltage and current if batteries are keeping bus energized.

4.3 If 250V CHARGER 1A (1B) or 250 VDC BATTERY CHARGER 1A (1B) feeder breaker on 250 VDC SWITCHGEAR 1A (1B) has failed and cannot be returned to service, place 250V CHARGER 1C in service per Procedure 2.2.24.

4.21 When fault has been determined and isolated, energize NBPP per Procedure 2.2.22.

5. PROBABLE CAUSE

5.1 Electrical fault on NBPP.

5.2 Blown fuse.

6. DISCUSSION

6.1 Power from the NBPP is obtained through an inverter fed from 250 VDC Bus A. The inverter feeds NBPP through a static switch inside the inverter or a manual bypass switch on the inverter. An emergency AC power feed is also provided from MCC-R. When inverter output voltage or frequency is abnormal, the internal static switch will automatically transfer to MCC-R. This static switch can also be transferred to MCC-R using the NBPP PWR TRANSFER switch on Panel C or by depressing the ALTERNATE SOURCE SUPPLYING LOAD button on the inverter. The NBPP power supply can also be transferred by placing the MANUAL BYPASS SWITCH on the inverter to ALTERNATE SOURCE TO LOAD. The NBPP is necessary for the operation of the station but is not critical to station safety. This procedure outlines the actions to be taken in the event of the failure of any of the supplies to the NBPP.

6.2 NBPP feeds the following major loads: reactor vessel level controllers and instrumentation, high off-gas activity isolation logic timers and valve control power, ERP flow indicating transmitter which sends process flow signal to ERP Kaman, Gaitronics, rod select power, rod position information system, NAWAS System, Ronan CRTs and printers, neutron monitoring recorders, condensate pump, condensate booster pump and reactor feed pump minimum flow valve control power and alarms, Reactor Building exhaust plenum and drywell high range radiation recorders, main generator voltage regulator alarms, fire protection manual pull stations and alarms, REC System low pressure alarms and low pressure non-essential isolation valve logic, and the SGT System low flow to stack alarm.

6.3 NBPP also supplies backup power to DEH and the RFPT speed controllers. NBPP can also supply power to drywell fan coil unit temperature recorders, drywell nitrogen purge controls, drywell temperature indicators and alarm units, SW rad monitor sample flow selector, main condenser hotwell level indicators and controls, Kaman RICs and recorders, and PC-TR-24, SUPPR POOL TEMP RECORDER, when the NORMAL/ALTERNATE POWER SUPPLY - DW TEMP RECORDERS & RECORDERS - TORUS TEMP RECORDER TR-24 switch is placed to ALT.

Q#	Question Description
RO18	ILT

Rev #	Rev Date	Topic Area	Diff
1	02/24/01		

Q type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C		1		

Objective #	Reference	K/A #	10CFR 55 41/43/45
COR002-18-02-8.d	COR002-18-02	295008 AK3.08	41(b).7

K/A Text:

AK3.08 – Knowledge of the reasons for the following responses as they apply to HIGH REACTOR WATER LEVEL: RCIC steam supply valve closure.

Question:

Which one of the following explains why the Reactor Core Isolation Cooling (RCIC) system is automatically shutdown on high RPV water level?

a. Prevents tripping the feedwater pumps to allow them to be used for level control.

b. Protects the RCIC steam line piping from damage and flooding by isolating the steam line.

c. Protects RCIC turbine blades from damage caused by low quality steam due to moisture carryover.

d. Prevents tripping the main turbine on high level to maximize use of the main condenser as a heat sink.

Answer:

ANSWER: c. Protects RCIC turbine blades from damage caused by low quality steam due to moisture carryover

REFERENCE: COR002-18-02

K/A System: 295008
K/A Number: AK3.08
K/A Value: 3.4
Cognitive Level: 1

Justification:

Distracter a: This does not prevent a feed pump trip.

Distracter b: Does not protect the steam lines, if level rose high enough to flood the steam lines there would be water in the line up to the valve.

Distracter d: Does not prevent tripping the main turbine.

SOURCE: SOURCE: Cooper Exam Bank

extinguish. At the same time, the trip-throttle valve red open light comes on.

LO-11d

2) The high water level turbine shutdown (≤54") signals the turbine steam supply block valve (MO-131) to close to prevent turbine damage from water in the steam line. Turbine coastdown causes a loss of oil to the oil trip cylinder and closure of the turbine trip-throttle valve. When the motor operated trip reset valve (MO-14) receives a signal that both the turbine steam inlet and the trip-throttle valves are closed, it will automatically relatch the trip linkage and reopen the trip-throttle valve. This operation now places the RCIC system in standby status for automatic restart on a low water level initiation signal without any required operator action.

Fig 6

3) Reset of the turbine overspeed condition requires operator action locally at the turbine. To perform the overspeed reset, the operator must first verify visually that the RCIC turbine is not damaged due to the excessive speed that it experienced. At the front of the RCIC turbine, the trip linkage rod for the overspeed must be moved toward the turbine trip throttle valve ensuring that the overspeed ball tappet on the turbine shaft casing is in a vertical position. The Control Room operator may now reset the trip as described previously by closing the steam supply blocking valve (MO-131).

CAUTION

On a turbine overspeed trip, the Control Room operator should not close the turbine steam supply block valve (MO-131) immediately. If MO-131 is closed before resetting the linkage locally, the motor operated trip reset valve (MO-14) will attempt to automatically reset the turbine trip as in the high water level trip reset operation. Since the overspeed trip linkage is still in the displaced condition, the motor operated trip reset valve will cycle continuously attempting to reset the trip. The cycling may cause the motor operator to overheat.

Fig 4
SO-02e
LO-01f

6. Governor Control System

a. The governor control is an electric-hydraulic control system used to adjust RCIC governor valve position in order to maintain a constant system flow rate to the reactor.

LO-01f; 10i,o

b. The governor valve is a single-seated stainless steel valve which is used to throttle the reactor steam admitted to the RCIC turbine in order to control turbine speed. The valve is hydraulically closed using oil pressure from the lube oil system, and is opened by spring pressure. On a loss of oil or control signal the valve will fail to its open position.

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO19	ILT	0	02/15/01	HIGH DRYWELL PRESSURE	

Q type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C		1		

Objective #	Reference	K/A #	10CFR 55 41/43/45
	Figure 9, EOP/SAG Graphs	295012 AA2.02	41(b).10

K/A Text:

AA2.02 – Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE: Drywell pressure.

Question:

A steam line break in the drywell has occurred while at power. Drywell temperature is approaching 280°F. The SRO is trying to determine if Drywell Sprays can be initiated per the DW Temperature leg of EOP–3A.

Which one of the following **parameter** values will the SRO need to make this determination?

a. Torus Temperature

b. RPV Pressure

c. Torus Pressure

d. Drywell pressure

Answer:

ANSWER: d. Drywell pressure

REFERENCE: Figure 9, EOP/SAG Graphs

K/A System: 295012
K/A Number: AA2.02
K/A Value: 3.9
Cognitive Level: 1

Justification: The SRO must check the drywell spray initiation limit curve, which plots containment pressure against drywell temperature.

Distracter a: not needed

Distracter b: not needed

Distracter c: not needed

SOURCE: NEW

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO20	ILT		01/03/01	HIGH SUPPRESSION POOL TEMP	

Q type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C		1		

Objective #	Reference	K/A #	10CFR 55 41/43/45
	STOMT0080613 Flowchart 3A	295013 2.4.18	41(b).10

K/A Text:

2.4.18 – Knowledge of the specific bases for EOPs

Question:

While performing the Torus Temperature Control leg of EOP-3A, Primary Containment Control, the operator is directed to enter EOP-1A, RPV Control, AND execute it concurrently before Torus Temperature reaches 110°F

Which one of the following is the bases for entering EOP-1A, RPV Control, AND executing it concurrently without a specific entry condition being met?

a. This ensures Torus temperature is maintained below 120°F while the reactor is being shutdown.

b. This directs a scram and removes the source of a potential energy addition to the Torus before conditions warrant injection of boron.

c. This assumes the high drywell temperatures are from a primary system break that will require emergency core cooling systems for RPV level control.

d. This provides direction for reactor pressure control and a path for emergency depressurization using the turbine Bypass Valves if temperatures continue to rise.

Answer:

ANSWER: b. This directs a scram and removes the source of a potential energy addition to the Torus before conditions warrant injection of boron.

REFERENCE: STOMT0080613 Flowchart 3A – Primary Containment Control, Page 20, Section II.J.3, Rev 9

K/A System: 295013
K/A Number: 2.4.18
K/A Value: 2.7
Cognitive Level: 1

Justification: Ensures a scram is initiated because EOP-3A does NOT require a scram

Distracter a: There is no guarantee this temperature will NOT be exceeded

Distracter c: Entry conditions for RPV control ensure RPV level control

Distracter d: Pressure control becomes the bases is the reactor scram is NOT sucessful

SOURCE: Cooper Exam Bank

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO21	ILT	0	01/03/01	Control Room Abandonment	

Q type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C		1		COR002-34-02

Objective #	Reference	K/A #	10CFR 55 41/43/45
COR002-11-02-13	5.2.1, COR002-34-02	295016 2.4.2	41(b).7, 41(b).8, 41(b0.10

K/A Text:

2.4.2 – Knowledge of system setpoints/interlocks and automatic actions associated with EOP entry conditions.

Question:

Given the following conditions:

- The Control Room is abandoned
- ALL Immediate Actions of Procedure 5.2.1, "Shutdown from Outside the Control Room", are complete, **NO** further actions have been taken.
- Reactor water level is 35 inches (Wide Range)
- Drywell pressure is 0.3 psig

Which one of the following statements below describes the **CURRENT** status of the High Pressure Coolant Injection (HPCI) system?

a. HPCI can <u>only</u> be started from the ASD room and will be available for automatic initiation.

b. HPCI can <u>only</u> be started from the control room and will be available for automatic initiation.

c. HPCI can <u>only</u> be started from the ASD room and will NOT be available for automatic initiation.

d. HPCI can <u>only</u> be started from the control room and will NOT be available for automatic initiation.

Answer:

ANSWER: b.

REFERENCE: EP 5.2.1, COR0023402 ASD

K/A System: 295016
K/A Number: 2.4.2
K/A Value: 3.9
Cognitive Level: 1

Justification: HPCI control is not shifted in the immediate actions and remains operable from the control room with full auto functions.

Distracter b: HPCI CANNOT be started from the HPCI panel and auto functions are operable.

Distracter c: HPCI CANNOT be started from the HPCI panel.

Distracter d: HPCI CANNOT be started from the HPCI panel and auto functions are operable.

SOURCE: NEW

Revise

LO-051 J. Condensate Makeup

LO-10d The HPCI system piping is maintained full of water from the condensate makeup (Pressure Maintenance) system. This prevents damage to the HPCI system due to water hammer.

LO-10i K. Reactor Equipment Cooling System

Cooling water to the HPCI room fan coil unit is supplied from the REC system for maintaining the room temperature within habitable limits, and preventing a Group 4 isolation due to high area temperature of ≤195°F.

REC-MO-711 or 714 must be manually opened before starting HPCI to provide adequate cooling. The valves only automatically open on a Group 6 isolation signal.

LO-08p,09i L. Alternate Shutdown Room

1. Isolation switches on the HPCI Panel transfer control of specified HPCI system components from the Control Room to the ASD Room Panel. The following HPCI system components can be controlled from the ASD Room:

   a. Steam Supply Valves (MO-14, 15, and 16)

   b. Auxiliary Oil Pump

   c. Gland Seal Condenser Condensate Pump

   d. Gland Seal Condenser Blower

   e. ECST Test Line Valves (MO-21, 24)

   f. ECST Suction Valve (MO-17)

   g. Pump Discharge Valve (MO-20)

   h. Injection Isolation Valve (MO-19)

   i. Minimum Flow Valve (MO-25)

j. Torus Suction Valve (MO-58)

k. Indicators

l. Fan Coil Unit

m. HPCI Flow Controller

n. Isolation Switches for the components listed above

2. A HPCI auto start is bypassed when the ASD HPCI Panel "Control and Indication" switch is placed in ISOLATE.

3. The AOP still operates automatically 20 - 85 psig.

LO-13

4. All other interlocks are bypassed. HPCI will only trip on overspeed or a manual trip. The high level trip is bypassed. The automatic suction transfer circuit is also bypassed, so ECST level must be monitored. The interlock between MO-17 and MO-58 still exists such that both valves cannot be closed at the same time.

LO-10e,r

M. Nuclear Boiler Instrumentation

1. Narrow Range Barton LIS-101B(D) causes a HPCI turbine trip at $\leq$54". If cold reference leg 3B had a leaking equalizer valve or there was a leak in the reference line, an inadvertent HPCI trip could occur on high vessel level.

2. Wide Range instruments LIS-72A-D (#3 contacts) produce a HPCI system initiation at $\geq$-42". If the hot reference legs for condensing chambers 2A and 2B were to leak or if their equalizing valves leaked by, HPCI would not start on low level and would require manual initiation.

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO22	ILT	0	01/03/01	High Off-site Release Rate	

Q type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C		1		

Objective #	Reference	K/A #	10CFR 55 41/43/45
COR0011802001050Q COR0011802001080B	Radiation Monitoring Text	295017 AK3.01	41(b).7, 41(b).11, 41(b).13

K/A Text:

AK3.01 – Knowledge of the reasons for the following responses as they apply to HIGH OFF-SITE RELEASE RATE: System Isolations

Question:

Which one of the following describes the radwaste liquid effluent process radiation monitor protective function?

a. Closes the Radwaste effluent valve before federal limits are exceeded.

b. Trips the discharge pump and closes the radwaste effluent valve when federal limits are reached.

c. Provides an alarm only to alert control room and radwaste operators before federal limits are exceeded.

d. Trips the discharge and dilution pumps and closes the radwaste effluent valve when federal limits are reached.

Answer:

ANSWER: a. Closes the Radwaste effluent valve before federal limits are exceeded.

REFERENCE: Radiation Monitoring Text

K/A System: 295017
K/A Number: AK3.01
K/A Value: 3.6
Cognitive Level: 1

Justification: Closes the valve to isolate and stop the discharge

Distracter b: Does not trip any pumps and initiates prior to reaching federal limits.

Distracter c: Provides an effluent valve trip

Distracter d: Does not trip any pumps and initiates prior to reaching federal limits.

SOURCE: Cooper Exam Bank

LO-05m
SO-02d,02m

The liquid process radiation monitoring system consists of three channels which monitor the radiation level of the Service Water effluent, Radwaste effluent, and Reactor Equipment Cooling water. Each channel makes a continuous permanent record of the observed radiation level. If the observed radiation level exceeds the acceptable limits (upscale or downscale), the effected monitoring channel will activate the appropriate annunciator alarm. In the case of Radwaste effluent it will also trip closed the effluent valve terminating Radwaste release before limits in 10CFR20 are exceeded.

B. Design Basis

LO-03i

1. Safety Design Basis

Indicate when operation limits are exceeded for the normal release of radioactive material from those process streams that normally discharge to the environs for Service Water and Radwaste effluent. In the case of REC it is used to show a leak into the system and a means to determine the source of the leak.

2. Power Generation Design Basis

Process liquid radiation monitors which are located in streams that may become contaminated. The monitors will provide a clear indication to the operator whenever the radioactivity level in the stream reaches or exceeds a pre-established limit above the normal radiation level of the stream indicating.

LO-02

3. Technical Specifications - none

4. Offsite Dose Assessment Manual

a) D 3.1.1, Liquid Effluents Concentration.
b) D 3.1.2, Liquid Waste Concentration.
c) D 3.1.3, Liquid Effluents Dose.
d) D 3.3.1, Liquid Effluent Monitoring.
e) D 3.3.3, Liquid Radwaste Discharge Isolation.
f) D 3.4.1, Liquid/Gaseous Effluents Dose.

LO-06a
SO-06a

C. Power Supplies

1. REC monitor - LPREMG
2. SW monitor - LPREMG
3. Radwaste Monitor - LPRW

Fig 4

D. System Components Overview

1. The REACTOR EQUIPMENT COOLING (REC) and Service Water discharge monitors each consists of:

a. A scintillation detector

b. A pulse preamplifier

c. A process radiation monitor

d. A recorder

e. A shared trip auxiliary unit

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO23	ILT	0	02/14/01	Partial or Complete Loss of CCW	

Q type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C		1		

Objective #	Reference	K/A #	10CFR 55 41/43/45
	5.2.4	295018 2.4.24	41.4, 41.7, 41.10

K/A Text:

2.4.24 – Knowledge of loss of cooling water procedures.

Question:

The unit is operating at 100% power when the following alarms are energized:

- M-1/A-1, REC SYSTEM LOW PRESSURE
- M-1/A-3, REC SURGE TANK LOW LEVEL

Three (3) REC pumps are operating but REC pressure is lowering. Which one of the following actions is required immediately?

a. Shutdown both reactor recirculation pumps.

b. Isolate the RWCU System to raise the cooling capacity of the REC system.

c. Shutdown the REC Pumps and Scram the reactor anticipating a loss of REC.

d. Close REC-MO-711, NORTH CRITICAL LOOP SUPPLY to conserve pressure.

Answer:

ANSWER: c. Shutdown the REC Pumps and Scram the reactor anticipating a loss of REC.

REFERENCE: 5.2.4

K/A System: 295018
K/A Number: 2.4.24
K/A Value: 3.3
Cognitive Level: 1

Justification: These alarms indicate a break in the REC piping that will exceed makeup capacity and result in a loss of ability to cool the recirc pumps and reactor auxiliary equipment. The system must be shutdown, the reactor scrammed, and then limited cooling may be accomplished with one pump.

Distracter a: Starting a fourth pump is necessary if pressure is low, but, with a low tank level a break is indicated and starting a fourth pump would raise the leak rate.

Distracter b: This is a subsequent action after the reactor has been shutdown

Distracter d: This valve automatically isolates when REC pressure falls to 40 psig, there are no immediate actions to close it.

SOURCE: NEW

COOPER NUCLEAR STATION OPERATIONS MANUAL EMERGENCY PROCEDURE 5.2.4 LOSS OF ALL REACTOR EQUIPMENT COOLING (REC) WATER	CLASS: REFERENCE USE ⊕ EFFECTIVE: 4/2/00 APPROVAL: SORC OWNER: D. W. BREMER DEPARTMENT: OPS

1. SYMPTOMS

1.1 Annunciator M-1/A-1, REC SYSTEM LOW PRESSURE, alarms.

1.2 Annunciator M-1/A-3, REC SURGE TANK LOW LEVEL, alarms.

1.3 Drywell temperature and pressure are rising.

1.4 The temperature of equipment cooled by REC is rising.

1.5 Low REC flow alarms on VBD-M.

1.6 Pump failure alarms on VBD-M.

1.7 Low REC System pressure.

2. AUTOMATIC ACTIONS

CAUTION - If pumps trip on loss of power and normal power is restored prior to emergency power energizing 4160V Bus 1F and 1G, REC pumps will not automatically start.

2.1 REC pumps selected to standby will automatically start 20 seconds after 4160V Bus 1F and 1G are energized by emergency power.

2.2 Following valves close when REC header pressure drops below specified pressure and a 40 second time delay has timed out:

2.2.1 REC-MO-700, NON-CRITICAL HEADER SUPPLY (61.2 psig).

2.2.2 REC-MO-702, DRYWELL SUPPLY ISOLATION (61.2 psig).

2.2.3 REC-MO-712, HX A OUTLET (62.4 psig).

2.2.4 REC-MO-713, HX B OUTLET (60.2 psig).

2.2.5 REC-MO-1329, AUGMENTED RADWASTE SUPPLY (61.2 psig).

3. IMMEDIATE OPERATOR ACTIONS

3.1 Ensure REC-MO-700, REC-MO-702, REC-MO-712, REC-MO-713, and REC-MO-1329 have closed.

3.2 Immediately attempt to restore REC System by starting available pumps or isolating leak if possible.

3.3 If annunciators M-1/A-1, REC SYSTEM LOW PRESSURE, and M-1/A-3, REC SURGE TANK LOW LEVEL, alarm simultaneously, perform following:

3.3.1 Shut down all operating REC pumps.

3.3.2 Close REC-MO-709, DRYWELL RETURN ISOLATION.

3.3.3 Scram reactor.

3.3.4 Restart one REC pump to supply cooling water to equipment connected to critical cooling loops.

3.3.5 Open REC-MO-711, NORTH CRITICAL LOOP SUPPLY, and/or REC-MO-714, SOUTH CRITICAL LOOP SUPPLY.

4. SUBSEQUENT OPERATOR ACTIONS

4.1 Notify Shift Supervisor.

4.2 Determine if declaration of an EAL per Procedure 5.7.1 is appropriate.

4.3 If REC System cannot be restored within ~ 1 minute, perform following:

4.3.1 Scram reactor if not already scrammed.

4.3.2 Isolate RWCU System.

4.3.3 Shut down both recirculation pumps and associated oil pumps when MG Sets have stopped.

4.4 If SW flow to REC HX has been lost, enter Procedure 2.4.8.3.1 or Procedure 5.2.3, as appropriate.

4.5 Dispatch an Operator to REC pump/heat exchanger area to evaluate and inspect system.

4.6 Vent Drywell through SGT System if loss of cooling is causing Drywell pressure to rise.

4.7 De-inert Drywell per Procedure 2.2.60 to remove some of the REC heat load and help cool Drywell.

4.8 If REC pumps are supplying only critical loops and REC Surge Tank level has not returned to normal, perform following:

4.8.1 Determine if REC Surge Tank Level Control Valve has failed and bypass, as required.

4.8.2 If REC Surge Tank level cannot be restored, split REC critical loops per Procedure 2.2.65.1.

4.9 If an REC pump is cavitating or does not have a flow path, shut down pump.

4.10 If critical loops are required to be operating and REC pumps are unable to supply cooling to critical subsystem, use service water backup per Procedure 2.2.65.1.

4.11 Monitor CSCS Quad temperatures and perform following applicable steps to maintain Annunciator R-2/A-5, REACTOR BLDG PUMP ROOM HIGH TEMP, clear:

4.11.1 Shut down affected equipment if temperature rises 50°F above normal, unless equipment is required to assure adequate core cooling, inject boron, or suppress a working fire.

WARNING - Protective clothing and/or SCBAs may be required to enter area with high temperature.

4.11.2 Establish fire watches and open following doors to establish or raise natural circulation flow through affected quad:

4.11.2.1 NE and SE Quads - Open 903 level door to quad.

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO24	ILT	0	01/05/01	Part. or Comp. Loss of Instr. Air	

Q type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C		1		

Objective #	Reference	K/A #	10CFR 55 41/43/45
	EP-5.2.8,	295019 AA1.01	41(b).7

K/A Text:

AA1.01 – Ability to operate and monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: Backup air supply

Question:

A malfunction of the Instrument Air Dryers has raised their d/p high enough to restrict instrument air flow and cause a lowering of instrument air pressure. Which one of the following actions is required to restore instrument air pressure?

a. Open SA-MO-81, SA to IA CROSSTIE.

b. Open SA-14, AIR RECEIVER 1A 6" OUTLET.

c. Close SA-PCV-609, SERVICE AIR SYSTEM ISOLATION.

d. Close IA-MO-80, NON CRITICAL INSTRUMENT AIR ISOLATION.

Answer:

ANSWER: a. Open SA-MO-81, SA to IA CROSSTIE.

REFERENCE: EP-5.2.8, Loss of Instrument Air

K/A System: 295019
K/A Number: AA1.01
K/A Value: 3.5
Cognitive Level: 1

Justification: SA-MO-81, SA to IA CROSSTIE allow Service Air to bypass the Instrument Air Dryers and re-supply IA.

Distracter b: This is a normally open valve

Distracter c: This valve automatically closes on low air pressure (77 psig) and would not raise or restore pressure.

Distracter d: This valve is manually closed to isolate non-essential loads and would not raise or restore pressure.

SOURCE: New

1. SYMPTOMS

1.1 Annunciator A-4/A-4, AIR RECEIVER A OR B LOW PRESSURE, alarms.

1.2 Annunciator A-4/A-5, CONTROL AIR LOW PRESSURE, alarms.

1.3 Annunciator A-4/B-4, SERVICE AIR ISOLATION PCV-609, alarms.

1.4 Annunciator A-4/B-5, SERVICE AIR LOW PRESSURE, alarms.

1.5 Annunciator A-4/F-5, AIR DRYER TROUBLE, alarms.

1.6 Annunciator A-4/G-4, INTAKE BLDG CONTROL AIR LOW PRESSURE, alarms.

1.7 Annunciator 9-3-1/C-2, DRYWELL PNEUMATIC HDR LOW PRESSURE, alarms; if header was supplied from instrument air and nitrogen supply was secured.

1.8 Annunciator 9-5-1/C-4, ROD DRIFT, alarms; caused by control rods inserting due to scram valves opening.

1.9 Annunciator 9-5-2/B-2, MSIV NOT FULL OPEN TRIP, alarms; caused by following:

1.9.1 Outboard MSIVs drifting shut when in MODE 1.

1.9.2 Inboard MSIVs drifting shut if drywell pneumatic header was being supplied by instrument air and nitrogen supply was secured when in RUN Mode.

1.10 Annunciator 9-5-2/F-5, SCRAM VALVE PILOT AIR LOW PRESSURE, alarms.

2. AUTOMATIC ACTIONS

2.1 Standby air compressors start as system air pressure drops.

2.2 SA-PCV-609, Service Air System Isolation, closes when service air pressure drops to < 77 psig.

2.3 Reactor scram if in MODE 1 due to MSIV closure.

3. IMMEDIATE OPERATOR ACTIONS

3.1 Ensure all available air compressors are running.

3.2 If instrument air pressure drops low enough to cause more than one control rod to insert, scram reactor.

4. SUBSEQUENT OPERATOR ACTION

4.1 If air drying and filtering components are at fault, open SA-MO-81, SA TO IA CROSSTIE (Panel A), to supply service air to Instrument Air System.

4.1.1 Minimize period of time that SA-MO-81 is open. Attempt to place in service one IA dryer with pre and post filters. If necessary, manually bypass an obstructed component and use available IA dryers and/or filters to provide driest, cleanest air possible to IA System.

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO 25	ILT	0	2/2001	HPCI	

Q Type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C				COR002-11-02

Objective #	Reference	K/A #	10CFR 55 41/43/45
8	2.1.22	295020, AK2.06	41(b)(7) 41(b)(8)

K/A Text:

AK2.06 – Knowledge of the interrelations between INADVERTENT CONTAINMENT ISOLATION and the following: HPCI

Question:

A false high drywell pressure signal caused an automatic initiation of HPCI. An operator then depresses the <u>Manual Isolate pushbutton</u> instead of the <u>Turbine Trip pushbutton</u> on the 9-3 panel when attempting to secure HPCI.

Which one of the following describes the HPCI system response?

a. HPCI Inboard Steam Isolation valve, HPCI-MO-15 closes, the ECST Suction valve HPCI-MO-17 receives a close signal and the HPCI turbine trips.

b. HPCI Outboard Steam Isolation valve, HPCI-MO-16 closes, the Suppression Pool suction valve HPCI-MO-58, receives a close signal and the HPCI turbine trips.

c. Both HPCI Inboard and Outboard Steam Isolation valves, HPCI-MO-15 and HPCI-MO-16, close and both HPCI Suction valves, HPCI-MO-17 and HPCI-MO-58 receive a close signal and the HPCI turbine trips.

d. Both HPCI Inboard and Outboard Steam Isolation valves, HPCI-MO-15 and HPCI-MO-16, close and both HPCI Suction valves, HPCI-MO-17 and HPCI-MO-58 receive a close signal, HPCI turbine coasts down but does **NOT** trip.

Answer:

ANSWER: b.

REFERENCE: 2.1.22

Tier: 1
Group: 2
K/A System: 295020
K/A Number: AK2.06
K/A Value: 3.8
Cognitive Level: 2
Bank/Mod/New: Bank

Distracter a: This is logic A which is not tripped by the manual pushbutton.
Distracter c: The manual pushbutton only trips logic B.
Distracter d: The manual pushbutton only trips logic B and the turbine trips on an isolation signal.

Proposed references to be provided to applicants during the examination: None.

NOTE - Rejecting water to radwaste for at least 1 minute ensures bypass line has been flushed of any resin or trapped air.

[ ] 6.8.2 If RWCU-MO-15 and RWCU-MO-18 can be opened but pumps/filter demineralizers cannot be placed in service, maintain reactor water chemistry parameters by rejecting reactor water to radwaste for at least 1 minute, and then to main condenser, if available, per Procedure 2.2.66.

[ ] 6.8.3 If RWCU-MO-15, INBD ISOL VLV, and RWCU-MO-18, OUTBD ISOL VLV, will be closed > 26 hours and average reactor coolant temperature is > 212°F, take long-term isolation corrective action per Procedure 2.2.66.©

7. GROUP 4 ISOLATION

7.1 Upon 1/2 Group 4 Isolation, following will occur:

[ ] 7.1.1 If Logic A trips, HPCI-MO-15 closes, HPCI turbine trips, and HPCI-AO-70 and HPCI-AO-71 close, if open.

[ ] **NOTE** - Manual actuation of Group 4 Isolation (Panel 9-3) trips Logic B only.

[ ] 7.1.2 If Logic B trips, HPCI-MO-16 and HPCI-MO-58 close, HPCI turbine trips, and HPCI-AO-70 and HPCI-AO-71 close, if open.

7.2 Upon full Group 4 Isolation, ensure following valves are closed (Panel 9-3):

[ ] 7.2.1 HPCI-MO-16, STM SUPP OUTBD ISOL VLV.

[ ] 7.2.2 HPCI-MO-15, STM SUPP INBD ISOL VLV.

[ ] 7.2.3 HPCI-MO-58, TORUS PUMP SUCT VLV (Panel 9-3), if HPCI-MO-17, ECST PUMP SUCT VLV, is full open.

[ ] 7.2.4 HPCI-AO-70 and HPCI-AO-71, TURB EXH LINE DR POT TO GLD SEAL CNDSR, valves (Panel 9-3).

7.3 Ensure HPCI turbine has tripped.

7.4 Close RHR-MO-920, SUPPLY VALVE (Panel 9-3).

7.5 Close RHR-MO-921, SUPPLY VALVE (Panel 9-3).

7.6 Shut down AOG, as required, per Procedure 2.2.58, 2.2.58.3, or 2.2.58.4.

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO26	ILT	0	02/13/01	High Suppression Pool Water Temp	

Q type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C		1		

Objective #	Reference	K/A #	10CFR 55 41/43/45
	EOP/SAG Graphs, Graph 5	295026 EK2.03	41.5, 41.14, 45.7, 45.8

K/A Text:

EK2.03 – Knowledge of the interrelations between SUPPRESSION POOL HIGH WATER TEMPERATURE and the following: Suppression chamber pressure: Mark I & II

Question:

A LOCA resulted in the following conditions:

- RPV level –135 inches and steady (Wide Range)
- Drywell pressure 8 psig
- Drywell temperature 225°F
- Torus pressure 6 psig
- Suppression Pool Temperature 215°F
- Containment Level 14 feet
- RHR Pump "A" flow into the RPV 8000 gpm

Which one of the following describes the effect of using of "A" RHR pump for torus sprays at this time?

a. Adequate core cooling is NOT assured.

b. The primary containment boundary will fail.

c. RHR Pump "A" NPSH requirement will NOT be met.

d. Reactor Building to Torus Vacuum Breakers will open.

Answer:

ANSWER: c. RHR Pump A NPSH requirement will NOT be met.

REFERENCE: EOP/SAG Graphs, Graph 5

K/A System: 295026
K/A Number: EK2.03
K/A Value: 3.2
Cognitive Level: 3

Justification: Torus sprays would lower torus pressure, which would lower torus overpressure, which would lower NPSH below the limit in Graph 5 of the EOP.

Distracter a: Adequate core cooling is assured – water level is above TAF

Distracter b: Torus sprays require very little flow, there is plenty available.

Distracter d: Reactor Building to Torus Vacuum Breakers would NOT open.

SOURCE: NEW

Q#	*Question Description*	*Rev #*	*Rev Date*	*Topic Area*	*Diff*
RO27	ILT	0	01/05/01	High Drywell Temperature	

Q type	*Response Time*	*Max Point Value*	*Passing Point Value*	*Lesson #*
M/C		1		

Objective #	*Reference*	*K/A #*	*10CFR 55 41/43/45*
	2.4.8.4.2	295028 EK3.05	41.7, 45.3

K/A Text:

EK3.05 - Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL TEMPERAURE: Reactor SCRAM.

Question:

Following a loss of Drywell Cooling which one of the following requires a reactor scram?

a. Drywell cooling is lost and is unavailable.

b. Wide range level indication run temperatures reach 197°F.

c. Drywell temperature reaches 150°F and EOP entry is required.

d. Reactor water level indications are effected by drywell temperature.

Answer:

ANSWER: a. Drywell cooling is lost and is unavailable.

REFERENCE: Abnormal Procedure 2.4.8.4.2, Ventilation System Failure - Loss of Drywell Cooling

K/A System: 295028
K/A Number: EK3.05
K/A Value: 3.6
Cognitive Level: 1

Justification: Rising Drywell temperatures will cause a high drywell pressure and Scram and ECCS initiation, 2.4.8.4.2 requires entry into 2.1.5, Reactor Scram, when drywell cooling cannot be restored.

Distracter b: At this temperature wide range level is not effected, and no scram is required.

Distracter c: EOP entry (EOP-3A) does not require a scram

Distracter d: Although this is a major concern, it does not require a scram for these conditions.

SOURCE: New

1. SYMPTOMS

1.1 Annunciator(s) H-1/A-2 (H-1/B-2 and/or H-1/C-2), DRYWELL ZONE 1 (2C and/or 3) HIGH TEMP, alarms.

1.2 Annunciator(s) H-1/A-1 (H-1/B-1, H-1/C-1, and/or H-1/D-1), DRYWELL FCU A (B, C, and/or D) HIGH DISCH TEMP, alarm(s).

1.3 Annunciator 9-5-2/F-3, DRYWELL HIGH PRESSURE, alarms.

2. AUTOMATIC ACTIONS

2.1 Reactor scram on drywell pressure of ≤ +1.84 psig.

2.2 Trip of the drywell FCUs on high drywell pressure ≤ +1.84 psig or low reactor water level ≥ -113".

2.3 PCIS Group 2 and 6 Isolation at ≤ +1.84 psig in drywell.

2.4 RHR, CS, HPCI, and DG initiate on high drywell pressure of ≤ +1.84 psig.

3. IMMEDIATE OPERATOR ACTIONS

CAUTION - If the diesel generators are supplying power to 4160 V Buses 1F and 1G, verify that the FCUs to be started will not overload the diesel generators.

3.1 Start drywell FCUs by placing their control switches to OVERRIDE.

3.2 Maintain drywell pressure ≤ 0.75 psig by venting via SGT.

4. SUBSEQUENT OPERATOR ACTION

NOTE - Be aware that elevated drywell temperatures affect RPV water level indications.

4.1 Verify REC-MO-702, DRYWELL SUPPLY ISOLATION, and REC-MO-709, DRYWELL RETURN ISOLATION, are open.

4.2 If drywell pressure cannot be maintained ≤ 0.75 psig, enter applicable Conditions and Required Actions of LCO 3.6.1.4.

4.3 If drywell average air temperature cannot be maintained ≤ 150°F, enter applicable Conditions and Required Actions of LCO 3.6.1.5.

4.4 If drywell cooling is lost and cannot be restored, shutdown the reactor per Procedure 2.1.5.

4.5 Attempt to lower drywell temperature by performing the following:

4.5.1 Initiate a reactor cooldown at the maximum allowable rate.

4.5.2 If plant conditions allow, de-inert containment; otherwise, purge the drywell with N_2, refer to Procedure 2.2.60.

4.5.3 If drywell pressure cannot be maintained below 1.84 psig or average drywell temperature cannot be maintained below 150°F, enter the EOPs.

5. PROBABLE CAUSE

5.1 Electrical or mechanical failure of the Drywell Fan Coil Units.

6. DISCUSSION

6.1 This procedure outlines the actions to be taken by station personnel to maintain reactor safety upon loss of any or all of drywell FCUs.

6.2 The major concern of high drywell temperatures is the effect elevated temperatures can have on RPV water level indications. Also, when drywell temperatures approach 300°F or more, equipment such as solenoids and wiring will start to fail.

6.3 Drywell heat load is primarily pressure related rather than power related since the major heat addition is due to temperature of piping and equipment. For this reason, power reduction alone will contribute little to reduction of excessive temperatures.

6.4 A total loss of drywell coolers, during power operation, will cause bulk drywell air temperature to initially increase rapidly. Reactor scram on high drywell pressure will occur in several minutes.

7. REFERENCES

7.1 TECHNICAL SPECIFICATIONS

7.1.1 LCO 3.6.1.4, Drywell Pressure.

7.1.2 LCO 3.6.1.5, Drywell Temperature.

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO28	ILT	0	01/03/01	High Suppression Pool Water Level	

Q type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C		1		

Objective #	Reference	K/A #	10CFR 55 41/43/45
	INT0080613, EOP-3A	295029 EK2.01	41.5, 41.7, 45.5

K/A Text:

EK2.01 – Knowledge of the interrelations between HIGH SUPPRESSION POOL WATER LEVEL and the following: RHR/LPCI

Question:

A loss of coolant accident has occurred with the following conditions:

• Reactor pressure	590 psig
• Drywell pressure	6.5 psig **AND** rising
• Drywell temperature	200°F
• Torus Temperature	160°F and rising
• Containment level	16.8 feet **AND** rising
• Torus spray is in service	
• Drywell Spray is in service	

Which one of the following actions is required and what are the reasons for those actions?

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

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

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

d. Terminate Drywell Spray because the primary containment vacuum relief is inoperable.

Answer:

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

REFERENCE: INT0080613, page 9-11, section II.F.3, rev. 9., EOP-3A

K/A System: 295029
K/A Number: EK2.01
K/A Value: 3.0
Cognitive Level: 2

Justification: Torus vacuum Breakers are submerged at 16.5' and will not pass sufficient flow to the Drywell.

Distracter a: Drywell Spray water is taken from the Torus.

Distracter b: The Spray header is submerged at 26.5'.

Distracter c: The Vacuum Breakers will not pass sufficient flow when covered.

MATERIAL REQUIRED FOR EXAMINATION: Flowchart 3A – Primary Containment Control

SOURCE: Cooper Exam Bank

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO 29	ILT	0	2/2001	SEC CONT CONTROL	

Q Type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C				INT008-06-17

Objective #	Reference	K/A #	10CFR 55 41/43/45
4	EOP-5A, INT008-06-17	295033, EK1.02	41(b)(10) 41(b)(12)

K/A Text:

EK1.02 – Knowledge of the operational implications of the following concepts as they apply to HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS: Personnel protection.

Question:

Which one of the following describes the EOP-5A, "Secondary Containment Control," basis for isolating a system discharging into the secondary containment?

a. To minimize reactor coolant losses.

b. To backup PCIS automatic functions.

c. To maintain the Recirc MG set room accessible.

d. To terminate rising temperatures and radiation levels.

Answer:

ANSWER: d.

REFERENCE: EOP-5A, INT008-06-17

Tier: 1
Group: 2
K/A System: 295033
K/A Number: EK1.02
K/A Value: 3.9
Cognitive Level: 1
Bank/Mod/New: Bank

Distracter a: This is covered by other EOPs
Distracter b: PCIS automatic actions may not have been required
Distracter c: Secondary Containment Control does not maintain habitability for all areas. The Max Safe values are based on equipment operability and personnel access necessary for EOP actions. the Recirc MG set room is not one of the areas requiring access.

Proposed references to be provided to applicants during the examination: None.

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO30	ILT	0	01/03/10	Sec Containment Vent. Rad. High	

Q type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C		1		COR002-28-02

Objective #	Reference	K/A #	10CFR 55 41/43/45
	2.2.73	295034 EA1.04	41.7, 41.11

K/A Text:

EA1.04 – Ability to operate and/or monitor the following as they apply to SECONDARY CONTAINMENT VENTILATION HIGH RADIATION: SBGR/FRVS

Question:

During surveillance testing of the Reactor Building Exhaust Radiation Monitors, the Mode Switches for "A" and "C" trip units where **NOT** returned to the OPERATE position.

Later, while moving contaminated refueling equipment, radiation levels in the area of the Reactor Building Exhaust Plenum rise to sixty (60) mr/hr. Standby Gas Treatment is in a normal lineup. Which one of the following describes the effects on SGT?

a. SBGT receives one half of an initiation signal and will **NOT** start.

b. SGT train B will automatically start and Secondary containment will isolate.

c. SGT train B will automatically start but Secondary containment will **NOT** isolate.

d. SGT trains A and B will automatically start and Secondary containment will isolate.

Answer:

ANSWER: d.

REFERENCE: 2.2.73

K/A System: 295034
K/A Number: EA1.04
K/A Value: 4.1
Cognitive Level: 2

Justification: The switches out of operate will produce one of the two signals necessary for complete system initiation. They will NOT effect the operability of the SBGT trains.

Distracter a: One half the signal is created by the switches out of operate, the other by the high radiation

Distracter b: Both trains start

Distracter c: Both trains start

MATERIAL REQUIRED FOR EXAMINATION: N/A

SOURCE: NEW

1.2.3.9 Air Operated Outlet Valve SGT-AO-251 (SGT-AO-252) opens when the SGT A (B) fan starts. The valve fails open on a loss of air or control power.

1.2.3.10 Air Operated Differential Pressure Control Valve SGT-DPCV-546A (SGT-DPCV-546B) opens when a Group 6 isolation signal is received to maintain negative Reactor Building pressure. The valve fails open on a loss of air or control power.

1.2.4 The SGT System discharges to the ERP through two 10" underground lines. These SGT discharge lines can potentially be blocked by excessively high water level in Z sump located at the base of the ERP. These SGT discharge lines have drain lines that are heat traced to prevent blockage due to freezing during cold weather conditions. Temperature of these drain lines are monitored to ensure proper operation of heat trace. Upon discovery of temperature below 70°F, if power is lost to heat trace, Attachment 2 is used to assist in determining continued SGT operability. Z sump pumps and support equipment are essential in support of the SGT System.©

1.2.5 When a Group 6 isolation signal is received, both SGT units start. The Operator can then select one subsystem to function with the other in the STANDBY Mode.

1.2.6 Cross-connections between the subsystems are provided to maintain the required decay heat removal cooling air flow through the carbon iodine adsorber in the inactive subsystem. When decay heat removal is required, a room air supply is used to cool the filter.

1.3 INTERLOCKS AND SETPOINTS

1.3.1 A Group 6 isolation is caused by following:

1.3.1.1 A Group 2 isolation caused by:

a. High drywell pressure of ≤ 1.84 psig.

b. Low reactor water level of ≥ 3".

1.3.1.2 Reactor Building Exhaust Plenum Trip Channel A or C <u>and</u> Channel B or D high-high at ≤ 49 mrem/hr (actual setpoint 10 mr/hr).

Q#
RO31

Question Description
ILT

Rev #	Rev Date	Topic Area
1	02/24/01	High Off-Site Release

Diff

Q type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C		1		

Objective #	Reference	K/A #	10CFR 55 41/43/45
	4.7.4	295038 EK2.06	41.4, 41.5, 41.11

K/A Text:

EK2.06 – Knowledge of the interrelations between HIGH OFF-SITE RELEASE RATE and the following: Process liquid radiation monitoring system.

Question:

During an outage it is necessary to operate the "B" Residual Heat Removal Heat Exchanger (RHR HX) without Service Water Booster Pumps (SWBP) running. While in this lineup which one of the following would alert operators to a tube leak in the RHR HX?

a. RPV high water level alarm.

b. Operators observe SWBP flow rising.

c. SW Liquid Process Radiation Monitor alarm

c. Chemistry reports deteriorating RPV water chemistry.

Answer:

ANSWER: c. SW Liquid Process Radiation Monitor alarm

REFERENCE: 4.7.4

K/A System: 295038
K/A Number: EK2.06
K/A Value: 3.4
Cognitive Level: 2

Justification: In this lineup where RHR Pressure will exceed the SW Pressure and any tube leak in the HX would allow a radioactive discharge to the river. This discharge would be detected by the SW Liquid Process Radiation Monitor.

Distracter a: Leakage would be from the RPV into the SW , RPV level would not rise.

Distracter b: SWBP flow is measured by a flow element on the inlet to the RHR HX, any leakage occurring in the HX would not be detected by the flow element.

Distracter d: Leakage would be from the RPV into the SW , RPV chemistry would not change.

SOURCE: NEW

18.4 Verify or place following AUTO STREAM SELECTOR Switches in OFF:

[ ] 18.4.1 Switch 1 for SW-AO-850, REC HX A SW OUT VLV.

[ ] 18.4.2 Switch 2 for SW-AO-851, REC HX B SW OUT VLV.

[ ] 18.4.3 Switch 3 for SW-AO-852, RHR HX A SW OUT VLV.

18.5 Ensure SW-AO-853 is open and place AUTO STREAM SELECTOR power supply switch in HOLD.

| 19. RECORDS

| 19.1 No quality records are generated by this procedure.

1. DISCUSSION

1.1 FUNCTION

1.1.1 The liquid process radiation monitoring system consists of three channels which monitor radiation level of service water effluent, radwaste effluent, and reactor equipment cooling water. Each channel makes a continuous permanent record of observed radiation level. If observed radiation level exceeds acceptable limits (upscale or downscale), affected monitoring channel will activate appropriate annunciator alarm. In the case of radwaste effluent, it will also trip closed the effluent valve, terminating radwaste release before limits in 10CFR20 are exceeded.

1.2 OPERATING CHARACTERISTICS

1.2.1 The radiation level of the individual liquids (reactor equipment cooling water and service water effluent) is constantly monitored by individual gamma sensitive scintillation detectors. If observed radiation level falls outside acceptable range, annunciators on Panel 9-4 will alarm and appropriate action should be taken to limit the release.

1.2.2 The detector for the service water effluent can monitor the SW discharge from four different sources. Each individual source supply can be monitored by the detector continuously or intermittently through a planned programmed cycle of 15 minutes each.

1.2.3 Radwaste effluent monitor sample is taken prior to discharge into the service water effluent piping. Activity level of effluent is recorded in Control Room on VBD-Q. The effluent monitor shall be set to alarm and automatically close waste discharge valve prior to exceeding limits in 10CFR20, Appendix B, Table II, Column 2.

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO32	ILT	1	02/24/01	Plant Fire On Site	

Q type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C		1		

Objective #	Reference	K/A #	10CFR 55 41/43/45
INT0320134B0B0200	2.3.2.37	600000 AA2.03	41.4, 41.10

K/A Text:

AA2.03 – Ability to determine and interpret the following as they apply to PLANT FIRE ON SITE: Fire alarm

Question:

Which one of the following is indicated by annunciator FP-1/D-4, RX BLDG S.W. QUAD ZONE 20 in alarm?

a. One detector has activated and the deluge system has initiated.

b. Two fire detectors have activated and the deluge system has initiated.

c. Floor drain sump area drain valves for ALL the corner rooms have isolated.

d. Floor drain sump area drain valves ~~for the~~ in the S.W. Corner room have isolated.

Only 1

Answer:

ANSWER: c ~~d.~~ ~~Floor drain sump~~ area drain valves for the S.W. ~~Corner~~ room have isolated.

All valves go close

REFERENCE: 2.3.2.37

K/A System: 600000
K/A Number: AA2.03
K/A Value: 2.8
Cognitive Level: 1

Justification: All corner room drains are isolated to prevent flooding, there are NO automatic actions associated with the deluge system.

Distracter a: There are NO automatic actions associated with the deluge system.

Distracter b: There are NO automatic actions associated with the deluge system.

Distracter c: All corner room drains isolate.

SOURCE: New

	SETPOINT	CIC
RX BLDG S.W. QUAD ZONE 20	Presence of smoke or high temperature	FP-SD-20(1) FP-TD-20(2) FP-TD-20(3) FP-TD-20(4) FP-TD-20(5)

1. AUTOMATIC ACTIONS

   1.1 RW-AOV-767, RW-AOV-768, RW-AOV-769, RW-AOV-770, RW-AOV-771, RW-AOV-772, and RW-AOV-773, REACTOR BUILDING FLOOR DRAIN SUMPS A, B, C, and D AREA DRAIN VALVES, close if their respective switches are in AUTO.

2. OPERATOR OBSERVATION AND ACTION

   2.1 Assume active fire that may affect safe shutdown equipment. Enter Procedure 5.4.1 and dispatch Fire Brigade to fire equipment locker and one Station Operator to alarming area to investigate.

   NOTE - Any one detector in alarm status prevents remaining detectors in loop from annunciating because annunciator will not reflash.

   2.2 Review Technical Requirements Manual TLCO 3.11.1 to determine if any action is required.©

   CAUTION - Reactor Building floor drain sump A, B, C, and D drain isolation valves close when any Reactor Building fire detector is activated to prevent flooding of corner rooms due excessive drainage flow. When drain isolation valves are closed, drain piping can fill up, drain to, and potentially contaminate torus area.

   2.3 If no fire detected and detector can not be reset, perform one of following:

      2.3.1 Place drain valve control switch to OPEN and monitor sump level periodically while drain valve control switch in OPEN; or,

      2.3.2 Ensure drain valve is closed and notify Radiological Protection that floor drains to that sump are diverted to torus area.

   2.4 When detector has been reset, ensure drain valve control switch is in AUTO.

(continued on next page)

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO 33	ILT	0	2/2001	SHUTDOWN COOLING	

Q Type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C				COR002-22-02, 5 COR002-23-02, 9

Objective #	Reference	K/A #	10CFR 55 41/43/45
COR002-22-02, 5 COR002-23-02, 9	2.4.2.4.1, Attachment 4	295021, AK3.05	41(b)(5) 41(b)(7)

K/A Text:

AK3.05 – Knowledge of the reasons for the following responses as they apply to LOSS OF SHUTDOWN COOLING: Establishing alternate heat removal flow paths.

Question:

The reactor has been shutdown for 18 hours and is currently in Cold Shutdown (MODE 4). A cooldown is in progress with reactor coolant temperature at 162 °F. RHR Loop "A" is in Shutdown Cooling with both reactor recirculation pumps tripped.

Subsequently, a Group 2 isolation signal trips the RHR system and RHR **CANNOT** be restarted.

Which one of the following describes where RPV water level is required to be maintained for the current conditions and why?

a. At least +48 inches on the narrow range RPV water level instruments to promote natural circulation.

b. At 0.0 inches on the wide range RPV water level instruments to support alternate heat removal using RWCU.

c. Flooded (solid) on the shutdown range RPV water level instruments to support alternate heat removal using the SRVs.

d. Between +27.5 inches and +42.5 inches on the narrow range RPV water level instruments to minimize thermal stratification in the reactor pressure vessel.

Answer:

ANSWER: a.

REFERENCE: 2.4.2.4.1, Attachment 4

Tier: 1
Group: 3
K/A System: 295021
K/A Number: AK3.05
K/A Value: 3.7
Cognitive Level: 1
Bank/Mod/New: Bank

Distracter b: Water level is not high enough to support this method of heat removal.
Distracter c: Not an approved method of heat removal.
Distracter d: Circulation is needed to minimize thermal stratification.

Proposed references to be provided to applicants during the examination: None.

1. CONTINGENCY ACTIONS

1.1 If RHR Subsystem available <u>and</u> plant conditions allow, place RHR Subsystem in SDC Mode per Procedure 2.2.69.2.

1.2 Control RPV level > 48" to aid in thermal convection flow.

[ ] **CAUTION** - Step 1.3 shall <u>not</u> be performed if blade guides are in RPV <u>or</u> if a fuel bundle is removed from around core instrumentation.©

1.3 Place or maintain one RR pump in service per Procedure 2.2.68, if available.

1.4 Place RWCU System in service per alternate heat removal section of Procedure 2.2.66.©

[ ] **NOTE** - Time to boiling and time to core uncovery graphs are based on conservative estimates; actual times are longer than indicated.

1.5 Review Attachment 5, monitor following temperatures and pressures frequently, and log every 4 hours:©

[ ] 1.5.1 If an RR pump is in-service, monitor RR-TI-151A(B).

[ ] 1.5.2 If an RR pump is <u>not</u> in service, monitor RPV metal temperatures on NBI-TR-89, REACTOR VESSEL METAL TEMPERATURE RECORDER (Panel 9-21), for indications of stratification and approach to boiling.

[ ] 1.5.3 If RWCU is in-service, monitor inlet temperature on RWCU-TI-137, TEMP IND (Panel 9-4).

[ ] 1.5.4 Monitor reactor pressure PMIS Points B025, N013, and N014 for indication of pressurization.

1.6 If RPV head is <u>on</u>, perform following:©

[ ] 1.6.1 Close reactor head vents when any of following are met:

[ ] 1.6.1.1 Average reactor coolant temperature reaches 212°F.

[ ] 1.6.1.2 RPV pressure is rising.

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO 34	ILT	0	2/2001	REFUELING	

Q Type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C				SKL010-01-02

Objective #	Reference	K/A #	10CFR 55 41/43/45
A4	10.25 section 4.1.2.5 2.4.2.4.1 Section 4.2 2.4.8.6 T.S. 3.3.1.1	295023, 2.2.27	41(b)(13)

K/A Text:

2.2.27 – Knowledge of the refueling process.

Question:

Refueling activities are in progress with a new fuel bundle being lowered into reactor core location 21-40.

Per 10.25, "Refueling - Core Unload, Reload, and Shuffle," which one of the following conditions requires the Control Room Monitor to direct fuel loading be **immediately** terminated?

a. Failure of two (2) or more APRMs within the same trip system.

b. SRM "A" and SRM "B" count rates rise by a factor of ten (10) to 300 cps.

c. Shutdown Cooling is lost with less than 24 hours estimated for "time to boil."

d. Fuel Pool Cooling is lost with less than 24 hours estimated for "time to boil."

Answer:

ANSWER: b. Note below step 4.1.2.4 states "SRM count rates normally do not exceed 100 cps."

REFERENCE: 2.4.2.4.1, Attachment 4

Tier: 1
Group: 3
K/A System: 295023
K/A Number: 2.2.27
K/A Value: 2.6
Cognitive Level: 3
Bank/Mod/New: Bank

a, c, d - None of these conditions require fuel loading be terminated per 10.25.
Distracter a: APRM are not referenced in 10.25 and are not required to be operable for fuel handling.
Distracter c: Subsequent action of 2.4.2.4.1
Distracter d: Similar to "c," but not required. 2.4.8.6 does not specifically call for terminating fuel handling, but does have refuel floor evacuation required as a subsequent action if fuel pool cooling cannot be established.

Proposed references to be provided to applicants during the examination: None.

[ ] 8.1.10 Fuel Mover check for proper grapple orientation by observing that refueling platform console is parallel with fuel assembly bail.

[ ] 8.1.11 Fuel Mover notify Refueling Floor Supervisor that proper grapple orientation is ready for verification. After the Refueling Floor Supervisor has reported, the verification the Fuel Mover may continue.

[ ] 8.1.12 Fuel Mover engage grapple hook to bail.

[ ] 8.1.13 Refueling Floor Supervisor and Refueling Bridge Spotter acknowledge that correct bundle has been grappled.

[ ] 8.1.14 Notify Control Room Refueling Monitor upon start of each fuel transfer.

[ ] 8.1.15 Fuel Mover:

- [ ] 8.1.15.1 Transfer fuel assembly to its new location per Procedure 2.2.31.
- [ ] 8.1.15.2 Check that fuel grapple is positioned over desired location.
- [ ] 8.1.15.3 Check for proper fuel assembly orientation, if specified, by observing that channel fastener is pointed in specified direction.
- [ ] 8.1.15.4 Notify Refueling Floor Supervisor that proper location and orientation are ready for verification.

[ ] 8.1.16 Refueling Floor Supervisor and Refueling Bridge Spotter acknowledge proper location and orientation.

[ ] 8.1.17 Fuel Mover, with fuel assembly aligned over vacant cavity, lower fuel assembly into position.

[ ] **NOTE** - During shuffles SRM count rates normally do not exceed 100 cps.©

[ ] 8.1.18 Control Room Refueling Monitor, monitor response of SRMs and IRMs as fuel is loaded into the core. Immediately terminate fuel loading in the event of any unexpected increase until cause of increase is evaluated.

[ ] 8.1.19 Fuel Mover disengage grapple hook from fuel assembly bail.

[ ] 8.1.20 Notify Control Room Refueling Monitor upon completion of each step.

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO35	ILT	0	02/15/01	High Secondary Cont. Area Temp.	

Q type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C		1		

Objective #	Reference	K/A #	10CFR 55 41/43/45
	COR001-11-02	295032 EA1.01	41(b).7, 45(b)6

K/A Text:

EA1.01 – Ability to operate and/or monitor the following as they apply to HIGH SECONDARY CONTAINMENT AREA TEMPERATURE: Area temperature monitoring system

Question:

The plant is operating at 100% power when the following annunciator alarms:

- 9-3-1/E-10 AREA HIGH TEMPERATURE

Which one of the following is required to check the temperature and setpoint of the alarm?

At the Control Room 9-21 panel...

a. depress the TEMP pushbutton, release and then depress the ALARM pushbutton for the channel with the illuminated LED

b. check the meter with the illuminated LED, its setpoint is indicated by a yellow arrow on the meter.

c. locate the module with the flashing LED, check the module meter, its setpoint is listed on the adjacent chart.

d. rotate the CHANNEL SELECTOR switch to the channel with the flashing LED and depress the SETPOINT button.

Answer:

ANSWER: a. depress the TEMP and ALARM pushbuttons for the channel with the illuminated LED

REFERENCE: COR001-11-02, Pages 14 and 15

K/A System: 295032
K/A Number: EA1.01
K/A Value: 3.6
Cognitive Level: 1

Justification: A common meter is read by depressing the buttons on the alarming channel

Distracter b: Buttons must be depressed and no arrows exist.

Distracter c: Buttons must be depressed and no charts exist.

Distracter d: Buttons must be depressed, there are no switches to rotate.

SOURCE: NEW

Item/Location	Switch Positions	Functions
e. Equipment sump heat exchanger valves, MO-92 & MO-93, Panel 9-4	RECIRC/AUTO/DISCH	Allows for positioning of heat exchanger valves based on sump temperature or operator control
f. Equip Sump Inbd Isol Vlv AO-94 keylock switch, Panel 9-4	NORMAL/BYPASS	To allow for PASS sampling with a Group 2 isolation.
g. Equip Sump Outbd Isol Vlv AO-95 keylock switch, Panel 9-4	NORMAL/BYPASS	To allow for PASS sampling with a Group 2 isolation.
h. Sump Recirc/Disch Byp Vlv MO-92/93, Panel 9-4	NORMAL/BYPASS	To allow for PASS sampling with a Group 2 isolation.
i. Equip Sump Isol Isol Vlv AO-1002 switch, Panel 9-4	CLOSE/OPEN	To allow for PASS sampling with a Group 2 isolation.
j. Temperature channel selection pushbutton switch 2F-DS1, Panel 9-21	DEPRESSED	When pushbutton is depressed, and the master and common service module temp. switch selected connects temperature output signal to meter.
k. Meter & Common Service Module, TEMP pushbutton	DEPRESSED	With master module in TEMP position and the pushbutton on the temp. channel switch is depressed, temp. indicator indicates temp. of monitored area.
l. Meter & Common Service Module ALARM pushbutton	DEPRESSED	With Meter and Common Service module in ALARM position and the channel pushbutton on the temp. switch is depressed, temp. indicator indicates alarm temp. setting.
m. Meter & Common Service Module ZERO pushbutton	DEPRESSED	When depressed, pushing the channel pushbutton will display the lower end of the Temperature range for the selected channel.
n. FSD	DEPRESSED	When depressed, pushing the channel pushbutton will display the high end of the Temperature range for the selected channel.

2. Local Controls

None

IV. OPERATIONAL SUMMARY

A. Operation of Drywell Sump System

As the water which has been collected in the sumps is pumped out, the discharge flow from each sump is individually metered by flow totalizers (Panel 9-19). "Grand Total" and "Batch Total" leakage rate is periodically logged from these flow totalizers and a record is maintained in order to detect increases in total leakage rate. The totalizers also have the ability to monitor flow "RATE" in gallons per minute. A 2-pen flow recorder on Panel 9-4 is provided for recording sump discharge flow rates.

Each sump has an alarm system and automatic starting sequence of pumps on rising water level. Upon a high level the preferred pump starts and upon a high-high level the second pump starts. On decreasing sump levels the pumps will automatically stop.

The Drywell equipment isolations may be opened for post accident sampling by taking the equipment sump inboard isolation AO-94 keylock switch, equipment sump outboard isolation AO-95 keylock switch, and the sump recirc bypass keylock switches to bypass. Then the equipment sump isolation AO-1002 switch is taken to isolate and the equipment sump isolation valves AO-94 and 95 may be opened.

B. Temperature, Pressure, and Humidity Detection

LO-02d

Leaks may also be detected by observing the Drywell pressure and temperature indications and recorders on Panel 9-3, 9-4, and Panel H. Any increase in the normal readings on any of the instruments requires the following operator action.

1. Compare all readings to determine if all instruments show an increase.
2. Trend Drywell pressure, temperature, and humidity on the computer and note any continuing rise.
3. Proceed to vent the Drywell if pressure is high.
4. Calculate the leak rate from the flow integrators in the Drywell sump system.
5. If the Technical Specifications limits of section 3.4.4 are reached, shut down the reactor as instructed by station technical specifications.

C. Leak Detection Systems

LO-02c

Leaks outside the Drywell can be detected by the temperature alarm system. A high area temperature alarm is indicative of a steam leak in the vicinity of either the RHR, RWCU, HPCI, RCIC systems, or the Main Steam lines.

In the event a High Area Temperature alarm is received on Panel 9-3, the following operator actions are taken.

1. On Panel 9-21, find out which RTD channel has alarmed (ALARM LED).
2. Find out what the area temperature is reading by the TEMP pushbutton and pressing the appropriate channel pushbutton.
3. If the setpoint is not known, press the TEMP pushbutton on the element that alarmed, and the alarm pushbutton on the Meter and Common Service Module read the setpoint on the meter.

Q#	*Question Description*	*Rev #*	*Rev Date*	*Topic Area*	*Diff*
RO 36	ILT	0	2/2001	SEC CONT CONTROL	

Q Type	*Response Time*	*Max Point Value*	*Passing Point Value*	*Lesson #*
M/C				COR001-11-02

Objective #	*Reference*	*K/A #*	*10CFR 55 41/43/45*
2,5	2.2.27, Section 4.2, 2.2.27, step 1.2.1.3	295036, EK2.03	41(b).4, 41(b).7, 41(b).13

K/A Text:

EK2.03 – Knowledge of the interrelations between SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL and the following: Radwaste.

Question:

Given the following for Reactor Building Equipment Drain Sump E alignment:

- SUMP PUMPS DRAIN SELECTOR switch in AUTO
- SUMP PUMP E1 and SUMP PUMP E2 switches are in AUTO
- The last Reactor Building Equipment Drain Sump to operate was pump E2.

Subsequently:

- Time = zero (0) seconds: Sump reaches HI level
- Time = thirty (30) seconds: Sump reaches HI-HI level
- Time = forty (40) seconds: Sump temperature reaches 141°F

Which one of the following describes the CURRENT status of the sump pumps E1 and E2 and the discharge to radwaste?

a. Only SUMP PUMP E1 is operating and there is flow from the sump to radwaste. Water is **NOT** recirculated through the heat exchanger.

b. Only SUMP PUMP E2 is operating and water is recirculated through the heat exchanger. There is **NO** flow from the sump to radwaste.

c. Both SUMP PUMP E1 and SUMP PUMP E2 are operating and there is flow from the sump to radwaste. Water is **NOT** recirculated through the heat exchanger

d. Both SUMP PUMP E1 and SUMP PUMP E2 are operating and water is recirculated through the heat exchanger. There is **NO** flow from the sump to radwaste.

Answer:

ANSWER: d.
When the Hi-Hi level is reached, both pumps start. When the hi temperature is reached, the discharge to radwaste closes and water is recirculated to the heat exchanger until below 140°F and a 5-minute timer times out.

REFERENCE: 2.2.27, Section 4.2, and Att. 3, step 1.2.1.3
Tier: 1
Group: 3
K/A System: 295036
K/A Number: EK2.03
K/A Value: 2.8
Cognitive Level: 3
Bank/Mod/New: New

Distracter a. Both pumps start when the hi-hi level is received. The recirculation valve opens and the discharge to Radwaste closes on high temperature.
Distracter b. Both pumps start when the hi-hi level is reached.
Distracter c. The recirculation valve opens and the discharge to Radwaste closes on high temperature.

Proposed references to be provided to applicants during the examination: None.

[ ] 4.1.1.2 At Panel 9-4, place following switches to AUTO:

a. RW-P-G1, PUMP G1.

b. RW-P-G2, PUMP G2.

[ ] 4.1.2 Place Drywell Floor Drain Sump F in operation as follows:

[ ] 4.1.2.1 At Panel 9-4, open following floor drain drywell isolation valves:

a. RW-AO-82, DISCH ROOT VLV.

b. RW-AO-83, DISCH VLV.

[ ] 4.1.2.2 At Panel 9-4, place following switches to AUTO:

a. RW-P-F1, PUMP F1.

b. RW-P-F2, PUMP F2.

4.2 REACTOR BUILDING SUMPS

[ ] 4.2.1 Place Reactor Building Equipment Drain Sump E in service as follows:

[ ] 4.2.1.1 Place SUMP PUMPS DRAIN SELECTOR switch to AUTO.

[ ] 4.2.1.2 Place SUMP PUMP E1 and SUMP PUMP E2 switches to AUTO.

[ ] **NOTE 1** - When temperature exceeds 140°F, valves will automatically align to direct flow through heat exchanger.

[ ] **NOTE 2** - When temperature goes back below 140°F and 5 minute timer has timed out, valves will automatically transfer discharge to radwaste.

[ ] **NOTE 3** - Left reset on RW-TI-534 hand is not functional, resetting is done by a 5 minute timer.

[ ] 4.2.1.3 If a high temperature alarm is received, perform following:

a. Verify sump level is above low level shutoff for pump.

b. Start either E1 or E2 SUMP PUMPS to reduce temperature.

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO 37	ILT	0	2/2001	CRD HYDRAULIC	

Q Type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C				COR002-04-02

Objective #	Reference	K/A #	10CFR 55 41/43/45
8, 10, 13	2.4.1.1.4, Section 6.1	201001, K6.05	41(b)(5) 41(b)(10)

K/A Text:

K6.05 – Knowledge of the effect that a loss or malfunction of the following will have on the CONTROL ROD DRIVE HYDRAULIC SYSTEM: A.C. Power.

Question:

A plant startup is in progress. The reactor mode switch is just placed into RUN when the following annunciator alarms:

- 9-5-2/A-6, CRD PUMP A BREAKER TRIP
- Per 2.4.1.1.4, Loss of CRD Pump, the immediate operator actions are taken and are successful

At the completion of the immediate operator actions for 2.4.1.1.4, which one of the following describes the rod motion capability using RMCS (available or not available) and how the control rod scram times are affected (will exceed or will be within technical specification limits)?

RMCS is:

a. Available. Scram times will exceed technical specification limits.

b. Available. Scram times will be within technical specification limits.

c. **NOT** available. Scram times will exceed technical specification limits.

d. **NOT** available. Scram times will be within technical specification limits.

Answer:

ANSWER: b.
The reactor is at operating pressure and scram times are okay as long as the accumulators are charged. Per 2.4.1.1.4, The standby CRD pump is started to restore CRD system pressure and maintaining rod motion capability.

REFERENCE: 2.4.2.4.1, Attachment 4

Tier: 2
Group: 1
K/A System: 201001
K/A Number: K6.05
K/A Value: 3.3
Cognitive Level: 1
Bank/Mod/New: Modified. The previous question used a reactor pressure of 850 psig and stated that both CRD pumps are lost. The modified question uses a startup condition where the applicant must assess the associated reactor pressure and also states that the immediate operator actions of the AP for a loss of CRD pump are taken. Per the AP, the standby pump is started maintaining rod motion capability. This changes the answer to "b". Because the correct answer changed, no distracters were changed since this meets the criteria for a significantly modified question.

Distracter a: Scram times are OK as long as accumulators are charged.
Distracter c: The other CRD pump is started per the immediate operator actions to maintain rod motion capability. Scram times are OK as long as accumulators are charged.
Distracter d: The other CRD pump is started per the immediate operator actions to maintain rod motion capability.

Proposed references to be provided to applicants during the examination: None.

1. SYMPTOMS

[ ] 1.1 Annunciator 9-5-2/D-6, CRD PUMP B LOW SUCTION PRESSURE, alarms.

[ ] 1.2 Annunciator 9-5-2/C-6, CRD PUMP B BREAKER TRIP, alarms.

[ ] 1.3 Annunciator 9-5-2/B-6, CRD PUMP A LOW SUCTION PRESSURE, alarms.

[ ] 1.4 Annunciator 9-5-2/A-6, CRD PUMP A BREAKER TRIP, alarms.

[ ] 1.5 Low CRD Accumulator pressure indicated by following:

[ ] 1.5.1 Annunciator 9-5-2/G-6, CRD ACCUM LOW PRESS OR HIGH LEVEL, alarms.

[ ] 1.5.2 Amber ACCUM light(s) on full core display turns on.

[ ] 1.6 Low CRD System pressure as indicated by CRD-PI-302.

[ ] 1.7 Green light for operating CRD pump turns on.

[ ] 1.8 CRD (XX;YY) high temperature prints on alarm typer.

[ ] 1.9 PMIS display (CRD:TOC) indicates rising temperatures.

2. AUTOMATIC ACTIONS

[ ] 2.1 None.

3. IMMEDIATE OPERATOR ACTIONS

[ ] 3.1 Attempt to restore CRD System pressure by performing following:

[ ] 3.1.1 If available, start standby pump.

[ ] 3.1.2 If standby pump is not available, attempt to restart tripped pump by placing switch to STOP and then to START.

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO 38	ILT	0	2/2001	RMCS	

Q Type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C				COR002-20-02

Objective #	Reference	K/A #	10CFR 55 41/43/45
4, 7	IOP 4.3, Step 4.4 note	201002, 2.1.32	41(b)(5) 41(b)(6)

K/A Text:

2.1.32 – Ability to explain and apply system limits and precautions.

Question:

During the approach to criticality, a control rod is to be single notch withdrawn from notch 08 to notch 10. After the control rod is selected, the operator places the ROD MOVEMENT CONTROL SWITCH to NOTCH OUT and holds the switch in this position.

Which one of the following describes the final position of this control rod?

a. Notch 00.

b. Notch 10.

c. Notch 12.

d. Notch 48.

Answer:

ANSWER: b. The control rod stops at position 10 but the RMCS timer does not reset until the switch is released.

REFERENCE: IOP 4.3, Step 4.4 note

Tier: 2
Group: 1
K/A System: 201002
K/A Number: 2.1.32
K/A Value: 3.4
Cognitive Level: 1
Bank/Mod/New: New

Distracter a: A Rx scram will not occur because the control rod stops at position 10.
Distracter c: This would be the correct response if the master timer also failed. The rod would be deselected after 2 seconds, 1/2 second longer than the normal timer, causing the rod to be move to position 12.
Distracter d: This would be correct if the EMERGENCY NOTCH OVERRIDE switch was also positioned to OVERRIDE when withdrawing the control rod.

Proposed references to be provided to applicants during the examination: None.

2.7 All control rod movements in MODES 3, 4, and 5 shall be made per Procedure 6.CRD.303. If performing testing under another procedure, requirements of this procedure shall be performed.

3. REQUIREMENTS

3.1 Following support systems are available:

3.1.1 CRD Hydraulic System.

3.1.2 Neutron Monitoring System.

3.1.3 Rod Worth Minimizer.

3.1.4 Rod Position Information System.

3.1.5 Reactor Protection System.

3.2 Power Supply Checklist, Attachment 1, is complete to support system operation.

4. SINGLE NOTCH CONTROL ROD WITHDRAWAL

4.1 Check ROD SELECT POWER SWITCH to ON. If REACTOR MODE switch in REFUEL, ROD SELECT POWER SWITCH must be placed to OFF and then to ON to select a different rod.

4.2 Check white ROD OUT PERMIT light on.

4.3 At ROD SELECT MATRIX, select rod to be withdrawn by pressing applicable button and ensuring following:

4.3.1 Only button on ROD SELECT MATRIX that backlights brightly is selected rod.

4.3.2 Only select light on FULL CORE DISPLAY that backlights is selected rod.

[ ] **NOTE** - Rod movement timer will complete cycle but will not reset until ROD MOVEMENT CONTROL SWITCH is released.

4.4 While monitoring reactor power, momentarily place ROD MOVEMENT CONTROL SWITCH to NOTCH OUT and ensure rod stops at next even notch position before ROD SETTLE light turns off.

4.5 If control rod is withdrawn to notch Position 48, perform coupling test per Section 9.

5. CONTINUOUS CONTROL ROD WITHDRAWAL

5.1 Check ROD SELECT POWER SWITCH to ON. If REACTOR MODE switch in REFUEL, ROD SELECT POWER SWITCH must be placed to OFF and then to ON to select a different rod.

5.2 Check white ROD OUT PERMIT light on.

5.3 At ROD SELECT MATRIX, select rod to be withdrawn by pressing applicable button and ensuring following:

5.3.1 Only button on ROD SELECT MATRIX that backlights brightly is selected rod.

5.3.2 Only select light on FULL CORE DISPLAY that backlights is selected rod.

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO 39	ILT	0	2/2001	RECIRC FLOW CONTROL	

Q Type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C				COR002-22-02

Objective #	Reference	K/A #	10CFR 55 41/43/45
5, 6, 10	2.2.68, COR002-22-02	202002, A1.07	41(b)(6) 41(b)(7)

K/A Text:

A1.07 – Ability to predict and/or monitor changes in parameters associated with operating the RECIRCULATION FLOW CONTROL SYSTEM controls including: Recirculation loop flow.

Question:

During a plant startup, the "A" recirculation pump MG set field breaker trips causing the following conditions:

- Reactor power is 39%
- "B" recirculation pump is operating
- Both recirculation MG sets M/A transfer stations are in MANUAL set at 57 % demand

What is the resulting speed demand signal to the "A" scoop tube positioner? (**Assume** the operator actions for the tripped recirculation pump are complete.)

a. 0 %

b. 22%

c. 45%

d. 57%

Answer:

ANSWER: b.
Pump speed is limited by the dual limiter to 22% speed because the discharge valve is closed on the tripped ("A") pump.

REFERENCE: 2.2.68

Tier: 2
Group: 1
K/A System: 202002
K/A Number: A1.07
K/A Value: 3.1
Cognitive Level: 2
Bank/Mod/New: Bank

Distracter a: Speed is limited by the dual limiter to 22% speed since the discharge valve is closed on the tripped ("A") pump.
Distracter c: Speed is limited by the dual limiter to 22% speed since the discharge valve is closed on the tripped ("A") pump.
Distracter d: Speed is limited by the dual limiter to 22% speed since the discharge valve is closed on the tripped ("A") pump.

Proposed references to be provided to applicants during the examination: None.

Fig 10
SO-02g

4. The generator provides the variable voltage and frequency used to operate and control the speed of the Recirc pump.

The generator has the following characteristics:

a. Power rating: 5100 kw at 56 Hz.

b. Rated voltage: 3920 VAC at 56 Hz, and 4025 VAC at 57.5 Hz.

c. Maximum current: 832 amps.

d. Maximum winding temperature: 248°F.

The Recirc MG set generator is "hard wired" to the Recirc pump motor such that, whenever the generator is operating, so is the Recirc pump.

LO-05g,08f

By varying the output frequency of the MG generator, the speed of the Recirc pump will change and therefore the loop flow rate can be varied. speed between 230 and 1150 rpm (maximum speed will change when scoop tube stops are reset for operation in ICF region). Increasing output frequency will cause current, voltage, power and temperature of the generator to increase.

Fig 10

SO-02h

5. The exciter provides the DC power necessary to produce the field within the MG
set generator.

The exciter is a small AC generator whose output is converted to DC by rotor mounted diodes. The output is then applied to the generator's field windings via the field breaker and slip rings.

During MG set operation, a portion of the generator output is routed back via the voltage regulator to supply the field of the exciter. This arranges the MG set as a "self excited" machine.

During MG set startups the excitation for the generator field is supplied from the 120 VAC Vital Instrument system.

a. MG set 1A receives excitation from Panel CCP 1A.

b. MG set 1B receives excitation from Panel CPP 2.

The transfer to self excitation is time delayed following closure of the field breaker.

Fig 23
Table 2

The field breaker for the MG set does not have a control switch in the Control Room, though position indication is provided. The field breaker automatically operates from the position of the drive motor breaker. Unless a trip condition is present, closing either drive motor breaker will close the field breaker after a time delay. If the field breaker trips on any fault, except the Anticipated Transient Without Scram (ATWS) trips, the drive motor breaker will also trip. The conditions that will cause the field breaker to trip are listed on Table 2.

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO40	ILT		02/15/01	RHR/LPCI Injection Mode	

Q type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C		1		

Objective #	Reference	K/A #	10CFR 55 41/43/45
VOR002-23-02-3.a	2.2.69.1 and COR002-23-02	203000 K1.17	41(b).7, 41(b).8

K/A Text:

K1.17 – Knowledge of the physical connections and/or cause effect relationships between RHR/LPCI INJECTION MODE (PLANT SPECIFIC) Reactor pressure

Question:

The plant was operating at 100% power with the "A" Residual Heat Removal (RHR) system aligned for suppression pool cooling. A LOCA occurs. The following conditions exist:

- Drywell Pressure 2.5 psig
- RPV Water Level -45 inches (Wide Range)
- RPV Pressure 600 psig

Which one of the following is the CURRENT status of the RHR valves listed below?

	Inboard Injection Valve RHR-MO-25A	Outboard Injection Valve RHR-MO-27A	Minimum Flow Valve RHR-MO-16A	Supp. Pool Cooling Valve RHR-MO-39A
a.	Open	Closed	Closed	Open
b.	Closed	Open	Open	Closed
c.	Open	Open	Closed	Closed
d.	Closed	Closed	Open	Open

Answer:

ANSWER: b.

REFERENCE: 2.2.69.1 and COR002-23-02
K/A System: 203000
K/A Number: K1.17
K/A Value: 4.0
Cognitive Level: 2

Justification:

Distracter a: 27A is a normally open valve, 39A closed on LOCA causing 16A to open on low flow

Distracter c: 25A will NOT open until 436 psig so 16A is open for min flow

Distracter d: 27A is a normally open valve, 39A went closed on the LOCA

SOURCE: Cooper Exam Bank

d. Recirculation pumps trip on low-low water level at ≥ -42".

e. All valves not needed for LPCI injection automatically isolate and interlock shut as previously described.

f. Minimum flow valves automatically shut as the injection valves open and flow rises to greater than 2107 gpm for 3.5 seconds.

g. An ADS permissive is established when RHR pump discharge pressure rises to $108 \leq P \leq 160$ psig.

LO-04n

h. Reactor pressure lowers due to the break and/or ADS.

1) As reactor pressure lowers to < 436 psig, the LPCI injection valves open. The MO-27 valves are interlocked full open for 5 minutes. They can be throttled after 5 minutes.

LO-15f

2) As pressure lowers below 210 psig ($199 \leq P \leq 221$ psig) both Recirculation pump discharge valves shut and are interlocked shut.

LO-06d

3) As pressure lowers below pump discharge pressure, the RHR system injects into both Recirculation system loops. The relatively cooler Suppression Pool water will enter the reactor vessel via the jet pumps.

If the water level in the vessel is high enough so stratification does not occur, the bulk water temperature should lower with a corresponding reduction in reactor pressure.

i. Even with the most limiting component failure (LPCI injection valve failed shut), reflood time on a design basis LOCA is sufficient to keep clad temperature below 2200°F when combined with Core Spray pumps.

j. After level is regained, only one pump is required to maintain level. The other loop can be used for containment cooling.

LO-04b
LO-15e, 17a

2. Suppression Pool Cooling (during LPCI initiation)

Suppression pool cooling during LPCI injection is the same as normal torus cooling except that the operator must first take spray valve control. This is done by positioning the Spray Valve Control switch to MANUAL. (The second high drywell pressure signal is not required for torus cooling valves.) If reactor water level is not yet above 2/3 core height, and it is necessary to divert LPCI flow to the torus, then the Manual Override switch may be taken to OVERRIDE.

The system is designed to provide a minimum of 11,550 gpm during two pump operation and 7700 gpm during one pump operation (DC 89-252A/A1).

LO-04h

3. Containment Sprays

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO41	ILT	1	02/24/01	HPCI	

Q type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C		1		

Objective #	Reference	K/A #	10CFR 55 41/43/45
	OP 2.2.33	206000 A2.09	41(b).7, 41(b).8

K/A Text:

A2.09 – Ability to (a) predict the impacts of the following on HIGH PRESSURE COOLANT INJECTION SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Low condensate storage tank level.

Question:

Following a small break LOCA the following conditions exist:

- Drywell pressure is 4.1 psig
- HPCI is injecting into the RPV with its flow controller in AUTO
- HPCI-MO-17, ECST Pump Suction Valve is OPEN
- Both Emergency Condensate Storage Tank (ECST) levels are at 2.0 feet and lowering
- Containment level is 13.0 feet

Which one of the following automatic actions will occur as ECST level continues to lower?

a. HPCI will lose suction pressure and trip.

b. HPCI injection valve, HPCI-MO-19, closes.

c. HPCI speed will rise attempting to maintain flow.

d. HPCI suction will transfer to the Suppression Pool.

Answer:

ANSWER: d. HPCI suction will transfer to the Suppression Pool.

REFERENCE: OP-2.2.33
K/A System: 206000
K/A Number: A2.09
K/A Value: 3.5
Cognitive Level: 2

Justification: When ECST level lowers below 23 inches HPCI suction will automatically transfer to the suppression pool.

Distracter a: HPCI suction will automatically transfer, suction will not be lost.

Distracter b: There are no auto closures of HPCI-19 for loss of suction. HPCI suction will automatically transfer.

Distracter c: HPCI suction will automatically transfer speed will not be affected.

SOURCE: New

2.1.1.3 HPCI-MO-14, STM TO TURB VLV

a. Opens on either high drywell pressure or low reactor water level.

b. Closes on a high reactor water level ≤ 54.0".

2.1.1.4 HPCI-MO-17, ECST PUMP SUCT VLV

a. Opens on either high drywell pressure or low reactor water level, if HPCI-MO-58, TORUS PUMP SUCT VLV, is not fully open.

b. Closes when HPCI-MO-58 is fully open (overrides the open signal from either high drywell pressure or low reactor water level).

2.1.1.5 HPCI-MO-58, TORUS PUMP SUCT VLV

a. Opens on emergency condensate storage tank low water level at ≥ 23" (from tank bottom) remaining or Suppression Pool high level at ≤ 4", provided there is no low steam supply pressure (≥ 107 psig) or HPCI System isolation signal present.

b. Closes on low steam supply pressure (≥ 107 psig) or HPCI System isolation signal if HPCI-MO-17 is full open and cannot be reopened until the isolation signal or low steam supply pressure signal is cleared.

2.1.1.6 HPCI-MO-20, PUMP DISCHARGE VLV

a. Opens on either high drywell pressure or low reactor water level.

b. No auto close interlocks.

2.1.1.7 HPCI-MO-19, INJECTION VLV

a. Opens on either high drywell pressure or low reactor water level.

b. No auto close interlocks.

Q#
RO42

Question Description
ILT

Rev #	Rev Date	Topic Area
	01/03/01	LPCSI

Diff

Q type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C		1		COR002-06-02

Objective #	Reference	K/A #	10CFR 55 41/43/45
COR002-06-02-9.d	2.3_9-3-3, COR002-06-02	209001 A4.11	41(b).7, 41(b).8

K/A Text:

A4.11 – Ability to manually operate and/or monitor in the control room: System flow.

Question:

During operation at full power the following annunciator is received:

9-3-3/A-5, CORE SPRAY B BREAK DETECTION

NO other annunciators alarm. A station operator sent to the d/p indicating switch reports that the d/p is +4.0 psid.

Which one of the following states the significance of this alarm and d/p indication on core spray flow during a subsequent Core Spray initiation?

Core spray flow will…

a flood the Drywell through the broken pipe.

b flow inside the core shroud and out the broken pipe.

c. flood the secondary containment because of a broken pipe.

d enter the annulus region of the reactor through the broken pipe.

Answer:

ANSWER: d.

REFERENCE: 2.3_9-3-3, COR002-06-02
K/A System: 209001
K/A Number: A4.11
K/A Value: 3.7
Cognitive Level: 2

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

Distracter a: The indicated d/p would be pegged high (+1000 psig).

Distracter b: The indicated d/p would be low –3.5 psig.

Distracter c: The instrument measures d/p downstream of the check valve inside the primary containment.

SOURCE: Cooper Exam Bank Modified

LO-03e,g

3. The high pressure side of the dPIS is connected to the Standby Liquid Control "outer" pipe, which detects the pressure in the bypass region above the Core Plate.

4. Theory of Operation

a. Cold Shutdown - The dPIS is calibrated to read zero at Cold Shutdown conditions, dP = 0.

LO-06b

b. Normal Operation - As the reactor is heated up reactor pressure increases. The reactor pressure is seen on both sides of the dPIS and therefore does not cause indicated dP to change. However, the density of the water in the core bypass region above the core plate decreases, while the density of the water in the low pressure side of the dPIS remains relatively constant as it is at approximately drywell temperature. When HIGH is less than LOW, dP is negative. At normal operating conditions, indicated dP is ≈ -3.5 psid.

c. Break Conditions - Assuming there is a break in the Core Spray line downstream of the check valve, there are three possible locations for the break: inside the core shroud; outside the shroud but inside the reactor; in the drywell. In all three cases the HIGH pressure side of the dPIS "sees" the pressure in the core bypass region above the core plate, but the LOW pressure side "sees" three different pressures.

LO-09c

1) Break inside the shroud - The pressure on the LOW side is reactor pressure inside the shroud plus the height of the water leg in the instrument line. As discussed earlier, the higher density water on the LOW side causes the dPIS to indicate a dP of about -3.5 psid. Notice that if a Core Spray Sparger is unable to perform its design spray function due to a fractured sparger <u>inside the Core Shroud</u>, the Core Spray Line Break Detection system will <u>not</u> cause an alarm. The Core Spray system could perform a flooding function, but the spray from <u>that system</u> may not provide full Core Spray coverage. However, the redundant system will provide 100% Core Spray coverage.

LO-09c

2) Break outside the core shroud, but inside the reactor vessel - The pressure on the LOW side now is the pressure outside the Core Shroud. There is about 7.5 psi drop across the steam

separators and dryers. If we assume normal operating sensed dP is -3.5 psid (HIGH - LOW = -3.5 psid) and the LOW side pressure is decreased by 7.5 psi, the equation becomes HIGH - (LOW - 7.5 psid), or HIGH - LOW + 7.5 psid. Substituting, we obtain -3.5 psid + 7.5 psid = +4.0 psid. As the alarm setpoint is +0.5 psid, a Core Spray line break outside the shroud inside the reactor will cause the Control Room alarm.

LO-09c

3) Break outside the reactor vessel, inside the drywell - The pressure on the LOW side is now drywell pressure (about 0.25 psig). Any reactor pressurization will cause (HIGH-LOW) to exceed +0.5 psid. Therefore, a Core Spray line break in the drywell will also cause the Control Room alarm, along with indications of a break in the drywell.

LO-03e
LO-01j

M. Core Spray Spargers (Figure 4)

1. Inside the Core Shroud above the Top Guide there are four 180°, Stainless Steel Core Spray spargers, mounted at two levels.

2. The two spargers at the lower elevation are supplied from Core Spray system A and the two upper spargers from Core Spray system B.

3. The Core Spray spargers are held on the Core Shroud by a bracket assembly. The sparger support brackets are welded to the Core Shroud, and the spargers are loose fitted to the brackets. This allows for differential expansion of the Core Shroud and the Core Spray spargers.

SO-02i

4. Each Core Spray system's spargers have the nozzles set up to provide 100% Core Spray coverage. The nozzles were tack-welded and aimed during construction.

LO-01k, 03c
LO-05b, 07f
SO-02j

N. Core Standby Cooling Systems Pressure Maintenance System

To ensure the discharge lines of the Core Spray system are filled, a pressure maintenance system is provided to fill the lines with water. This prevents a water hammer condition from causing piping/valve/hangar damage on system operation.

III. INSTRUMENTATION AND CONTROLS

A. Instrumentation

1. Control Room Instrumentation

Instrument/Location		Sensing Point/Type	Description
a.	Core Spray Pump 1A/B discharge pressure indicator, PI-48 A/B, Range 0-500 psig, Panel 9-3	PT-38A/B	Core Spray pump discharge

Q#
RO43

Question Description
ILT

Rev #	Rev Date	Topic Area
	02/15/01	Standby Liquid Control

Diff

Q type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C		1		

Objective #	Reference	K/A #	10CFR 55 41/43/45
	Figure 7 of COR002-29-02	211000 K3.02	41(b).2, 41(b).3, 41(b).7

K/A Text:
K3.02 – Knowledge of the effect that a loss or malfunction of the STANDBY LIQUID CONTROL SYSTEM will have on the following: Core plate differential pressure indication (Plant Specific)

Question:

During operation at full power the Standby Liquid Control (SLC) system piping entering the reactor vessel breaks upstream of the core plate d/p instrument tap (see drawing provided). Which one of the following describes the effect of this break on CORE PLATE D/P indication?

Core Plate d/p indication will indicate...

a **lower** because the low pressure line has broken.

b **higher** because the low pressure line has broken.

c. **lower** because the high pressure line has broken.

d **higher** because the high pressure line has broken.

Answer:

ANSWER: c. lower because the high pressure line has broken.

REFERENCE: Figure 7 of COR002-29-02
K/A System: 211000
K/A Number: K3.02
K/A Value: 2.6
Cognitive Level: 2

Justification: The d/p would indicate lower since the high pressure (under the core plate) side has broken.

Distracter a: The high pressure side (under the core plate) has broken

Distracter b: The high pressure side (under the core plate) has broken

Distracter d: The d/p would indicate lower since the high pressure side has broken.

SOURCE: NEW

Proposed references to be provided to applicants during the examination: COR002-29-02 Figure 7 marked for br3ak location.

Fig 12
LO-01c

E. Flow Instruments

1. The lowest instrument penetration on the reactor vessel used by the Nuclear Boiler Instrumentation system is the STANDBY LIQUID CONTROL (**SLC**) injection line, which is located about 80 in. above vessel zero. The SLC line is a line within a line allowing one penetration to be used for two purposes. The SLC injection line and the outer line are used as the sensing point for the following:

   a. Reactor Core Differential Pressure Instrument

LO-04d

Differential pressure transmitter dPT-62 is connected across the bottom core plate in the reactor. The high pressure inlet is connected to an instrument tap on the Standby Liquid Control system inlet to the vessel, while the low pressure input is connected to an outer pipe that runs into the vessel around the Standby Liquid Control system inlet pipe. The low pressure line penetrates the bottom core plate between fuel support pieces. The high pressure line is located below the core plate.

Differential pressure transmitter dPT-62 indicates the pressure drop across the core plate (reactor core). The output of the differential pressure transmitter is coupled to the red pen of recorder dPR/FR-95 mounted on Panel 9-5. Power to drive the differential pressure transmitter and associated recorder is supplied by power panel CPP.

LO-02c

   b. Core Spray Line Break Detection

A tap off the outer line at the SLC injection line penetration supplies the Core Spray line break detection system. The Core Spray system uses the pressure input from this line to sense the dP across the core, allowing a Core Spray line break to be detected.

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO 44	ILT	0	2/2001	SLC	

Q Type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C				COR002-29-02

Objective #	Reference	K/A #	10CFR 55 41/43/45
5	2.2.74, Step 11.3.1 COR002-29-02	211000, A3.07	41(b)(6)

K/A Text:

A3.07 – Ability to monitor automatic operations of the STANDBY LIQUID CONTROL SYSTEM including: Lights and alarms: Plant-Specific.

Question:

The keylock switch for Standby Liquid Control (SLC) Pump "A" is turned to the START position.

In addition to starting the "A" SLC pump, which one of the following describes the SLC squib valve(s) and the Reactor Water Cleanup (RWCU) valve(s) that change state in response to the SLC start signal?

a. Only the "A" squib valve fires, only RWCU-MO-15 isolates.

b. Both "A" and "B" squib valves fire, only RWCU-MO-18 isolates.

c. Only the "A" squib valve fires, both RWCU-MO-15 **and** RWCU-MO-18 isolate.

d. Both "A" and "B" squib valves fire, both RWCU-MO-15 **and** RWCU-MO-18 isolate.

Answer:

ANSWER: a.

REFERENCE: 2.2.74, Step 11.3.1

Tier: 2
Group: 1
K/A System: 211000
K/A Number: A3.07
K/A Value: 3.7
Cognitive Level: 1
Bank/Mod/New: Bank

Distracter b: Only the "A" squib fires, RWCU-MO-15 closes, RWCU-MO-18 does not close.
Distracter c: Only the RWCU-MO-15 isolates.
Distracter d: Only the "A" squib fires, Only the RWCU-MO-15 isolates.

Proposed references to be provided to applicants during the examination: None.

IV. SYSTEM OPERATIONAL SUMMARY

A. Condition for System Initiation

The injection of the Standby Liquid Control system is determined by the Emergency Operating Procedures (EOPs). The EOPs require that in a condition where a reactor scram should have occurred and power is greater than 3% or cannot be determined, the following, should be attempted in an effort to reduce reactor power:

- Place Reactor Mode Switch to Shut Down
- Initiate ARI (Alternate Rod Insertion)
- Run back Reactor Recirc System to Minimum
- Trip Reactor Recirc System

If these actions do not reduce power to less than 3%, the Standby Liquid Control System is required to be injected into the vessel if either of the following conditions exist:

- Average Torus Water Temperature reaches BITT (Graph 8)
- Periodic Neutron Flux Oscillations exceed 25% peak-to-peak

Fig 8

B. Injection into Reactor

The operation of the SLC pump(s) and associated valve(s) is controlled by a keylock switch(es) in the Control Room. The switch is normally in the "STOP" position with the key removed, to ensure that injection is a deliberate act.

LO-05f,g;8f

1. Sequence of Events

Each pump switch has two positions: "START" and "STOP". Using the key, the switch is turned to the "START" position, which starts pump A (B), fires a squib valve 14A (14B), and automatically isolates the RWCU-MO-15 (RWCU-MO-18). Isolating RWCU prevents the filtration removal of the boron.

LO-08a-i

Indication of proper system operation is provided by the following indications:

a. The pump discharge pressure increases

b. The "A" ("B") pump red indicating light illuminates

c. The tank level decreases

d. The Squib valve 14A (14B) indicating light extinguishes, and loss of continuity alarm sounds

e. RWCU valve(s) MO-15 (MO-18) isolate(s)

f. Power decreases

g. If pressure increases above RPV pressure; and tank level and power drop; adequate flow is assumed. There is no actual flow indicator.

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO 45	ILT	0	2/2001	RPS	

Q Type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C				COR002-21-02

Objective #	Reference	K/A #	10CFR 55 41/43/45
5	2.3.2.28 COR002-21-02, Figure 3	212000, A1.07	41(b)(6) 41(b)(7)

K/A Text:

A1.07 – Ability to predict and/or monitor changes in parameters associated with operating the REACTOR PROTECTION SYSTEM controls including: Reactor power.

Question:

While operating at100% power a power excursion to 125% occurs and the following annunciators are received:

- 9-5-2/A-3, REACTOR SCRAM CHANNEL B
- 9-5-2/B-1, NEUTRON MONITORING TRIP

NO control rods moved. At the 9-5 vertical panel, you observe the following:

- White Scram Solenoid Group lights for RPS Trip System "A" are **ON**
- White Scram Solenoid Group lights for RPS Trip System "B" are **OFF**

NO operator actions have been taken in response to the conditions stated above.

If the 5A-K15A and the 5A-K15C relays will **NOT** change state, which one of the following operator actions will cause ALL control rods to fully insert?

a. Depressing the "A" manual scram pushbutton.

b. Placing the Reactor Mode Switch to SHUTDOWN.

c. Resetting RPS and then inserting a manual reactor scram.

d. Placing "A" or "C" RPS trip channel test switches to TRIP.

Answer:

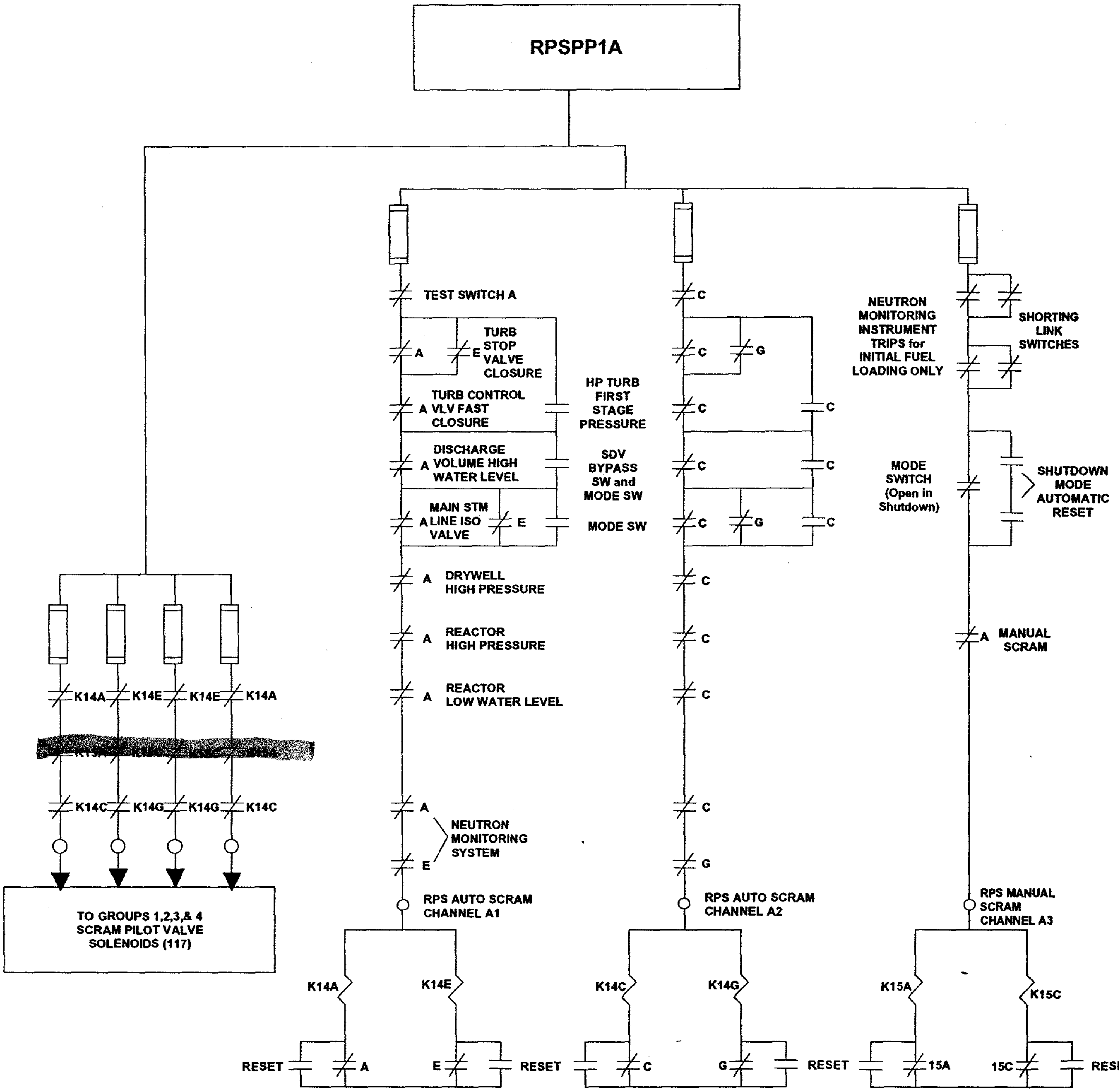
ANSWER: d.

REFERENCE: 2.3.2.28

Tier: 2
Group: 1
K/A System: 212000
K/A Number: A1.07
K/A Value: 4.2
Cognitive Level: 2
Bank/Mod/New: Bank

Distracter a: K15A and K15C must both actuate to insert all control rods.
Distracter b: K15A and K15C must both actuate to insert all control rods.
Distracter c: K15A and K15C must both actuate to insert all control rods.

Proposed references to be provided to applicants during the examination:
RPS Trip System A Figure (COR002-21-02, Figure 3)



NOTE: CONTACTS Shown with POWER>30% in the RUN Mode with no SCRAM SIGNAL present.

RPS TRIP SYSTEM A (System B Similar)

Figure 3, Rev. 10

COR002-21-02

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO46	ILT		02/16/01	IRMs	

Q type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C		1		

Objective #	Reference	K/A #	10CFR 55 41/43/45
	4.5 (also in 4.1.2 and 4.1.3)	215003 K3.05	41(b).2, 41(b).5, 41(b).7

K/A Text:

K3.05 – Knowledge of the effect that a loss or malfunction of the INTERMEDIATE RANGE MONITORING SYSTEM will have on the following: APRM (Plant Specific)

Question:

During a plant shutdown the following conditions exist:

- Reactor Mode Switch is in **RUN**
- All **APRMs are DOWNSCALE**

IRMs "E" and "H" go UPSCALE. Which one of the following will occur?

a Half scram on RPS "A".

b Half scram on RPS "B".

c. Full scram.

d Neither a half scram nor a full scram.

Answer:

ANSWER: c Full scram due to APRM downscale and companion IRM upscale in both RPS channels.

REFERENCE: 4.5
K/A System: 215003
K/A Number: K3.05
K/A Value: 3.7
Cognitive Level: 2

Justification: An APRM downscale with it's companion IRM upscale is a scram signal in RUN. IRM E is associated with APRM E in RPS Channel A, IRM H is associated with APRM B in RPS Channel B. With these two signals a full scram occurs.

Distracter a, b, d: A full scram occurs because both trip channels are activated.

SOURCE: NEW

1.2.7 DRYWELL HIGH PRESSURE

The drywell is maintained inerted with N2. An increase in drywell pressure could indicate a steam or water leak from the primary coolant system. Pressure Switches PC-PS-12A through PC-PS-12D provide inputs to the RPS when drywell pressure reaches ≤ 1.84 psig to scram the reactor. This logic is set up on a one out of two taken twice for full scram.

1.2.8 NEUTRON MONITORING SYSTEM - APRM

1.2.8.1 Three APRM channels provide outputs to RPS A and RPS B. One APRM channel in each logic (A or B) may be bypassed with an interlocked switch (Panel 9-5). Signals which generate a scram signal from the APRM are:

a. APRM high flux trip $\leq 0.58W + 61.0 - 0.58\Delta W$ (max at 119.0%).

b. With the MODE switch in any position except RUN, the APRM high flux trip is set at ≤ 14.5%.

c. Downscale ≥ 3.0%.

1. This downscale trip occurs only if you have an APRM downscale and a high high or inop trip in the associated IRM channel.

2. With the REACTOR MODE switch in any position, a high high or inop trip in an APRM channel will send a scram signal to the RPS.

d. APRM inop.

1. Less than 11 LPRM inputs.

2. Circuit boards not in circuit.

3. MODE switch not in OPERATE.

Q#	*Question Description*	*Rev #*	*Rev Date*	*Topic Area*	*Diff*
RO 47	ILT	0	2/2001	SRM	

Q Type	*Response Time*	*Max Point Value*	*Passing Point Value*	*Lesson #*
M/C				COR002-30-02

Objective #	*Reference*	*K/A #*	*10CFR 55 41/43/45*
3, 5	2.4.CRD	215004, 2.2.2	41(b)(1) 41(b)(2)

K/A Text:
2.2.2 – Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels.

Question:

During a reactor startup with the reactor close to criticality, control rod 18-19 is withdrawn from position 08 to 12. During movement of the Control Rod Drive Mechanism (CRDM), **ALL** of the SRM count rate meters remain at 4 x 10^4 cps.

Which one of the following is the cause of this indication?

a. The SRM detectors have been withdrawn too far out of the core.

b. The source neutron contribution is insignificant at this power level.

c. This control rod is uncoupled from it's control rod drive and is stuck.

d. This control rod is located too far from any SRM for this movement to be detected.

Answer:

ANSWER: c.

REFERENCE: 4.1.1, 2.4.1.1.2

Tier: 2
Group: 1
K/A System: 215004
K/A Number: 2.2.2
K/A Value: 4.0
Cognitive Level: 1
Bank/Mod/New: Bank

Distracter a: This would not prevent an indicated flux change from occurring. 4 x 10^4 cps is within the required value for detection of changes.
Distracter b: Source neutrons are the major contributor at this power level.
Distracter d: This control rod is right next to the SRM. Any rod movement near criticality would be detected by at least one SRM detector.

Proposed references to be provided to applicants during the examination: None

1. ENTRY CONDITIONS

[ ] 1.1 Changing RPIS indications when drive is not moved intentionally.

[ ] 1.2 ROD DRIFT light (red) on full core display.

[ ] 1.3 Reactor power or flux indication does not change when a control rod is moved.

[ ] 1.4 Control rod position indication does not change when drive movement is attempted.

[ ] 1.5 Control rod fails to insert when given a SCRAM signal.

[ ] 1.6 CRD high temperature alarm on PMIS.

[ ] 1.7 Abnormal insert/withdrawal drive flows/ΔPs.

[ ] 1.8 Abnormal cooling water flow/ΔPs.

[ ] 1.9 Continuous blank 4-rod display position indication on Panel 9-5.

2. AUTOMATIC ACTIONS

[ ] 2.1 None.

3. IMMEDIATE OPERATOR ACTIONS

[ ] 3.1 None.

4. SUBSEQUENT OPERATOR ACTIONS

[ ] 4.1 If more than one rod is drifting, scram and concurrently enter Procedure 2.1.5.

[ ] 4.2 Using table below, concurrently perform applicable attachment:

[ ]	Single Rod Drifting **OUT**	Attachment 1
[ ]	Rod(s) Not Full-In	Attachment 2
[ ]	Uncoupled Rod	Attachment 3
[ ]	Single Rod Drifting **IN**	Attachment 4
[ ]	Cooling Water Trouble	Attachment 5
[ ]	Stuck Rod(s)	Attachment 6
[ ]	Drive Flows Abnormal	Attachment 7

[ ] 4.3 Notify Reactor Engineering and CRD System Engineer that their support is required.

[ ] 4.4 Ensure the following Technical Specification requirements are satisfied:

[ ] 4.4.1 LCO 3.1.1, Shutdown Margin (SDM).

[ ] 4.4.2 LCO 3.1.2, Reactivity Anomalies.

[ ] 4.4.3 LCO 3.1.3, Control Rod Operability.

[ ] 4.4.4 LCO 3.1.4, Control Rod Scram Times.

[ ] 4.4.5 LCO 3.1.5, Control Rod Scram Accumulators.

[ ] 4.4.6 LCO 3.1.6, Rod Pattern Control.

[ ] 4.4.7 LCO 3.2.1, APLHGR.

[ ] 4.4.8 LCO 3.2.2, MCPR.

5. DISCUSSION

5.1 This procedure provides instructions for various abnormal CRD/HCU conditions. Each applicable Appendix (flow chart) is to be performed concurrently when directed, with the steps in the procedure body (i.e., in parallel with the steps in the procedure body). This is to ensure all necessary actions are completed.

5.2 Reactor Engineering and CRD System Engineer are expected to provide support and guidance for any or all of the following:

5.2.1 Thermal limit determinations.

5.2.2 Reactivity control limitations.

5.2.3 Shutdown Margin determination.

5.2.4 Rod pattern adjustments.

5.2.5 Troubleshooting techniques for RMCS and HCU directional control valves, including techniques already identified in Procedure 2.2.8.

5.2.6 Contacting GE per requirements of SIL-292.

5.2.7 Flushing collet rings per Procedure 2.2.8.2.

5.2.8 Continued operation and evaluation of potential slower scram time with high CRDM temperature.©

5.2.9 Continued operation without cooling water flow.

5.3 Each time a drive is inserted or withdrawn, it should be observed to latch before it is deselected. Double notching is not uncommon.

5.4 Loss of or inadequate cooling water to the CRDMs or improper operation of the CRD flow control valve can cause the inability to move rods and elevated CRDM temperatures. The CRDMs can operate without cooling water flow but seal life may be shortened by exposure to reactor operating temperatures. CRDM temperatures over 350°F may result in a measurable delay in scram response times. A rise to 400°F could result in up to a 0.150 second rise in the 90% insertion time for an otherwise normally performing CRD. The actions listed in this procedure and Procedure 2.2.8 provide guidance for manual control of flow control valves and swapping control to the standby flow control valve, which will provide the quickest means to establish correct cooling water flow rate and regain automatic control. If CRD cooling water flow cannot be established, the CRD System Engineer, Reactor Engineering, and Management need to be consulted for guidance on continued operation with elevated CRDM temperatures.

5.5 The major items of concern with an uncoupled control rod is to maintain the reactor in a safe condition and to prevent the conditions from occurring which could result in a rod drop accident.

5.6 PROBABLE CAUSE

5.6.1 Missing cooling water orifices.©

5.6.2 The CRD uncoupling rod is not properly aligned, causing the spud fingers to disengage.

5.6.3 Stuck collet.

5.6.4 CRD-FC-301 failure.

5.6.5 Foxboro Controller or E/P failure.

5.7 PROBABLE ANNUNCIATORS

5.7.1 9-5-1/C-4, ROD DRIFT.

5.7.2 9-5-2/E-6, CRD CHARGING HEADER HIGH PRESSURE.

5.7.3 9-5-1/B-4, ROD OVERTRAVEL.

6. REFERENCES

6.1 TECHNICAL SPECIFICATIONS

6.1.1 Section 3.1, Reactivity Control Systems.

6.1.2 Section 3.2, Power Distribution Limits

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO 48	ILT	0	2/2001	APRM/LPRM	

Q Type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C				COR002-22-02

Objective #	Reference	K/A #	10CFR 55 41/43/45
5, 7, 15	2.4.1.6	215005, 2.1.25	41(b).2, 41(b).3

K/A Text:

2.1.25 – Ability to obtain and interpret station reference materials such as graphs/monographs/and tables which contain performance data.

Question:

The plant is at 100% power with the "B" RRMG scoop tube locked because its controller failed upscale. Subsequently, a loss of 4160V Bus 1C occurs. Conditions after the power loss are:

- Reactor power is 45%
- Core Flow is 25 mlb/hr
- APRM indication show increasing oscillations without a corresponding increase in thermal power
- Thirty (30) seconds after the power loss, LPRM upscale alarms occur every 2 seconds

Per 2.4.1.6, ABNORMAL NEUTRON FLUX OSCILLATIONS OR OPERATIONS IN THE STABILITY EXCULSION REGION, which one of the following describes the action(s) to be taken NEXT?

a. Immediately insert a manual reactor scram.

b. Immediately raise operating Recirc pump speed at Panel 9-4.

c. Continue to monitor nuclear instruments and insert control rods.

d. Scram the reactor when LPRM upscale and downscale alarms occur.

Answer:

ANSWER: c.
The indicated power oscillations do not require a reactor scram.

REFERENCE: 2.4.1.6, 2.1.10

Tier: 2
Group: 1
K/A System: 215005
K/A Number: 2.1.25
K/A Value: 2.8
Cognitive Level: 2
Bank/Mod/New: Modified. The APRM indications were changes to indicate power oscillations are occurring. Previously the indications were not indicative of power oscillations.

Distracter b: This action is appropriate if power oscillations are not present to exit the restricted area. Power oscillations are present requiring a reactor scram.
Distracter c: This action is appropriate if power oscillations are not present to exit the restricted area is recirc pump speed cannot be raised. Recirc pump speed can be raised however power oscillations are present requiring a reactor scram.
Distracter d: This action is appropriate if power oscillations are not present as the core is monitored. Power oscillations are present requiring a reactor scram.

Proposed references to be provided to applicants during the examination: Power to Flow Map.

CNS OPERATIONS MANUAL ABNORMAL PROCEDURE 2.4.1.6 ABNORMAL NEUTRON FLUX OSCILLATIONS OR OPERATION IN THE STABILITY EXCLUSION REGION	USE: REFERENCE ⊗ EFFECTIVE: 7/19/99 APPROVAL: SORC OWNER: D. W. BREMER DEPARTMENT: OPS

1. SYMPTOMS

1.1 Unexpected LPRM upscale or downscale indications alarming and clearing:

1.1.1 Annunciator 9-5-1/B-7, LPRM UPSCALE.

1.1.2 Annunciator 9-5-1/C-7, LPRM DOWNSCALE.

1.1.3 Full core display indicators.

1.2 Increasing oscillations on APRMs without a corresponding increase in thermal power.

1.3 Unexpected SRM period alarms or positive to negative SRM period swings.

1.4 Operation within Stability Exclusion Region of Power-To-Flow Map.

2. AUTOMATIC ACTIONS

2.1 Reactor scram.

3. IMMEDIATE OPERATOR ACTION

> **CAUTION 1** - Core flow may indicate higher than actual if an RR pump is tripped and reverse core flow summer is not operating; Annunciator 9-4-3/E-3 (9-4-3/E-7), RECIRC LOOP A (B) OUT OF SERVICE, alarming indicates summer is operating.
>
> **CAUTION 2** - Operation of RRMG Set at greater than rated speed (100% or 1120 rpm) shall be avoided.©

NOTE - It may take ~ 1 minute from time pump has tripped for indicated core flow to stabilize.

3.1 If operation is in Stability Exclusion Region of Power-To-Flow Map, insert control rods and/or increase the speed of an operating recirculation pump.

4. SUBSEQUENT OPERATOR ACTION

4.1 Notify Reactor Engineering of event. Reactor Engineering should perform following:

4.1.1 Interview operating crew for indications observed and operator actions taken.

4.1.2 Obtain available event data when PMIS event is suspended.

4.1.3 Obtain available SOLOMON output.

4.2 If reactor scrams automatically, enter Procedure 2.1.5.

5. PROBABLE CAUSES

5.1 Reactor recirculation pump(s) trip.

5.2 Reactor Recirculation Flow Control System Failure.

5.3 Reactor recirculation pump(s) run back.

5.4 Operation in or near Stability Exclusion Region.

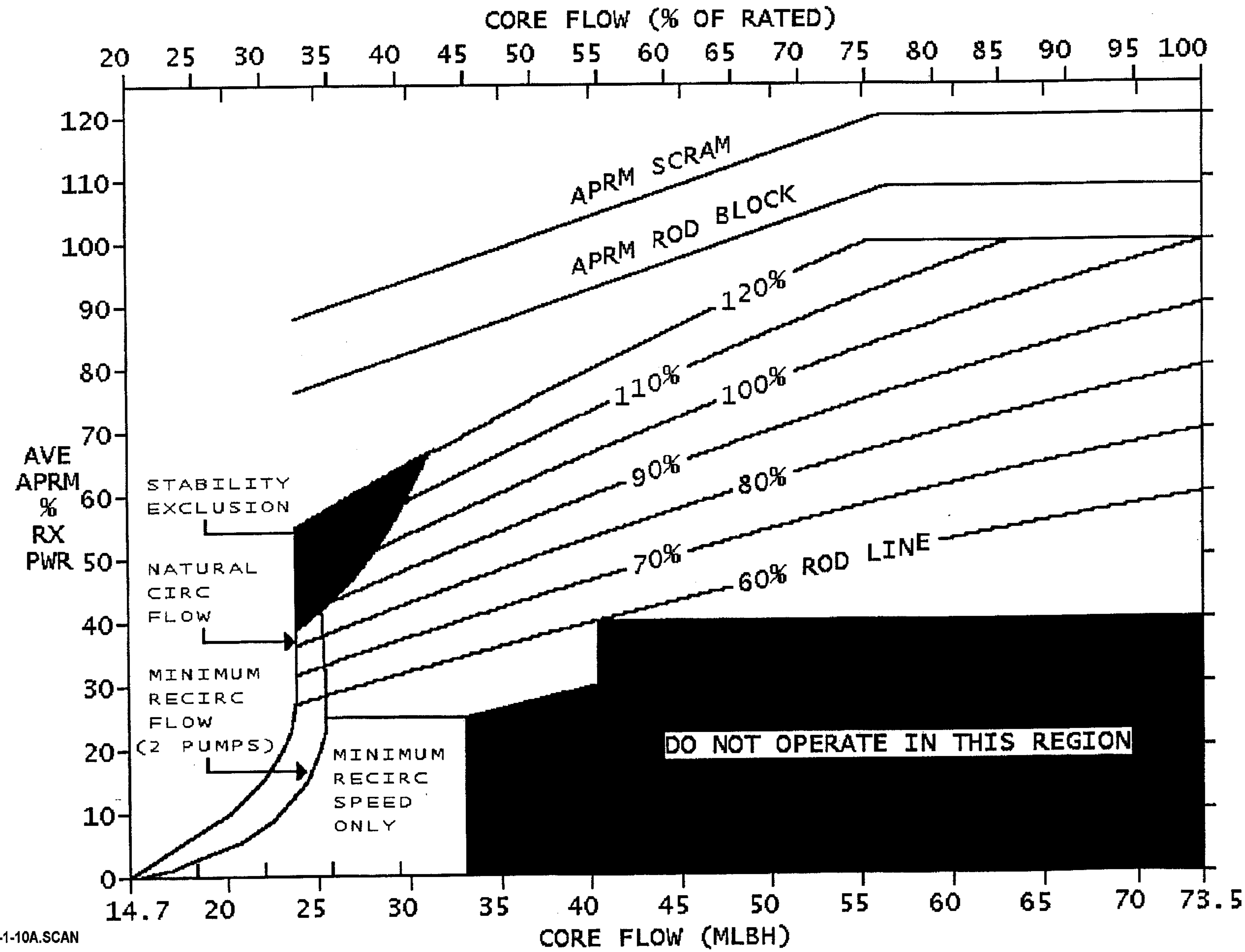
5.5 Operation at high power and low flow with strongly peaked power shape.

6. DISCUSSION

6.1 BWR cores typically operate with neutron flux noise levels of 1% to 12% of rated power (peak-to-peak) due to random boiling and flow noise. Reactor operation at high power, low flow conditions enhances the possibility of abnormal neutron flux oscillations, which can reach as high as 120%. CNS is required to be able to detect and suppress any oscillations that occur (in accordance with Option I-D of NEDO-31960-A, BWROG Long-Term Stability Solutions Licensing Methodology). This is done by analytically determining the power and flow conditions where oscillations may occur and limiting operation there (Stability Exclusion Region on the Power-To-Flow Map), monitoring near this region for potential instabilities which may cause oscillations (SOLOMON stability monitor), and automatic suppression of any oscillations by the APRM flow-biased scram.

6.2 Following a core flow transient which has not required scramming the reactor, it may take ~ 1 minute for indicated total core flow to stabilize. Therefore, in the absence of an instability occurring, it takes ~ 1 minute to determine the operating point on the Power-To-Flow Map.

POWER TO FLOW MAP



2-1-10A.SCAN

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO49	ILT		02/16/01	APRM/LPRM	

Q type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C		1		

Objective #	Reference	K/A #	10CFR 55 41/43/45
	4.1.3	215005 A2.02	41(b).2, 41(b).5, 41(b).7

K/A Text:
A2.02 – Ability to predict the impacts of the following on the AVERAGE POWER RANGE MONITOR/ LOCAL POWER RANGE MONITOR SYSTEM; and based on those predictions, use procedures to correct control or mitigate the consequences of those abnormal conditions or operations: Upscale or downscale trips

Question:

With the plant at 100% power, APRM "C" is observed at 115% power and constant at this level. All other APRMs are indicating 100%.

Which one of the following describes the automatic plant response and required action to correct this situation?

a. Only a rod block. Bypass APRM "C" at Panel 9-5.

b. Only a rod block. Bypass APRM "C" at the APRM drawer.

c. A rod block and 1/2 scram. Bypass APRM "C" at Panel 9-5 then reset the 1/2 scram.

d. A rod block and 1/2 scram. Bypass APRM "C" at the APRM drawer then reset the 1/2 scram.

Answer:

ANSWER: a. Rod block occurs at 107.5%. 1/2 scram occurs at 117.5%. Per 2.3.2.27 (9-5-1, A-7 step 2.4) bypass the affected channel.

REFERENCE: 4.5
K/A System: 215005
K/A Number: A2.02
K/A Value: 3.6
Cognitive Level: 2

Distracter b: APRM cannot be bypassed at the drawer. APRM is bypassed at Panel 9-5.
Distracter c: A 1/2 scram is not received.
Distracter d: APRM cannot be bypassed at the drawer. APRM is bypassed at Panel 9-5. A 1/2 scram is not received.

SOURCE: NEW

2.1.3 When not in the RUN Mode, the downscale rod blocks ($\geq$ 3.0%) are bypassed.

2.2 SETPOINTS - LPRM

Trip Function	Setpoint	Action
LPRM downscale	3 W/cm^2	Light and annunciator
LPRM upscale	100 W/cm^2	Light and annunciator
LPRM bypass	N/A	Light and APRM averaging compensation

NOTE - Any one APRM can initiate a rod block and half scram. One APRM rod block and half scram input can be bypassed in each trip circuit.

2.3 SETPOINTS - APRM

Table

Trip Function	Tech Spec/TRM Limit	Action
APRM downscale	$\geq$ 3.0%	Rod block annunciator, white light IRM scram interlock
APRM upscale (High) flow bias	$\leq 0.66W + 60.0\% - 0.66\Delta W$	Rod block, annunciator, amber light
APRM upscale (High) fixed	$\leq$ 109.0% RTP	Rod block, annunciator, amber light
APRM upscale (High-High)	$\leq 0.66W + 71.5\% - 0.66\Delta W$ and 119.0% RTP	Scram, annunciator, red light
APRM upscale (High-High) fixed	$\leq$ 120% RTP	Scram, annunciator, red light
APRM inoperative	APRM mode switch not in OPERATE or < 11 LPRM inputs or module unplugged	Scram, annunciator, red light

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO 50	ILT	0	2/2001	RPV INSTRUMENTATION	

Q Type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C				COR002-15-02

Objective #	Reference	K/A #	10CFR 55 41/43/45
2, 4, 5, 6	4.6.1, Section 4.1 COR002-15-02	216000, A1.07	41(b)(5) 41(b)(7)

K/A Text:

A1.07 – Ability to predict and/or monitor changes in parameters associated with operating the NUCLEAR BOILER controls including: Removing or returning a sensor (transmitter) to service

Question:

The plant is at 100% power with NBI-LT-52C level transmitter (Narrow Range Reactor Water level instrument) failed upscale.

Prior to removing the NBI-LT-52C level transmitter from service, the equalizing valve for NBI-LT-52A is fully opened by I&C.

Assume **NO** operator actions are taken.

Which one of the following describes the effect of these failures on plant operation?

a. The RFPs and the Main Turbine will trip.

b. Only a low reactor water level alarm is received.

c. Only a high reactor water level alarm is received.

d. Only a half scram is received on RPS trip system "A".

Answer:

ANSWER: a.

REFERENCE: 2.4.1.6, 2.1.10

Tier: 2
Group: 1
K/A System: 216000
K/A Number: A1.07
K/A Value: 3.4
Cognitive Level: 3
Bank/Mod/New: Bank

Distracter b: A full scram is received.
Distracter c: A high level trip occurs.
Distracter d: A full scram is received.

Proposed references to be provided to applicants during the examination: None

e. A sticking pointer

f. Excessive force used to operate the contacts

g. Static electricity effect on point movement

h. Hysteresis

I. Drift

6. Equalizing Valve Leaks

LO-05a

An equalizing valve leak allows the pressure in the reference and variable legs to equalize. This causes a zero difference in pressure between the two legs. The zero differential pressure initiates a high level signal.

7. Leakage From Instrument Lines

LO-05c

If the reference leg isolation valve packing glands were to leak, the reference leg would get hotter. This would set up a recirculation path from the condensing pot to the transmitter and then out of the leak, causing a decrease in reference leg density. This decrease in density would cause the indication to read higher than actual. A large leak may cause an actual decrease in reference leg level. In this case, the indicated level would also increase.

8. Time Delay Response

a. With a pressure change traveling to the dP cell at sonic velocity. The instrument response is 25 - 50 milliseconds.

b. Since the greatest expected rate of change in level is 10 in./second a combination of the above effects would result in a maximum error of 0.5 in.

9. Rapid Decreases in Pressure or Rapid Increases in Steam Flow

LO-04e,05h
SO-06d

a. As pressure decreases, the saturated water in the vessel is suddenly superheated and large quantities of voids are formed. The total mass of water is unchanged, and at first one might think the dPs are unchanged. If the variable leg tap was off the bottom drain line, this would be true.

LO-04k

b. What actually happens has the same effect as inflating a large balloon in the bottom of the vessel. The actual level in the core region increases, but the mass of water present is the same. The increased flow resistance, due to the increase in void content, causes the mass of water above the variable leg tap to increase, causing an increase in the variable leg head. This causes the indicated level to increase. However, this is normally not a significant error.

LO-05f

c. Another effect of rapidly decreasing pressure is boiling in the reference legs. If the reactor pressure is reduced to below the saturation pressure

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO51	ILT		02/16/01	Nuclear Boiler Instrumentation	

Q type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C		1		

Objective #	Reference	K/A #	10CFR 55 41/43/45
	2.4.5.4	216000 A4.03	41(b).10, 45(b).3, 45(b).4

K/A Text:

A4.03 – Ability to manually operate and/or monitor in the control room: Process Computer

Question:

During a plant startup, prior to going critical the following conditions occur:

- The time display has stopped updating on the primary systems PMIS displays.
- There are no responses from any PMIS display consoles.

Which one of the following has occurred and what actions are required?

a. The process computer has failed, contact Nuclear Information Services (NIS) and continue the startup.

b. The process computer has failed, halt the startup and place the current rod group at the same notch position.

c. The RWM/RPIS computer has failed, halt the startup and if it cannot be returned to service, insert the control rods.

d. The 3D Monicore system has failed, contact Reactor Engineering (RE) and with their permission continue the startup.

Answer:

ANSWER: b. The process computer has failed, halt the startup and place the current rod group at the same notch position.

REFERENCE: 2.4.5.4
K/A System: 216000
K/A Number: A4.03
K/A Value: 3.0
Cognitive Level: 1

Justification: Symptoms and Immediate Actions from Abnormal Procedure 2.4.5.4

Distracter a: The startup may not continue.

Distracter c: Failure of the RWM/RPIS computer is not indicated. The process computer has failed and there are no requirements to insert the control rods.

Distracter d: Failure of 3D Monicore is not indicated. The process computer has failed and there are no provisions in the procedure for RE permitting the startup to continue.

SOURCE: NEW

1. SYMPTOMS

1.1 LOSS OF PRIMARY SYSTEM (PMIS SYSTEMS HAVE FAILED)

1.1.1 Time display is not updating on all Primary Systems PMIS Display Consoles.

1.1.2 Loss of response from all PMIS Display Consoles on Primary System.

1.1.3 PMIS Display Consoles only response is PMISA: or PMISB:.

1.2 LOSS OF BACKUP SYSTEM

1.2.1 Primary status indicates that no backup is available (i.e., CONSOLE=PRIMARY, not CONSOLE=PRI/BACK).

1.2.2 Loss of Backup System alarm is generated by PMIS.

1.2.3 Time display on Backup System does not update (except in PLAYBACK MODE).

1.2.4 Loss of response from PMIS Display Consoles on Backup System.

1.2.5 PMIS Display Consoles on Backup System only response is PMISA: or PMISB:.

1.3 LOSS OF RWM/RPIS COMPUTER

1.3.1 Lack of response of PMIS Rod Position Indication System (RPIS) or Rod Worth Minimizer (RWM) panel display on a known operational PMIS Display Console on Primary System.

1.3.2 RWM Panel display MODE field displays NOT COMMUNICATING.

1.4 LOSS OF PMIS MULTIPLEXERS (DATA LINKS)

1.4.1 PMIS Points SYS019-SYS029 indicate status as DOWN.

1.4.2 IDT alarm indicating a multiplexer is OFF-LINE.

1.4.3 Loss of power from PMIS UPS affects PMIS Data Links 0, 1, 2, 3, 8, and 9.

1.4.4 Loss of power from CDP-1A affects 1E DIV 1 PMIS Data Link 6.

1.4.5 Loss of power from CPP affects 1E DIV 2 PMIS Data Link 7.

1.4.6 Loss of power from CPP-2 affects PMIS Data Link 4.

1.4.7 Loss of power from LPTG-8 affects PMIS Data Link 5.

1.5 LOSS OF 3D MONICORE COMPUTER

1.5.1 PMIS Point SYS031 indicates status DOWN.

1.5.2 3D Turn-on Code indicates 3D Monicore is UNKNOWN.

2. AUTOMATIC ACTIONS

2.1 Computer hardware failures may result in rod blocks being initiated from Rod Worth Minimizer.

2.2 Loss of primary computer when backup computer is available will result in automatic fail-over to backup computer.

3. IMMEDIATE OPERATOR ACTIONS

3.1 Halt any power changes in progress. If control rods are being moved, place rod group at same notch position.

4. SUBSEQUENT OPERATOR ACTIONS

4.1 Notify Shift Supervisor.

4.2 If PMIS fail-over occurs, perform following:

4.2.1 Attempt failed system restart using SS turn on code.

4.2.2 If restart fails, attempt to boot Backup System per Procedure 2.6.3.

4.2.3 Contact Nuclear Information Services (NIS) personnel during normal or next normal working hours and inform them of problem.

4.2.4 If PMIS Point N067, REACTOR BUILDING EFFLUENT FLOW, 3rd line alarm is needed during a refuel outage, contact Nuclear Information Services (NIS) personnel and request re-activation of the function (program RXEFFMONZ).

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO52	ILT		02/16/01	Reactor Core Isolation Cooling	

Q type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C		1		

Objective #	Reference	K/A #	10CFR 55 41/43/45
	2.3_9-4-1, COR002-18-02	217000 K2.02	41(b).7, 45(b).3, 45(b).4

K/A Text:

K2.02 – Knowledge of the electrical power supplies to the following: RCIC initiation signals (logic)

Question:

The plant is operating at 100% power when the following annunciator alarms:

- 9-4-1/A-3, RCIC LOGIC POWER FAILURE

After investigation it is determined that 125V DC Panel AA2 has been lost. Which one of the following effects does this power failure have on the Reactor Core Isolation Cooling (RCIC) system?

a. RCIC is NOT operable and CANNOT be manually started from the control room.

b. RCIC has automatically started and must be manually shutdown from the control room.

c. RCIC will NOT automatically start, but it can be manually started from the control room.

d. RCIC will start with only one half an initiation signal and can be manually started from the control room.

Answer:

ANSWER: a. RCIC is NOT operable and CANNOT be manually started from the control room.

REFERENCE: 2.3_9-4-1, COR002-18-02
K/A System: 217000
K/A Number: K2.02
K/A Value: 2.8
Cognitive Level: 2

Justification: RCIC initiation logic has lost power and it cannot be started because power was also lost to the flow controller.

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

SOURCE: NEW

k. Supervisory alarm circuitry
l. AO-12/13, Condensate Pump discharge to equipment drains
m. AO-32, Condensate drain pot trap bypass
n. AO-34/35, Steam Line drain to condenser
o. MO-14, Turbine Trip and Throttle.

2. 125V DC Panel AA2

Provides power to the RCIC GEMAC flow controller, EGM, test circuit logic, remote turbine trip, channel A isolation logic, and initiation logic. A loss of AA2 would result in the following problems:

a. No initiation since power lost to relay 13A-K2 which opens MO-131.

b. No isolation since power lost to relays K15, K16, and K17.

c. No high level trip due to loss of K38x which is in the MO-131 close circuit.

d. Loss of remote turbine trip due to loss of K8.

3. 125V DC Panel AA3

Provides 125V DC to outboard steam isolation valve (MO-16).

4. 125V DC Panel BB2

Provides power to channel B isolation logic (K30, 31, and 34) and Channel B high level trip logic relay K34.

5. 250V DC RCIC Starter Rack

Provides power to the following Loads:

a. Condensate Pump

b. Vacuum Pump

6. NBPP

For RCIC pump suction and discharge valve indicators and turbine steam inlet and exhaust pressure indicators.

7. 460V AC Critical MCC-Y

Provides power to the inboard steam isolation valve (MO-15).

8. 120V AC, CCP1A Circuit 2

Provides power to RCIC-CV-26CV indicating light

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO53	ILT		02/16/01	Reactor Core Isolation Cooling	

Q type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C		1		

Objective #	Reference	K/A #	10CFR 55 41/43/45
	2.2.67	217000 K4.02	41(b).5, 41(b).7, 45(b).4

K/A Text:

K4.02 – Knowledge of the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) design feature(s) and/or interlocks which provide for the following: Prevent overfilling reactor vessel.

Question:

Following a Group 1 isolation signal and automatic reactor scram, the Reactor Core Isolation Cooling (RCIC) system automatically started and injected as designed. When an operator verifies RCIC performance the following indications are noted:

- RCIC-MO-131, STM SUPP TO TURB VLV, is CLOSED
- RCIC turbine speed is 100 rpm and lowering
- RCIC discharge pressure is ZERO (0)
- RCIC turbine inlet pressure is 900 psig

Which one of the following is the cause of these indications?

a. Turbine overspeed

b. High RPV water level

c. Ramp generator failed low

d. RCIC high exhaust pressure

Answer:

ANSWER: b. High RPV water level

REFERENCE: 2.2.67
K/A System: 217000
K/A Number: K4.02
K/A Value: 3.2
Cognitive Level: 2

Justification: High RPV level closed the steam admission valve to shutdown RCIC.

Distracter a: This would not close the steam admission valve.

Distracter c: This would not close the steam admission valve.

Distracter d: This would not close the steam admission valve.

SOURCE: Modified

Lesson Number:	COR002-18-02	Revision:	12

8) Verify MO-27, the min flow bypasss valve, closes when the RCIC pump flow increases > 80 gpm

9) Verify AO-34 and AO-35, steam line drain to condenser isolation valves

10) Verify that the condensate pump operates automatically to maintain barometric condenser level.

LO-12e

2. RCIC Turbine Trip

a. The following conditions will cause an automatic turbine trip:

1) Loss of turbine oil pressure

2) High turbine exhaust pressure (25 psig)

3) Low pump suction pressure (15" Hg vacuum)

4) Turbine overspeed (125% of rated, 5625 rpm)

5) Manual pushbutton on Panel 9-4

6) Manual (local) trip lever

7) Receipt of an auto isolation signal

LO-10g

b. On all turbine trip signals, the turbine trip-throttle valve will close. The minimum flow valve will close on trips that energize the trip solenoid.

LO-08b; 11d

c. In the event of a reactor high water level shutdown the steam supply block valve (MO-131) will close, then the turbine trip-throttle valve will close on low oil pressure. When both the trip-throttle valve and the (MO-131) valve are fully closed, the motor operated trip reset valve will automatically reset the trip-throttle valve. The RCIC system will then automatically restart on a low water level initiation signal without any operator action required.

NOTE: With the RCIC Isolation switch on Panel 9-30 in the Auxiliary Relay Room in the ISOLATE position, automatic closure of the steam supply block valve (MO-131) due to reactor high water level will be inhibited.

d. Except for the turbine overspeed and the manual trip lever turbine trip all other turbine trips are reset by the Control Room operator. The operator closes the steam supply block valve (MO-131). This energizes the trip-throttle valve reset valve (MO-14) which operates to reset the trip-throttle valve.

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO54	ILT	0	02/16/01	Automatic Depressurization System	

Q type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C		1		

Objective #	Reference	K/A #	10CFR 55 41/43/45
	2.2.1, 2,4CSCS	218000 K5.01	41(b).5, 41(b).7, 41(b).8

K/A Text:

K5.01 – Knowledge of the operational implications of the following concepts as they apply to AUTOMATIC DEPRESSURIZATION SYSTEM: ADS logic operation

Question:

The following conditions have been present two (2) minutes:

- RPV water level -114 inches
- RPV pressure 458 psig

- All low pressure ECCS pumps are operating

Which one of the following is the current status of the ADS valves and the actions necessary to close or maintain them closed?

a. OPEN – Either ADS inhibit switch must be placed in INHIBIT.

b. OPEN – Both ADS inhibit switches must be placed in INHIBIT.

c. CLOSED – Either ADS inhibit switch must be placed in INHIBIT.

d. CLOSED – Both ADS inhibit switches must be placed in INHIBIT.

Answer:

ANSWER: b. OPEN – Both ADS inhibit switches must be placed in INHIBIT.

REFERENCE: 2.2.1, 2.4CSCS
K/A System: 218000
K/A Number: K5.01
K/A Value: 3.8
Cognitive Level: 2

Justification: All the conditions are met for ADS auto initiation, -113", 109 second timer, RHR and CS pumps are running. Both switches must be placed in INHIBIT.

Distracter a: Both switches must be placed in INHIBIT.

Distracter c: Valves are open

Distracter d: Valves are open

SOURCE: NEW

1. DISCUSSION

1.1 FUNCTION

1.1.1 ADS logic and instrumentation is designed to lower the reactor pressure during postulated conditions, so that reflooding of the core can take place by the low pressure CSC Systems.

1.1.2 The LLS logic and instrumentation is designed to mitigate the effects of postulated thrust loads on the safety/relief valve (SRV) discharge lines by preventing subsequent actuations with an elevated water leg in the SRV discharge line. It also mitigates the effects of postulated pressure loads on suppression chamber structural components by preventing multiple actuations in rapid succession of the SRVs subsequent to their initial actuation.

1.1.3 The safety/relief valves and safety valves provide over pressure relief protection and over pressure safety protection by opening (self-actuated) at a predetermined pressure in the main steam line.

1.1.4 The relief valves may be manually opened by positioning switches in the Control Room when the reactor pressure is > 50 psig.

1.2 OPERATING CHARACTERISTICS

1.2.1 The ADS serves as a backup to the HPCI System under Loss Of Coolant Accident conditions. If the water level lowers to the initiation setpoint level and does not recover, a 109 second time delay relay energizes and starts timing. At the end of the time delay, if a low pressure CSCS pump is developing sufficient discharge pressure (AC interlock) to inject into the reactor vessel, relief valves MS-RV-71A, MS-RV-71B, MS-RV-71C, MS-RV-71E, MS-RV-71G, and MS-RV-71H open. This vents reactor steam to the suppression chamber; thereby lowering the reactor pressure where the CS or RHR pumps are able to inject water.

1.2.2 In the event of a Group 4 isolation where the HPCI System is not available, the RCIC System can be used to restore water level. However, the RCIC System does not have sufficient capacity to restore the water level in the time period allowed and actuation of the ADS valves will occur unless manually defeated. The manual INHIBIT switches allow the Operator to prevent ADS actuation if the RCIC System is in operation and restoring vessel water level or if directed by the EOPs.

2. INTERLOCKS AND SETPOINTS

2.1 INTERLOCKS

2.1.1 A 109 second timer is incorporated in the initiation logic to allow time for the HPCI System to restore the water level before relief valves MS-RV-71A, MS-RV-71B, MS-RV-71C, MS-RV-71E, MS-RV-71G, and MS-RV-71H are actuated.

2.1.2 The Operator may reset the ADS timer, to inhibit operation of the system, at any time by depressing <u>both</u> ADS LOGIC A TIMER 93-2E-S2A and ADS LOGIC B TIMER 93-2E-S2B pushbuttons. When the buttons are released, the 109 second timer will again start to time, if the initiation signal is still present.

2.1.3 Placing <u>both</u> ADS A INHIBIT and ADS B INHIBIT switches in the INHIB position prevents automatic actuation of the ADS by keeping the reactor low level signals from reaching the ADS logic. This may be done; for example, if the RCIC pump is in operation and supplying sufficient makeup.

2.1.4 Two different level signals are utilized in each initiation logic to prevent accidental initiation during level sensor testing.

2.1.5 A low pressure CSCS pump must be developing sufficient discharge pressure (AC interlock) before the ADS will initiate the opening of MS-RV-71A, MS-RV-71B, MS-RV-71C, MS-RV-71E, MS-RV-71G, and MS-RV-71H.

2.1.6 The logic is arranged into four channels (A, B, C, and D) and requires coincident signals from A and C or B and D in order to actuate the system.

2.2 SETPOINTS

2.2.1 Low reactor water level ≥ -113" (with ≥ 3" permissive) starts 109 second timer.

2.2.2 After 109 second timer elapsed and if a RHR or a CS pump is running and discharge pressure is between 108 and 160 psig, six relief valves (MS-RV-71A, MS-RV-71B, MS-RV-71C, MS-RV-71E, MS-RV-71G, and MS-RV-71H) will open and blow reactor pressure down using the water in the suppression pool for steam condensing.

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO 55	ILT	0	2/2001	TECH SPECS	

Q Type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C				

Objective #	Reference	K/A #	10CFR 55 41/43/45
	TECH SPEC 3.6.4.3	223001, 2.1.12	41(b)(13) 43(b)(2)

K/A Text:

2.1.12 – Ability to apply technical specifications for a system.

Question:

A core offload is in progress. The "A" train of SGT is declared inoperable at 1200 on 5/1. The "B" train is in STANDBY and is OPERABLE. At 1200 on 5/8, the "A" train is still inoperable.

Which one of the following describes if the CORE ALTERATIONS can be continued including why or why not?

a. Yes. CORE ALTERATIONS can be continued provided the OPERABLE train of SGT is placed into operation before continuing.

b. Yes. CORE ALTERATIONS can be continued because at least one train is still OPERABLE and will start automatically if required.

c. No. Both trains of SGT are required to be OPERABLE prior to and during the performance of any CORE ALTERATIONS.

d. No. The inoperable train was required to be OPERABLE in the 7 day Completion Time to continue CORE ALTERATIONS.

Answer:

ANSWER: a.
The 7-day allowed outage time for Condition A expires at 1200 on 5/8. Upon expiration Condition C is entered which allows fuel movement to continue if the OPERABLE SGT train is placed into operation.

REFERENCE: TECH SPEC 3.6.4.3

Tier: 2
Group: 1
K/A System: 223001
K/A Number: 2.1.12
K/A Value: 2.9
Cognitive Level: 2
Bank/Mod/New: New

Distracter b: The OPERABLE SGT train must be placed into operation before continuing fuel movements.
Distracter c: Fuel movement can be continued if the OPERABLE SGT train is placed into operation.
Distracter d: Fuel movement can be continued if the OPERABLE SGT train is placed into operation.

Proposed references to be provided to applicants during the examination:
TECH SPEC 3.6.4.2 and Bases, Section 1.0 (all), Section 3.0 (all)

3.6 CONTAINMENT SYSTEMS

3.6.4.3 Standby Gas Treatment (SGT) System

LCO 3.6.4.3 Two SGT subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of irradiated fuel assemblies in the secondary containment,
During CORE ALTERATIONS,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SGT subsystem inoperable.	A.1 Restore SGT subsystem to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, or 3.	B.1 Be in MODE 3.	12 hours
	AND	
	B.2 Be in MODE 4.	36 hours
C. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	----------------NOTE----------------- LCO 3.0.3 is not applicable. ---	
	C.1 Place OPERABLE SGT subsystem in operation.	Immediately
	OR	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. (continued)	C.2.1	Suspend movement of irradiated fuel assemblies in secondary containment.	Immediately
	<u>AND</u>		
	C.2.2	Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>		
	C.2.3	Initiate action to suspend OPDRVs.	Immediately
D. Two SGT subsystems inoperable in MODE 2, or 3.	D.1	Enter LCO 3.0.3	Immediately
E. Two SGT subsystems inoperable during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	E.1	---------NOTE----------- LCO 3.0.3 is not applicable. --------------------------- Suspend movement of irradiated fuel assemblies in secondary containment.	Immediately
	<u>AND</u>		
			(continued)

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO 56	ILT	0	2/2001	MAIN STEAM	

Q Type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C				COR002-14-02

Objective #	Reference	K/A #	10CFR 55 41/43/45
7f	2.4.2.3.3, Section 6.1	223002, K3.09	41(b)(5) 41(b)(7)

K/A Text:

K3.09 – Knowledge of the effect that a loss or malfunction of the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUTOFF will have on following: Main steam system

Question:

While operating at 100% power, outboard MSIV AO-86B closes.
Which one of the following describes the likely cause of the resultant reactor scram?

a. Low reactor pressure.

b. Low reactor water level.

c. High reactor pressure.

d. High reactor water level.

Answer:

ANSWER: c.
Inadvertent closing of one MSIV while at high power will initiate a small but rapid pressure increase resulting in a scram on either high neutron flux or high reactor pressure.

REFERENCE: 2.4.2.3.3, Section 6.1

Tier: 2
Group: 1
K/A System: 223002
K/A Number: K3.09
K/A Value: 3.4
Cognitive Level: 1
Bank/Mod/New: Modified. The original question was at 70% power where a reactor scram will not occur. At 100% it will. The question was changed to ask the cause of the resultant scram. Previous answer was high neutron flux. New answer is high reactor pressure which may also be the cause.

Distracter a: The pressure rise will cause a reactor scram on high reactor pressure or high neutron flux.
Distracter b: The pressure rise will cause a reactor scram on high reactor pressure or high neutron flux.
Distracter d: The pressure rise will cause a reactor scram on high reactor pressure or high neutron flux.

Proposed references to be provided to applicants during the examination: NONE

1. SYMPTOMS

1.1 Annunciator 9-5-2/A-1, RX SCRAM CHANNEL A, alarms.

1.2 Annunciator 9-5-2/A-3, RX SCRAM CHANNEL B, alarms.

1.3 Annunciator 9-5-2/B-2, MSIV NOT FULL OPEN TRIP, alarms.

1.4 Annunciator 9-3-1/C-2, DRYWELL PNEUMATIC HDR PRESSURE LOW, alarms.

1.5 MSIV indicates closed on Panel 9-3.

1.6 Reactor pressure rises, depending upon number of MSIVs that close.

1.7 Higher than normal steam flow in other main steam lines.

1.8 Power level rises due to higher pressure.

2. AUTOMATIC ACTIONS

NOTE - 100% steam flow with 3 steam lines gives ~ 133% through each line due to power rise as a result of higher reactor pressure.

2.1 If reactor is operating at > 75% power, following may occur:

2.1.1 Remaining MSIVs may close on high steam flow causing reactor scram.

2.1.2 High reactor pressure scram at ≤ 1050 psig.

2.1.3 High flux scram due to pressure transient.

2.1.4 Low-Low Set will actuate upon coincident signals of reactor high pressure scram and any relief valve open.

3. IMMEDIATE OPERATOR ACTIONS

3.1 If a SCRAM setpoint is exceeded, ensure reactor has scrammed.

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO57	ILT	0	02/16/01	Safety Relief Valves	

Q type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C		1		

Objective #	Reference	K/A #	10CFR 55 41/43/45
	2.2.1 and COR002-16-02	239000 K1.05	41(b).3, 41(b).7

K/A Text:

K5.01 – Knowledge of the physical connections and/or cause-effect relationships between SAFETY RELIEF VALVES and the following: Plant air systems (Plant Specific)

Question:

The reactor has scrammed following a loss of DEH fluid pressure. Safety Relief Valve "D" (RV-71D) is maintaining RPV pressure between 875 and 1015 psig. The following annunciators have alarmed:

- 9-3-1/C-2, DRYWELL PNEUMATIC HDR LOW PRESSURE
- 9-3-1/D-2, RELIEF VALVE ACCUMULATOR LOW PRESSURE

Which one of the following actions is necessary to restore and/or maintain the Low Low Set (LLS) function?

a. Press the LLS Logic Reset Pushbuttons on Panel 9-3.

b. Open IA-SOV-21, Instrument Air Backup to the Nitrogen system.

c. Verify the control switches for RV-71D and RV-71F are in AUTO.

d. Cycle the control switches for RV-71D and RV-71F to OPEN and back TO AUTO.

Answer:

ANSWER: b. Open IA-SOV-21, Instrument Air Backup to the Nitrogen system.

REFERENCE: 2.2.1 and COR002-16-02
K/A System: 239000
K/A Number: K1.05
K/A Value: 3,1
Cognitive Level: 2

Justification: Loss of pneumatic pressure will prevent LLS operation to restore LLS the Instrument Air Supply to the Drywell Nitrogen system must be opened

Distracter a: This will not restore LLS if pneumatic pressure is lost

Distracter c: The control switches had to be in AUTO for LLS to function initially. Switch position has no affect on pneumatic supply.

Distracter d: This will deplete the pneumatic supply further, not restore it.

SOURCE: NEW

DRYWELL PNEUMATIC HDR LOW PRESSURE	SETPOINT (1029) 90 psig	CIC IA-PS-1

1. AUTOMATIC ACTIONS

    1.1 MSIVs will start to drift closed at ~ 80 psig.

2. OPERATOR OBSERVATION AND ACTION

    2.1 If any MSIVs have closed due to low pneumatic pressure, place switches for any MSIVs that are closed to CLOSE.

    2.2 Open IA-SOV-SPV21, DRYWELL IA SUPPLY VLV, on Panel 9-3.

    2.3 Check nitrogen supply PI-631 (R-903-S above south HCUs).

> **<u>CAUTION</u>** - If pneumatic pressure is restored to any MSIVs that have closed and associated MSIV switch is still in AUTO OPEN, the MSIV may reopen and possibly against a high D/P.

    2.4 If nitrogen supply pressure is < 90 psig and IA-SOV-SPV21 fails to open, <u>immediately</u> open IA-571, REACTOR BUILDING DRYWELL SUPPLY (overhead between CRD HCUs and TIP Room).

    2.5 If pneumatic supply pressure drops below 73 psig, close inboard MSIVs.©

    2.6 If air is aligned to drywell pneumatic header, assume oxygen content is <u>not</u> within limit and enter Tech Spec LCO 3.6.3.1.©

3. PROBABLE CAUSES

    3.1 System leakage.

(continued on next page)

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO58	ILT	0		Reactor/Turbine Pressure Regulator	

Q type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C		1		COR002-22-02

Objective #	Reference	K/A #	10CFR 55 41/43/45
COR002-09-02-4.b	COR002-09-02	241000 A3.17	41(b)(6) 41(b)(7)

K/A Text:

A3.17 – Ability to monitor automatic operations of the REACTOR/TURBINE PRESSURE REGULATING SYSTEM including: Turbine runback.

Question:

The plant was at 100% power when the "B" Reactor Recirculation pump received a runback signal and its speed lowered as designed.

Which one of the following is a plant condition present while the "B" Reactor Recirculation pump speed was lowering?

a. Main Turbine speed lowering slowly.

b. Reactor Steam Dome pressure rising slowly.

c. Main Steam Equalizing Header pressure will be lower.

d. Flow Comparator Offnormal annunciator alarmed and then cleared.

Answer:

ANSWER: c.
Main Steam Equalizing Header pressure will be slightly lower due to the decrease in reactor power and pressure.

REFERENCE: COR002-09-02

K/A System: 241000
K/A Number: A3.17
K/A Value: 3.3
Cognitive Level: 1
Bank/Mod/New: Bank

Distracter a: Speed remains at 1800 rpm.
Distracter b: Reactor pressure remains the same.
Distracter d: Conditions for this alarm are not met.

Proposed references to be provided to the applicants during the exam: None position the turbine governor or bypass valves or to raise and lower the pressure setpoint, in the event of computer of Digital system failure, or if the controls are placed in manual by operator action.

In manual bypass valve control, if the turbine trips above 106 MWE to 116 MWE, the bypass valves will open and must be manually closed to control pressure.

c. Maintenance

In the DEH cabinets, in the computer room, the typer/printer, along with a cassette loader, is used to allow calibration inputs and to load or modify the control program or to load or retrieve data from the computer memory. There is also the DEH simulator, System Diagnostic panel and other test panels for use in measuring internal system parameters, and one which is used to insert the bias signal into the pressure controllers.

Fig 1
LO-03a,c

F. Generation of Valve Positioning Signals

Control signals which provide proper positioning of the governor and bypass valves are developed by the circuit represented by the block diagram in Figure 1.

1. The heart of the system is two pressure controllers. The pressure controllers compare several pressure signals and develop a pressure error signal which represents total steam flow demand.

2. Input signals to each pressure controller are; steam header pressure, pressure setpoint, and a bias signal applied to one of the controllers.

LO-04b, 05b

The steam header pressure signals come from pressure transducers which sense at the equalizing header (throttle header upstream of the Main Turbine stop valve header), and downstream of the MSIV's and MSL flow restrictors. (This will cause the throttle header pressure to always be less than Rx Pressure).

The pressure setpoint is an operator selected value. It represents the throttle header pressure that the DEH system will attempt to maintain. Pressure setpoint value is normally 926 psig. Pressure setpoint may be adjusted in order to improve plant efficiency.

The last input signal is a bias signal applied to one of the two pressure controllers. The effect of applying a -0.3 volt bias to one controller is to place the other controller in control and the biased controller then acts as a backup (this will be discussed in more detail shortly).

LO-04l,m,08a
LO-03a,c

As throttle header pressure increases above the pressure setpoint, the pressure error (throttle header pressure minus pressure setpoint) generates a flow demand signal calling for an opening of the governor and/or bypass valves. When throttle header pressure is at or below the pressure setpoint, the flow demand signal is zero, calling for closure of the governor and bypass valves.

3. The system is designed so that the pressure error will vary by 30 psid (0-30) over a range of 0 to 100% steam flow. Thus every 1 psi change in the pressure error will change the steam flow demand (pressure controller output) by 3.33%.

   For example, we'll determine the output from the "A" pressure controller which normally has no bias applied to it.

   - Pressure setpoint = 926 psig
   - Throttle pressure = 941 psig
   - Bias is 0 volt = 0 psi

   Steam flow demand = (Pressure error + bias) x 3.33%/1 psia

   = (941 - 926 + 0) psi x 3.33%/1 psia
   = <u>49.9%</u>

4. As mentioned earlier, the effect of adding a negative bias signal to one pressure controller will place the other pressure controller in control. Procedurally, a -0.3 volt bias is applied, which is equivalent to -4.5 psi.

   Normally, pressure controller "B" has the -0.3 volt bias applied:

   Pressure setpoint = 926 psig
   Header pressure = 941 psig
   Bias is -0.3 volt = -4.5 psi

   Steam flow demand = (Pressure error + bias) x 3.33%/1 psi = (941 - 926 - 4.5) psi x 3.33%/1 psi = <u>34.9%</u>

   The output of the biased pressure controller "B" is about 15% (49.9%-34.9%) lower than the output of the unbiased pressure controller "A".

5. If reactor power is increased slightly so that throttle header pressure increases by 1 psi to 942 psig, the new steam flow demand signal will be:

   Steam flow demand = (942 - 926 -4.5) psi x 3.33%/1 psi
   = <u>38.2%</u>, an increase of 3.33%

6. The two pressure controller outputs are both fed to a High Value Gate (HVG). The HVG only passes the highest of the signals applied to it. In our example, the unbiased signal from "A" is passed and the biased signal from "B" is blocked. From this point on, the signal passed from the "A" controller becomes the <u>Pressure Control</u> signal.

LO-04p

7. The Pressure Control signal is passed to a Low Value Gate (LVG) where it is compared with a Flow Limiter signal. A LVG passes the lowest of the signals supplied to it.

LO-04q

   The Flow Limiter signal is an operator adjustable signal and is normally set at 110%. Its purpose is to limit the Pressure Control signal in the event of a circuit failure upstream calling for maximum steam flow demand. The Flow Limiter signal is normally blocked by this LVG.

Q#	*Question Description*	*Rev #*	*Rev Date*	*Topic Area*	*Diff*
RO59	ILT	0		Reactor/Turbine Pressure Regulator	

Q type	*Response Time*	*Max Point Value*	*Passing Point Value*	*Lesson #*
M/C		1		

Objective #	*Reference*	*K/A #*	*10CFR 55 41/43/45*
	2.2.77.1, Att. 1, 1.2.37	241000 K4.06	41(b)(7)

K/A Text:

K4.06 – Knowledge of REACTOR/TURBINE PRESSURE REGULATING SYSTEM design feature(s) and/or interlocks which provide for the following: Turbine trip.

Question:

During a Main Turbine startup, the DEH Overspeed Protection Control (OPC) circuit is actuated and the OPC solenoid valves de-energize.

Which one of the following describes the position of the Main Turbine Governor, Intercept, and Bypass Valves BEFORE the OPC circuit actuated and AFTER the OPC actuation and resultant functions are complete?

	Governor Valves	Intercept Valves	Bypass Valves
a.	Open and then closed.	Open and then closed.	Open and remain open.
b.	Open and then closed.	Open and then closed.	Open and then closed.
c.	Open and remain open.	Closed and remain closed.	Closed and then open.
d.	Open and remain open.	Closed and then open.	Open and then closed.

Answer:

ANSWER: a
When OPC activates, the OPC solenoid valves de-energize to drain the governor emergency trip header. The governor and intercept valves trip closed and the bypass valves throttle to maintain reactor pressure. During the main turbine startup (roll), the governor and intercept valves are open and the bypass valves are throttled to maintain reactor pressure according to changes in turbine steam demand during the roll up.

REFERENCE: 2.2.77.1, Att. 1, 1.2.37

K/A System: 241000
K/A Number: K4.06
K/A Value: 3.6
Cognitive Level: 2
Bank/Mod/New: New

Distracter b, c, d: See explanation above.

1.2.37 BWR CONTROL PANEL - The BWR Control Panel (Analog System) is located in the upper benchboard section of the main turbine control panel. The Analog System provides a second means, independent of the automatic system, of controlling the turbine valves in the event of an automatic system failure or during certain maintenance activities. The Analog System gives the operator manual control of pressure setpoint, governor valve position, and bypass valve position. Speed, acceleration, load, load rate, pressure, and pressure rate can all be directly managed by the operator. The Analog System also provides the overspeed protection control (OPC) circuitry. OPC controls turbine overspeed in the event that the turbine exceeds 103% speed. When OPC activates or is placed in TEST, the OPC solenoid valves deenergize to drain the governor emergency trip header. This causes the governor and intercept valves to close and the bypass valves (if ≥ 25% RTP) to open.

1.2.38 OPERATOR CONTROL PANEL A - The Operator Control Panel A is located in the lower vertical board section of the main turbine control panel and provides the operator with indication only. The indications available are turbine speed, megawatts, GV signal, valve position limit, and various DEH System alarm indicating lights.

1.2.39 VALVE TEST AND LATCH PANEL - The Valve Test and Latch Panel is located in the upper vertical board section of the main turbine control panel. It provides indication of stop valve, governor valve, bypass valve, intercept valve, and reheat stop valve positions. It also has buttons for latching the main turbine and for testing the intercept and reheat stop valves.

1.3 CONTROL SIGNAL TRACKING

1.3.1 In most cases, the automatic and manual signals to pressure setpoint, governor valve positioning, and bypass valve positioning will track each other. For example, with governor valves in auto, as the automatic signal is changed, the manual signal will follow it. This allows a bumpless transfer from auto to manual and vice versa. The following conditions apply to control signal tracking:

1.3.1.1 PRESSURE SETPOINT - the automatic and manual signals track each other in all modes.

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO 60	ILT	0	2/2001	EDG	

Q Type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C				COR002-02-02

Objective #	Reference	K/A #	10CFR 55 41/43/45
3, 4, 8	2.2.28.1	259001, A2.03	41(b)(7)

K/A Text:

A2.03 - Ability to (a) predict the impacts of the following on the REACTOR FEEDWATER SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of condensate pump(s).

Question:

The plant is at 100% power when a loss of ALL Condensate Pumps occurs. Reactor Feedwater Pump (RFP) suction pressure <u>just lowered below</u> 260 psig (*referenced as T= 0 seconds*) and continues to lower.

Assuming NO operator action is taken, which one of the following describes when both RFPs will be in the tripped state?

a. Both are tripped at T= 0 seconds.

b. Both are tripped at T= 5 seconds.

c. Both are tripped by T= 10 seconds.

d. Both are tripped by T= 15 seconds.

Answer:

ANSWER: d.
RFP "A" trips in 10 seconds and RFP "B" trips in 15 seconds after RFP suction pressure lowers below 260 psig.

REFERENCE: 2.2.28.1, 2.4.9.4.2

Tier: 2
Group: 1
K/A System: 259001
K/A Number: A2.03
K/A Value: 3.6
Cognitive Level: 1
Bank/Mod/New: Modified. Although evaluating the same knowledge as a current bank question, the question was reworded with a time reference for when the low suction pressure trip signal is met. Because of this time reference, all distracters were changed to a time reference from T=0 in the question.

Distracter a: See justification above.
Distracter b: See justification above.
Distracter c: See justification above.

Proposed references to be provided to applicants during the examination: None

2.3 Reactor feed pumps trip on low suction pressure of 260 psig after a time delay of 10 seconds for Pump A and 15 seconds for Pump B.

2.4 RHR-920MV and RHR-921MV, AOG STM SUPPLY VALVEs, close when all condensate pump breakers are open.

3. IMMEDIATE OPERATOR ACTIONS

3.1 Reduce recirculation flow, as necessary, to match feedwater flow and steam flow to maintain reactor water level.

3.2 Attempt to start tripped pumps.

3.3 If all condensate pumps tripped, perform following:

3.3.1 Verify both reactor feed pumps tripped.

3.3.2 Verify all condensate booster pumps tripped.

3.3.3 Verify closed RHR-920MV, AOG STM SUPPLY VALVE.

3.3.4 Verify closed RHR-921MV, AOG STM SUPPLY VALVE.

4. SUBSEQUENT OPERATOR ACTIONS

CAUTION - Entry conditions to the EOPs may exist.

4.1 Refer to Procedure 5.7.1 to determine if declaration of an Emergency Action Level (EAL) is appropriate.

4.2 If loss of pump(s) results in reactor scram, refer to Procedure 2.1.5 in conjunction with this procedure.

4.3 If loss of pump(s) results in loss of feedwater, refer to Procedure 2.4.9.4.4 in conjunction with this procedure.

4.4 If condensate and condensate booster pumps tripped due to loss of 4160V Bus 1A, 1B, or 1E, place switches for tripped pumps to STOP to prevent restart when power restored.

4.5 If all condensate flow is lost, perform following:

4.5.1 Close following valves to prevent draining CST to hotwell:

4.5.1.1 MC-807, CST RECIRC THROTTLING VALVE (RW-877-basement above Condensate Backwash Transfer Pump).

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO61	ILT	0		Reactor Water Level Control	

Q type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C		1		

Objective #	Reference	K/A #	10CFR 55 41/43/45
COR002-32-02-6.i	COR002-32-02	295002 A3.03	41(b)(7)

K/A Text:

A3.03 – Ability to monitor automatic operations of the REACTOR WATER LEVEL CONTROL SYSTEM including: Changes in main steam flow.

Question:

The plant is at 100% power. The Reactor Level Control system is maintaining RPV level at +35 inches in three (3) element control. The “C” main steam flow transmitter output goes to ZERO and remains at zero.

Which one of the following describes the change in RPV level and the magnitude of the change?

a. RPV level lowers and the reactor scrams on low level.

b. RPV level rises and stabilizes at approximately +47 inches.

c. RPV level lowers and stabilizes at approximately +23 inches.

d. RPV level rises and the reactor scrams when the main turbine trips.

Answer:

ANSWER:	c. Level lowers until it stabilizes at 23 inches. A reactor scram will not occur.
REFERENCE:	COR002-32-02
K/A System:	295002
K/A Number:	A3.03
K/A Value:	3.2
Cognitive Level:	2
Bank/Mod/New:	Bank
Distracter a:	The reactor will not scram.
Distracter b:	Level lowers. It will rise if a feedwater transmitter failed low.
Distracter d:	Level lowers. It will rise if a feedwater transmitter failed low.

b. SF drops to 70% (38 ma).

c. Flow error signal is 28.4 ma (38 - 40) x .8 + 30.

d. Modified level signal is 34.9 ma (33.3 - 28.4) x 1 + 30.

e. Negative error signal to integrator

f. Integrator output decreases.

g. Feed pumps speed decreases until error signal is zero, then integrator output is constant.

3. Final conditions

a. Power, SF, and FF are 70%

b. Actual water level 35"

V. ABNORMAL SYSTEM OPERATION

Fig 2
LO-06i; 09c,d

A. Loss of One Steam Flow Signal

1. Initial conditions

a. Power 100%

b. SF and FF are 100% (50 ma)

c. Actual water level 35" (33.3 ma)

d. Level setpoint 35" (33.3 ma)

e. Controllers in Auto

2. Sequence

a. One of the steam flow detectors is isolated.

b. SF signal drops to 75% (40 ma) - Remember actual steam flow remains at 100%.

c. Flow error signal is 22 ma (40 - 50) x .8 + 30.

d. Modified level signal is 41.3 ma (33.3 - 22) x 1 + 30.

e. Creates negative error signal in master controller, feed pumps speed decreases.

f. Water level begins to drop because FF is less than actual steam flow.

g. The decreasing vessel level input signal to the level/flow error network begins to offset the erroneous steam flow signal.

h. When actual level has decreased to 23" (a 12" level change) the level input to the level vs. flow error network is 25.3 ma.

i. This counters the 22 ma from the steam flow/feed flow comparator, and the error network output returns to 33.3 ma. The master controller error signal is now at 0.

j. The reactor level stabilizes at 23".

3. Final conditions

a. Power, SF, FF are 100%.

b. Water level is 23".

c. Total steam flow signal indicates 75%.

NOTE: The above discussion was assumed to occur at 100% power. For a lower reactor power, the level decrease will be smaller. (ie 50% power a 6" level decrease)

Fig 2
LO-06i; 09c,d

B. Loss of All Steam Flow Signals

1. Initial conditions

a. Power, SF, FF are 100% (50 ma).

b. Water level 35" (33.3 ma).

2. Sequence

a. Total SF signal is lost.

b. The error signal drives the steam flow/feed flow comparator output to minimum signal.

c. The water level would have to decrease below the scram setpoint to compensate for this flow error.

d. The reactor would scram on low water level.

Fig 2
LO-06j; 09c,d

C. Loss of One Feed Flow Signal

1. Initial conditions

a. Power, SF, FF are 100% (50 ma)

b. Water level 35" (33.3 ma)

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO62	ILT	0		Standby Gas Treatment System	

Q type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C		1		COR002-28-02

Objective #	Reference	K/A #	10CFR 55 41/43/45
	2.2.73, Att. 2, 1.3.12.2	261000, K4.01	41(b)(7)

K/A Text:

K4.01 – Knowledge of STANDBY GAS TREATMENT SYSTEM design feature(s) and/or interlocks which provide for the following: Automatic system initiation.

Question:

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

- Standby Gas Treatment (SGT) Exhaust Train 1A is being placed in service to support a surveillance test
- SGT Train 1B is aligned for STBY operation
- The control switch for SGT Fan 1A is placed in RUN

If SGT-AO-251, SGT Train 1A Outlet Valve remains closed, which one of the following describes the response of SGT trains 1A <u>and</u> 1B in the next one (1) minute?

a. Train 1A automatically shuts down. Train 1B remains off because a group 6 isolation signal is **<u>NOT</u>** <u>present</u>.

b. Train 1A automatically shuts down. Train 1B starts on low flow because a group 6 isolation signal is **<u>NOT</u>** <u>required</u>.

c. Train 1A runs until manually shutdown. Train 1B remains off because a group 6 isolation signal is **<u>NOT</u>** <u>present</u>.

d. Train 1A runs until manually shutdown. Train 1B starts on low flow because a group 6 isolation signal is **<u>NOT</u>** <u>required</u>.

Answer:

ANSWER: c.
Low flow in a train will cause the standby fan to start if it is in STBY provided the operating train flow is <800 scfm, and a group 6 isolation signal is present or sealed in. There is no group 6 isolation signal for the conditions presented.

REFERENCE: 2.2.73, Att. 2, 1.3.1.2.2

K/A System: 261000
K/A Number: K4.01
K/A Value: 3.7
Cognitive Level: 2
Bank/Mod/New: New

Distracter a: No conditions will develop on the operating train to cause it to trip within 1 minute.

Distracter b: No conditions will develop on the operating train to cause it to trip within 1 minute.

Distracter d: A group 6 isolation is required, the 1B train will NOT automatically start.

1.3.11.1 The 2.8 kW heater will turn on when the flow through the subsystem > 800 scfm flow, the temperature of the stream < 170°F, and the SGT-HTR-SGHA (SGT-HTR-SGHB) switch is in LOW.

1.3.11.2 The 5 kW heater will turn on when the flow through the subsystem > 800 scfm flow, the temperature of the stream < 170°F, and the SGT-HTR-SGHA (SGT-HTR-SGHB) switch is in MEDIUM.

1.3.11.3 Both air heaters will trip off if flow through the subsystem drops below 800 scfm or the temperature of the air stream > 170°F. Temperature Switches SGT-TS-540A (SGT-TS-540B) and SGT-TS-541A (SGT-TS-541B) will not reset until the temperature of the air in the subsystem is ~ 160°F after tripping at 170°F.

1.3.12 EF-R-1E (EF-R-1F), SGT A (B) EXHAUST FAN, starts when the following conditions are met:

1.3.12.1 The EF-R-1E (EF-R-1F) switch is in AUTO and a Group 6 isolation signal is received.

1.3.12.2 The EF-R-1E (EF-R-1F) switch is in STANDBY, SGT B (A) flow < 800 scfm, and a Group 6 isolation signal is present or sealed-in.

2. REFERENCES

2.1 TECHNICAL SPECIFICATIONS

2.1.1 LCO 3.6.4.3, Standby Gas Treatment (SGT) System.

2.2 UPDATED SAFETY ANALYSIS REPORT

2.2.1 Volume II, Section V, Subsection 3.3.4, Standby Gas Treatment System.

2.2.2 Volume III, Section VII, Subsection 17, Standby Gas Treatment System.

2.3 DRAWINGS

2.3.1 B&R Drawing 2020, Reactor Building H&V.

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO 63	ILT	0	2/2001	EDG	

Q Type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C				COR002-27-02

Objective #	Reference	K/A #	10CFR 55 41/43/45
3, 4, 8	2.2.71, Section 4.0 5.2.5, Section 2.7 COR002-27-02	264000, K1.04	41(b)(7) 41(b)(8)

K/A Text:

K1.04 Knowledge of the physical connections and/or cause-effect relationships between EMERGENCY GENERATORS (DIESEL/Jet) INCLUDING: Emergency generator cooling water system.

Question:

The unit is operating at 100% reactor power. SW pump alignment is as follows:

- SW pumps "A," "B" and "C" are operating
- Mode Selector switches for the "A" and "B" SW pumps are in STANDBY
- Mode Selector switches for the "C" and "D" SW pumps are in AUTO

A loss of offsite power occurs. Both DGs start and energize busses 1F and 1G.

Assume **NO** operator actions are taken.

Which one of the following describes the Service Water pumps that will be operating by design two (2) minutes after offsite power was lost?

a. A and B

b. A and C

c. B and D

d. C and D

Answer:

ANSWER: a.
Only the SW pumps selected to STANDBY start 13 seconds after buses 1F and 1G are energized from an emergency power source.

REFERENCE: 2.2.71, Section 4.0
5.2.5, Section 2.7

Tier: 2
Group: 1
K/A System: 264000
K/A Number: K1.04
K/A Value: 3.2
Cognitive Level: 2
Bank/Mod/New: Bank

Distracter b: See justification above.
Distracter c: See justification above.
Distracter d: See justification above.

Proposed references to be provided to applicants during the examination: None

2.7 When 4160V Buses 1F and 1G have been reenergized by emergency power, following sequential loading occurs:

2.7.1 RHR Pumps A and D start when buses have been reenergized if an RHR initiation signal is present.

2.7.2 RHR Pumps B and C start 5 seconds after buses have been reenergized if an RHR initiation is present.

2.7.3 CS Pumps A and B start 10 seconds after buses have been reenergized if a CS initiation is present.

2.7.4 SW pumps selected to standby start 13 seconds after buses have been reenergized.

2.7.5 REC pumps selected to standby start 20 seconds after buses have been reenergized.

2.8 Hydrogen AIR SIDE SEAL OIL BACKUP PUMP starts.

2.9 Main Turbine EMERG BEARING OIL PUMP starts.

2.10 RFPT A and B EMERGENCY OIL PUMPS start.

2.11 RRMG LUBE OIL PUMP C and LUBE OIL PUMP D start.

3. IMMEDIATE OPERATOR ACTIONS

3.1 Ensure all rods are full-in.

3.2 Ensure PCIS Groups 1, 2, 3, 6, and 7 isolations have occurred.

3.3 Ensure 4160V Buses 1F and 1G are being supplied by either Emergency Transformer or DGs.

3.4 Ensure SW pumps selected to standby have started.

3.5 Ensure REC pumps selected to standby have started.

3.6 Ensure following DC lube oil pumps have started:

3.6.1 Hydrogen AIR SIDE SEAL OIL BACKUP PUMP.

3.6.2 Main Turbine EMERG BEARING OIL PUMP.

3.6.3 RFPT A and B EMERGENCY OIL PUMPS.

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO 64	ILT	0	2/2001	EDG	

Q Type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C				COR002-08-02

Objective #	Reference	K/A #	10CFR 55 41/43/45
9, 13	COR002-08-02	264000, A1.03	41(b)(7) 41(b)(8)

K/A Text:

A1.03 – Ability to predict and/or monitor changes in parameters associated with operating the EMERGENCY GENERATORS (DIESEL/ JET) controls including: Operating voltages, currents, and temperatures.

Question:

DG2 has been started and loaded to 3850 KW for the monthly surveillance when a reactor scram due to high drywell pressure occurs. Two (2) minutes following the LOCA, **ALL** offsite sources are lost.

Which one of the following describes the effect the above conditions will have on DG2 and 4160 Bus 1G?

a. DG2 engine <u>AND</u> output breaker will **NOT** trip. DG2 will remain connected to Bus 1G.

b. DG2 output breaker will trip when offsite power is lost. DG2 is **NOT** available until the Diesel Generator over current lockout is manually reset.

c. DG2 engine **AND** output breaker will trip when the LOCA signal is received. DG2 will automatically start and re-connect to Bus 1G when offsite power is lost.

d. DG2 output breaker will trip when the LOCA signal is received. DG2 output breaker will close when offsite power is lost.

Answer:

ANSWER: d.
The DG output breaker receives a trip signal opening the breaker when the LOCA signal occurs. The DG would then run unloaded. The DG will pick up 4160 Bus 1G when it is de-energized (LOOP).

REFERENCE: COR002-08-02

Tier: 2
Group: 1
K/A System: 264000
K/A Number: A1.03
K/A Value: 2.8
Cognitive Level: 2
Bank/Mod/New: Bank

Distracter a: See justification above.
Distracter b: See justification above.
Distracter c: See justification above.

Proposed references to be provided to applicants during the examination: None

preclude spurious undervoltage trips for 25 seconds if diesel generator output voltage drops during sequential loading or due to energizing a loaded bus. The load shedding feature will be reinstated 25 seconds after EG1(EG2) close (to allow completion of the load-sequencing action), but will only recur on first level (2300 VAC) undervoltage conditions.

4. In case of an emergency, the diesel can be stopped with an auto initiation signal present. It may be stopped by depressing the emergency stop push button on the local control panel. No emergency shutdown switch is available in the Control Room.

   If attempts to shutdown a diesel with the emergency stop push button fail, e.g. the control air system is lost, the diesel may be shutdown by manually tripping the air intake butterfly valves. This will shut off combustion air to the engine.

5. Each Diesel Generator is load tested once per month. If the Normal and Startup Transformers both de-energize concurrently while the Diesel Generator is connected to its bus, the Diesel Generator would then attempt to carry the load of two buses while continuing to operate with a droop characteristic. Overcurrent relay protection will trip breaker 1FA (1GB) and isolate the critical bus from 4160V bus 1A(1B) at 130% overcurrent.

LO-10h,i; LO-14e

   If an auto start signal is received while the Diesel Generator is connected to the bus, the diesel output breaker receives a 3 second trip signal. Receipt of an auto start signal transfers the governor and voltage regulator to the isochronous mode regardless of the DROOP PARALLEL switch position. Opening the output breaker, and then allowing the diesel to pick up the critical bus only if it is deenergized, ensures that the critical and noncritical buses have separated and that critical bus load shedding has occurred. This prevents the diesel from picking up excessive load which it would attempt to do in the isochronous mode.

6. A fire in the cable spreading room could damage the control cabling to both Diesel Generators such that their operation would be inhibited. Each Diesel Generator has been provided with four NORMAL-ISOLATE control switches, two mounted on their respective diesel engine panels and two on the EG1(EG2) breaker cubicles. The isolation switches are normally kept in their NORMAL position. When these red handled switches are concurrently placed in the ISOLATE position, the control cabling running between the Diesel Generator and the Control Room is isolated and bypassed. These switches protect the plant's safe shutdown capability in the event of a fire.

   If the DG isolation switches are placed in the ISOLATE position, automatic start of the diesel is blocked, the operator loses Panel C Diesel Generator and breaker (EG1/2) control, Panel C indicating lights for diesel control extinguish and annunciator C-1/C-3(C-4/C-4) DIESEL GEN 1(2) ISOLATION SW IN LOCAL alarms. The diesel can only be started from the local engine panel. The 1F(1G) bus voltage indicators and synchroscope on the diesel metering panel are inoperative. Since the instrumentation for paralleling is inoperative, breakers 1FA and 1FS (1GB and 1GS) are verified open prior to closing EG1(EG2). EG1(EG2) can only be closed by depressing the CLOSE button on its breaker.

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO 65	ILT	0	2/2001	CRDM	

Q Type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C				COR002-05-02, 11 COR002-04-02, 12

Objective #	Reference	K/A #	10CFR 55 41/43/45
COR002-05-02, 11 COR002-04-02, 12	COR002-08-02 COR002-04-02	201003, A2.05	41(b)(1) 41(b)(5)

K/A Text:

A2.05 – Ability to (a) predict the impacts of the following on the CONTROL ROD AND DRIVE MECHANISM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Reactor scram.

Question:

The plant is operating at 100% power near the end of cycle with all control rods fully withdrawn.
The scram inlet valve (CRD-AOV-126) for control rod 30-31 opens.

Which one of the following describes the response of the plant over the next five (5) minutes, including why?

Reactor power will ...

a. be downscale on APRMs. The reactor will scram due to high Scram Discharge Volume level.

b. remain at 100% reactor power. **NO** control rod motion will occur. **NO** leakage into the Scram Discharge Volume will occur.

c. lower, but the plant will continue to operate at power. The associated control rod will insert. **NO** leakage into the Scram Discharge Volume will occur.

d. lower, but the plant will continue to operate at power. The associated control rod will insert. The scram valve will leak into the Scram Discharge Volume, but **NO** scram will occur as the Scram Discharge Volume drain capacity exceeds the leakage from the scram valve.

Answer:

ANSWER: c.

REFERENCE: 2.4.1.1.3, Section 4.4

Tier: 2
Group: 1
K/A System: 201003
K/A Number: A2.05
K/A Value: 4.1
Cognitive Level: 2
Bank/Mod/New: Bank

Distracter a: A reactor scram will not occur. The SDV level will not change.
Distracter b: The control rod will insert into the core. A single control scram will reduce reactor power.
Distracter d: No leakage will occur into the SDV.

Proposed references to be provided to applicants during the examination: None

The exhaust water line connects the exhaust water header to the HCU directional control valves. It is normally pressurized to reactor pressure.

g. DRIVE WATER LINE

This line connects the drive water header to the HCU manifold. It is normally pressurized to 265 psi above reactor pressure.

Fig 5,6
LO-05g, 11h

3. Scram Valves

a. Scram inlet (126) valves

Air pressure is applied to the top of the diaphragm to hold the valve closed. When the air pressure is lost, the spring opens the valve.

The opening of the scram inlet (126) valve discharges the HCU accumulator to the CRDM underpiston area.

There is a position indicating switch on each of the scram valves (126 and 127 valves). When both valves are full open, a scram light for that control rod in the Control Room on the full core display is energized.

LO-5g, 11h

b. Scram outlet (127) valve

Like the scram inlet valve, the scram outlet valve is held closed by air pressure. When the air pressure is lost, the spring opens the valve.

Opening the scram outlet valve vents the CRDM overpiston area to the scram discharge volume.

c. Scram valve (126 and 127) operating characteristics

The scram outlet valve will open quicker than the scram inlet valve. To accomplish this, the outlet valve spring is adjusted for a stronger opening force, and the outlet valve air vent piping is shorter. This prevents a buildup of high pressure in the CRDM, which could occur if the inlet valve opened before the outlet valve.

LO-16c

d. Each valve is a globe valve with teflon seats to minimize leakage. If either valve leaked, the control rod could drift inward.

Fig 4, 5
LO-01h
SO-02f.1)

4. Scram pilot valves (117 and 118)

a. Each HCU has two three-way solenoid operated scram pilot valves. The power supply for the 117 valve is RPS-A and for the 118 valve is RPS-B. Both valves are normally energized.

LO-05h

b. If one or both of the valves are energized, they direct the air to the scram valves to maintain the scram valves closed.

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO 66	ILT	0	2/2001	CRDM	

Q Type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C				COR002-26-02

Objective #	Reference	K/A #	10CFR 55 41/43/45
8, 9	2.4.1.1.3 4.2	201006, K3.01	41(b)(7)

K/A Text:

K3.01 – Knowledge of the effect that a loss or malfunction of the ROD WORTH MINIMIZER SYSTEM (RWM) (PLANT SPECIFIC) will have on the following: Reactor manual control system.

Question:

The plant is operating at 9% reactor power. All control rods in the current rod group are at their insert limit of 36. One of the control rods in the current group drifts in from position 36 to position 00.

Which one of the following describes the effect on the Rod Worth Minimizer (RWM) if the drifting control rod is selected?

a. Insert Error only. A control rod block will **NOT** be enforced.

b. Withdrawal Error only. A control rod block will **NOT** be enforced.

c. Insert Error <u>and</u> Select Error. A control rod block will be enforced.

d. Insert Error <u>and</u> Withdrawal Error. A control rod block will be enforced.

Answer:

ANSWER: a.

A Select Error occurs when a non-error rod is selected. The drifting rod is an error rod. The rod will not be a Withdrawal Error at position 00. A rod past its' insert limit is an insert error. No rod block occurs for a single insert error.

REFERENCE: 2.4.1.1.3
4.2

Tier: 2
Group: 2
K/A System: 201006
K/A Number: K3.01
K/A Value: 3.2
Cognitive Level: 1
Bank/Mod/New: Bank

Distracter b: See justification above.
Distracter c: See justification above.
Distracter d: See justification above.

Proposed references to be provided to applicants during the examination: None

2.2.11 Manual Bypassed Mode - RWM is in MANUAL BYPASSED mode, as indicated on RWM IDT in RWM MODE display, if RWM BYPASS keylock switch on Panel 9-5 is in BYPASS position or RWM is OFF LINE. This mode will continue to be displayed even after RWM is unbypassed and is ON LINE until RWM output buffer is reinitialized.

2.2.12 Insert Error - An insert error occurs when a rod contained in currently latched group or any higher group is inserted past insert limit for that group or if a rod contained in a lower group is inserted past withdraw limit for the lower group. RWM IDT displays up to two rods with insert errors in INSERT ERROR windows.

2.2.13 Withdraw Error - A withdraw error occurs when a rod contained in currently latched group or any lower group is withdrawn past withdraw limit for the group or if a rod contained in a higher group is withdrawn past insert limit for the higher group. RWM IDT displays one rod with a withdraw error in WITHDRAW ERROR window.

2.2.14 Select Error - A select error occurs when a rod is selected other than one contained in currently latched group, or other than one causing a withdraw, insert, or group notch error, if such an error exists. A select error will also occur if position of rod selected is unknown (indicates "**", "99" or "-99"). When a select error is made, SELECT ERROR indicator on RWM IDT will turn red.

2.2.15 RPIS/RWM Hardware Error - Any of the following conditions indicate a failure of hardware required by RPIS or RWM Programs to function:

2.2.15.1 RPIS data acquisition system read/write failure.

2.2.15.2 RPIS hardware interface failure.

2.2.15.3 Rod selected and driving/no rod selected mismatch. RMCS indicates a rod selected and moving but RPIS does not indicate a rod is selected.

2.2.15.4 LPSP indication is on while LPAP indication is off.

2.2.15.5 Insert or withdraw permissive/echo mismatch.

2.2.16 When a hardware error occurs, RWM MODE display on RWM IDT indicates RPIS H/W TROUBLE and all rod movement is blocked.

2.2.17 Withdraw Block - A condition where rod withdrawal permissive signal applied by RWM Program is removed, thus preventing control rod withdrawal. When reactor power is below LPSP, RWM Program applies withdraw blocks for the following reasons:

2.2.17.1 A withdraw error exists and selected rod is not a rod that is causing an insert error.

2.2.18 A withdraw block occurs if a hardware error exists, unless RWM is operating in AUTO BYPASSED Mode or is manual bypassed with keylock switch.

Q#
RO67

Question Description
ILT

Rev #	***Rev Date***	***Topic Area***
0	2/20/01	Recirculation System

Diff

Q type	***Response Time***	***Max Point Value***	***Passing Point Value***	***Lesson #***
M/C		1		

Objective #	***Reference***	***K/A #***	***10CFR 55 41/43/45***
	2.4.2.2.1, 2.3.2.26, 2.4.2.2.4	202001 K1.18	41.2 to 41.9 / 45.7 to 45.8

K/A Text:

K1.18 – Knowledge of the physical connections and/or cause-effect relationships between RECIRCULATION SYSTEM and the following: RHR shutdown cooling mode.

Question:

A Reactor shutdown has just been completed, and preparations are being made to put "B" loop of RHR in Shutdown Cooling. "A" Reactor Recirculation pump is running and "B" Reactor Recirculation pump is secured. The following annunciators have alarmed:

- 9-4-3/E-2, RRMG A FIELD GROUND
- 9-4-3/A-1, RRMG A BKR 1CS TRIP

Which one of the following will minimize Reactor Vessel Bottom Head Temperature Gradients?

a. Maximize CRD flow and minimize RWCU blowdown.

b. Terminate RWCU blowdown and maximize CRD cooling flow.

c. Maximize CRD flow and raise RPV water level to at least +48 inches.

d. Maximize RWCU blowdown and raise RPV level to at least +48 inches.

Answer:

ANSWER: d. Maximize RWCU blowdown and raise RPV level to at least +48 inches.

REFERENCE: 2.4.2.2.1, 2.3.2.26, 2.4.2.2.4

K/A System: 202001
K/A Number: K1.18
K/A Value: 3.3
Cognitive Level: 1

Justification: Reducing CRD flow lowers the amount of cold water in the lower RPV head while raising RPV water level promotes natural circulation. Increasing RWCU blowdown helps prevent stratification in the RPV lower head.

Distracter a, b, c: Maximizing CRD flow will increase the introduction of cold water into the RPV bottom head.

SOURCE: Cooper Exam Bank

CAUTION - If RWCU System is isolated or cannot be placed in service to obtain an accurate bottom head drain temperature, Technical Specification SR 3.4.9.3 shall be considered not met.

4.2.2 Determine bottom head drain temperature from Point 6 on NBI-TR-89, REACTOR VESSEL METAL TEMPERATURE RECORDER (Panel 9-21), or PMIS Point M180.

4.2.3 Compute temperature differential by subtracting temperature obtained in Step 4.2.2 from temperature obtained in Step 4.2.1.

4.2.4 Maintain minimum vessel metal temperature vs. pressure in top head region in safe operating region of Technical Specification Figure 3.4.9-1.

4.3 If bottom head thermal stratification occurs, perform following:

4.3.1 If reactor is scrammed, reset scram as soon as possible.

4.3.2 Slowly lower CRD System cooling flow to 10 gpm by adjusting CRD-FC-301 CRD FLOW CONTROL (Panel 9-5).

4.3.3 Restore and maximize RWCU System flow per Procedure 2.2.66.

4.3.4 When bottom head temperature vs. saturation temperature is < 140°F, start an RR pump per Procedure 2.2.68.1.

CAUTION - If depressurization is necessary, extreme caution shall be used. A rapid depressurization will result in higher natural circulation flow and excessive heatup of bottom head.

4.3.5 If unable to restore bottom head temperature vs. saturation temperature to < 140°F, slowly depressurize to restore temperature differential using following guidance:

4.3.5.1 Limit rate of depressurization to 25°F for first hour.

4.3.5.2 Limit rate of depressurization to 50°F for second hour.

4.3.5.3 Limit rate of depressurization to 90°F/hr average over any 1 hour for hours 3 and beyond.

5. PROBABLE CAUSE

5.1 None.

4. SUBSEQUENT OPERATOR ACTIONS

[ ] **CAUTION** - Operation in Stability Exclusion Region is prohibited.

[ ] 4.1 If one RR pump trips, perform following:

[ ] 4.1.1 For operating RR pump, ensure RRFC-SIC-16A(B), SPEED CONTROL, is in MANUAL.

[ ] 4.1.2 For tripped RR pump, close RR-MO-53A(B), PUMP DISCHARGE VLV, for 5 minutes.

[ ] 4.1.3 After RR-MO-53A(B) has been closed for 5 minutes, open valve for 6 seconds. Continue with remaining steps in Section 4 of this procedure while waiting to open RR-MO-53A(B).

[ ] 4.1.4 If power is available, ensure operating RR pump is transferred to Startup Transformer per Procedure 2.2.18.

[ ] 4.1.5 Perform single loop operation actions of Procedure 2.2.68.1 concurrently with remaining steps in Section 4 of this procedure.

[ ] 4.2 Dispatch operators to R-976-W and Non-Critical Switchgear Room to record all lockout relays and targets for tripped pump.

[ ] **NOTE** - RPV temperature stratification can occur at < 20% total core flow.

[ ] 4.3 Enter Procedure 2.4.2.2.4 upon indications of reactor vessel temperature stratification.

[ ] 4.4 If one pump is running, align RRMG H&V System per single RRMG Set operation section or shut down system if both pumps are tripped per Procedure 2.2.85.

5. PROBABLE CAUSE

[ ] 5.1 Following conditions will cause trip of RRMG set drive motor breaker:

[ ] 5.1.1 4160V Bus 1C(1D) lockout relay.

[ ] 5.1.2 RR-MO-43A(B), SUCTION VLV, < 90% open.

[ ] 5.1.3 RR-MO-53A(B) < 90% open after 2 minute time delay from time drive motor breaker was closed has timed out.

[ ] 5.1.4 RR-MO-53A(B) not partially open after 1 minute time delay from time drive motor breaker was closed has timed out.

1. CONTINGENCY ACTIONS

1.1 If RHR Subsystem available <u>and</u> plant conditions allow, place RHR Subsystem in SDC Mode per Procedure 2.2.69.2.

1.2 Control RPV level > 48" to aid in thermal convection flow.

[ ] **<u>CAUTION</u> -** Step 1.3 shall <u>not</u> be performed if blade guides are in RPV <u>or</u> if a fuel bundle is removed from around core instrumentation.©

1.3 Place or maintain one RR pump in service per Procedure 2.2.68, if available.

1.4 Place RWCU System in service per alternate heat removal section of Procedure 2.2.66.©

[ ] **<u>NOTE</u> -** Time to boiling and time to core uncovery graphs are based on conservative estimates; actual times are longer than indicated.

1.5 Review Attachment 5, monitor following temperatures and pressures frequently, and log every 4 hours:©

[ ] 1.5.1 If an RR pump is in-service, monitor RR-TI-151A(B).

[ ] 1.5.2 If an RR pump is <u>not</u> in service, monitor RPV metal temperatures on NBI-TR-89, REACTOR VESSEL METAL TEMPERATURE RECORDER (Panel 9-21), for indications of stratification and approach to boiling.

[ ] 1.5.3 If RWCU is in-service, monitor inlet temperature on RWCU-TI-137, TEMP IND (Panel 9-4).

[ ] 1.5.4 Monitor reactor pressure PMIS Points B025, N013, and N014 for indication of pressurization.

1.6 If RPV head is <u>on</u>, perform following:©

[ ] 1.6.1 Close reactor head vents when any of following are met:

[ ] 1.6.1.1 Average reactor coolant temperature reaches 212°F.

[ ] 1.6.1.2 RPV pressure is rising.

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO 68	ILT	0	2/2001	RWCU	

Q Type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C				COR001-20-02

Objective #	Reference	K/A #	10CFR 55 41/43/45
4, 7	2.2.68.1 COR001-20-02	204000, A3.03	41(b)(5) 41(b)(7)

K/A Text:

A3.03 – Ability to monitor automatic operations of the REACTOR WATER CLEANUP SYSTEM including: Response to system isolations.

Question:

The unit is in MODE 2 with a startup in progress. Reactor pressure is being maintained at 300 psig using the main turbine bypass valves. The "A" reactor recirculation pump trips and then an inadvertent Group 3 isolation signal is received.

Assume **NO** operator action is taken. Which one of the following describes the consequence on the plant?

a. Reactor water level will rise to the high level trip setpoint.

b. Reactor water level will lower and a reactor scram will be received.

c. A prerequisite for the "A" reactor recirculation pump re-start is unable to be determined.

d. RWCU non-regenerative heat exchanger outlet temperature will rise, damaging the demineralizer resin.

Answer:

ANSWER: c.

REFERENCE: 2.2.68.1

Tier: 2
Group: 2
K/A System: 204000
K/A Number: A3.03
K/A Value: 3.6
Cognitive Level: 2
Bank/Mod/New: Bank

Distracter a: Reactor water level will rise but will be within the required band (a shutdown is not required based on water level)
Distracter b: Reactor water level will rise.
Distracter d: Temperature will lower.

Proposed references to be provided to applicants during the examination: None

1. PURPOSE

This procedure provides instructions for Operations personnel to operate the Reactor Recirculation (RR) System after initial startup of the system. This procedure will be used in conjunction with Procedure 2.2.68.

2. PRECAUTIONS AND LIMITATIONS

2.1 GENERAL PRECAUTIONS

[ ] 2.1.1 Do <u>not</u> exceed RRMG Set bearing oil temperatures of 194°F.

[ ] **<u>NOTE</u>** - Vibration monitoring system will cause an alert annunciator to alarm prior to danger alarm.

[ ] 2.1.2 If two or more RR pump/motor vibration monitors for a given pump exceed danger setpoint, trip affected pump.

[ ] 2.1.3 To prevent cold water stratification when an RR pump has tripped, attempt to restart pump(s) as soon as allowable.

[ ] 2.1.4 Do <u>not</u> operate RR pumps any more than is absolutely necessary when suction pressure is below 300 psig. It is preferable to shut down RR pumps for a day or more and restart them, if need be, than to operate them continuously at low pressures.

[ ] 2.1.5 To prevent uncontrolled RR pump speed changes due to feedwater flow interlock, ensure flow controllers are set to minimum when feedwater flow < 20% or prior to reducing below 20%.

[ ] 2.1.6 If operation is in Stability Exclusion Region <u>or</u> abnormal flux oscillations are observed, enter Procedure 2.4.1.6.

[ ] 2.1.7 Operation of RRMG Sets at greater than rated speed (100% or 1120 rpm) may cause high vibration and damage to reactor internal components, large and small piping, and pipe-mounted equipment.©

2.2 PUMP START PRECAUTIONS

[ ] 2.2.1 Do <u>not</u> start RR pump in an idle loop unless loop temperature is within 50°F of core inlet temperature. This is to avoid possible pump damage due to differential expansion between pump shaft and collar welded to shaft.

[ ] 2.2.2 Do <u>not</u> start a pump when > 70% rod line.

[ ] 2.2.3 Do <u>not</u> start RR pump with MG Set oil temperature < 90°F.

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO 69	ILT	0	2/2001	RPIS	

Q Type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C				

Objective #	Reference	K/A #	10CFR 55 41/43/45
	2.4.5.6, Step 6.1	214000, K5.01	41(b)(6) 41(b)(2)

K/A Text:

K5.01 – Knowledge of the operational implications of the following concepts as they apply to ROD POSITION INFORMATION SYSTEM: Reed switches.

Question:

During a control rod sequence exchange at power (currently 75% power), it is determined that the Rod Position Information System (RPIS) reed switch at position 40 for control rod 30-33 is failed.

Which one of the following describes the plant condition necessary to repair the failed reed switch for control rod 30-33?

a. Shutdown with a drywell entry.

b. Shutdown without a drywell entry.

c. At power but power must be lowered to 50%.

d. At power and can be performed at the current power.

Answer:

ANSWER: a.
Repairs to replace a CRDM position probe (reed switch) must be deferred until a drywell entry may be made.

REFERENCE: 2.4.5.6, Step 6.1

Tier: 2
Group: 2
K/A System: 214000
K/A Number: K5.01
K/A Value: 2.7
Cognitive Level: 1
Bank/Mod/New: New

Distracter b: Drywell entry is required.
Distracter c: Not permitted in MODE 1.
Distracter d: Not permitted in MODE 1.

Proposed references to be provided to applicants during the examination: None

6. DISCUSSION

6.1 Failures in the Rod Position Information System (RPIS) are usually not serious and appear as loss of group (fuse) indications. Correct control rod position information is required for proper reactor operation and repairs should be made as soon as practical. Repairs to replace a CRDM position probe must be deferred until a drywell entry may be made.

7. REFERENCES

7.1 TECHNICAL SPECIFICATIONS

7.1.1 LCO 3.1.3, Control Rod OPERABILITY.

7.1.2 LCO 3.9.4, Control Rod Position Indication.

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO 70	ILT	0	2/2001	RHR	

Q Type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C				COR002-23-02

Objective #	Reference	K/A #	10CFR 55 41/43/45
	2.2.19.A, Att. 2, MCC-Q	219000, A2.05	41(b)(7)

K/A Text:

A2.05 – Ability to (a) predict the impacts of the following on: RHR/LPCI: TORUS/ SUPRESSION POOL COOLING MODE and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A.C. electrical failures.

Question:

The plant was at 100% power when a loss of MCC-Q occurred. With MCC-Q still de-energized, an ATWS occurs and the CRS directs maximizing suppression pool cooling.

Which one of the following describes how a loss of MCC-Q affects the ability to maximize suppression pool cooling?

a. Only the "A" loop of RHR can be used.

b. Only the "B" loop of RHR can be used.

c. RHR Service Water is not available to either loop.

d. Only one Service Water Booster Pump per loop is operable.

Answer:

ANSWER: b.
Power is lost to RHR-MO-34A, LOOP A TORUS COOLING INBD THROTTLE VLV. Only the "B" loop can be used.

REFERENCE: 2.2.19.A, Att. 2, MCC-Q

Tier: 2
Group: 2
K/A System: 219000
K/A Number: A3.05
K/A Value: 3.3
Cognitive Level: 1
Bank/Mod/New: New

Distracter a: The "A" loop cannot be used. Power is lost to RHR-MO-34A, LOOP A TORUS COOLING INBD THROTTLE VLV.
Distracter c, d: Loss of MCC Q does not affect the SWBPs, however power is lost to SW-MO-89A in the A loop. The B loop of RHR Service Water is unaffected.

Proposed references to be provided to applicants during the examination: None

MCC-Q - (R-903-NW) FED FROM MCC-K

	DESCRIPTION	NORMAL POSITION	PERFORMED BY	VERIFIED BY	COMMENTS
1C	REC-MO-1329 AUGMENTED RADWASTE SUPPLY	ON			
1D	WELDING RECEPTACLES COL L11.7 & P11.7	OFF			MODE 1, 2, 3
1E	RHR-MO-15C RHR PUMP C SDC SUCTION VALVE (RH-503MV)	ON			
2A	RHR-MO-13A RHR PUMP A TORUS SUCTION VALVE (RH-504MV)	ON			
2B	RHR-MO-65A RHR HX-A INLET VALVE (RH-505MV)	ON			
2C	RHR-MO-12A RHR HX-A OUTLET VALVE (RH-506MV)	ON			
2D	RHR-MO-66A RHR HX-A BYPASS VALVE (RH-507MV)	ON			

	DESCRIPTION	NORMAL POSITION	PERFORMED BY	VERIFIED BY	COMMENTS
3A	RHR-MO-15A RHR PUMP A SDC SUCTION VALVE (RH-502MV)	ON			
3B	RHR-MO-26A LOOP A DW SPRAY OUTBD THROTTLE VLV (RH-510MV)	ON			
3C	RHR-MO-31A LOOP A DW SPRAY INBD VALVE (RH-511MV)	ON			
3D	RHR-MO-13C RHR PUMP C TORUS SUCTION VALVE (RH-513MV)	ON			
4A	RHR-MO-1485 RHR-MO-921 BYPASS VALVE	ON			
4B	RHR-MO-38A LOOP A TORUS SPRAY INBD THROTTLE VLV (RH-515MV)	ON			
4C	RHR-MO-39A TORUS COOLING/TORUS SPRAY OUTBD VLV (RH-514MV)	ON			
4D	RHR-MO-16A RHR LOOP A MIN FLOW VLV (RH-517MV)	ON			

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO 71	ILT	0	2/2001	RHR	

Q Type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C				COR002-23-02 INT008-06-13

Objective #	Reference	K/A #	10CFR 55 41/43/45
	INT008-06-13	226001, K6.05	41(b)(7) 41(b)(8)

K/A Text:

K6.05 - Knowledge of the effect that a loss or malfunction of the following will have on the RHR/LPCI: CONTAINMENT SPRAY SYSTEM MODE: Suppression pool (temperature level and pressure).

Question:

Per the Emergency Operating Procedure Bases, which one of the following is the basis for initiating Drywell Spray when Torus Pressure reaches 10 psig?

a. To prevent damage to the SRV tailpipes.

b. To prevent failure of the containment downcomer piping.

c. To prevent damage to the SRV T-quenchers and supports.

d. To prevent non-condensable gases from collecting in the drywell.

Answer:

ANSWER: b.
Drywell spray is intiated when torus pressure reaches 10 psig to prevent chugging in the downcomers which can lead to cyclic failure of the downcomer to ring header junction. The other distracters are consequences associated with failure to perform other EOP steps or to comply with the EOP curves.

REFERENCE: INT008-06-13

Tier: 2
Group: 2
K/A System: 226001
K/A Number: K6.05
K/A Value: 3.4
Cognitive Level: 1
Bank/Mod/New: Bank

Distracter a: See justification above.
Distracter c: See justification above
Distracter d: See justification above

Proposed references to be provided to applicants during the examination: None

concurrently. This is necessary because the primary containment functions as a closed thermodynamic system and the transient responses of all primary containment parameters are directly interrelated. For example:

1. Changes in torus water temperature can directly change primary containment pressure.

2. Changes in drywell temperature can directly change primary containment pressure.

3. Changes in torus water level can directly change torus pressure.

Prioritization of any single flow path is not possible since symptomatic response requires independence from initiating events.

F. Primary Containment Pressure Flow path

1. PC/P-1 - Initial action to control primary containment pressure is same as that during normal plant operations: monitoring its status and using pressure control systems (SGT) as required to maintain pressure below the high drywell pressure scram set-point. ESP 5.8.17 provides detailed instructions for controlling primary containment pressure below 1.84 psig.

2. PC/P-2 and TS/1 through TS/3 - Operation of torus sprays can reduce primary containment pressure by condensing steam in the suppression chamber airspace, and by absorbing heat from the atmosphere through evaporative and convective cooling.

   If the CRS reaches PC/P-2 and torus pressure has already exceeded 10 psig, he is to spray the torus immediately and then proceed to steps which spray drywell.

   Torus sprays are started between a torus pressure of 1.84 psig (high drywell pressure scram set-point) and 10 psig (Suppression Chamber Spray Initiation Pressure). Below 1.84 psig, normal methods of pressure control are to be employed.

   T[redacted] to [redacted]

[redacted]

[redacted]
downcomer pi[redacted]epeated occurrence of this stress can cause these joints to experience fatigue failure thereby creating a pathway to the torus airspace which bypasses the suppression pool. Subsequent steam discharges through the downcomers would directly pressurize the torus rather than being discharged and condensed in the suppression pool.

Scale model tests have demonstrated that chugging will not occur so long as the drywell contains at least 1% noncondensibles. To preclude the occurrence of conditions under which chugging may happen, the SCSIP is conservatively defined by requiring 5% noncondensibles.

[redacted] is [redacted]

Torus sprays are started only if primary containment water level is below the elevation of the torus spray nozzles, 26.5 ft. If the torus spray nozzles are submerged, no spray action will occur and therefore there is no benefit in starting torus sprays.

Torus sprays are operated using either RHR pumps or RHRSW cross-tie, if necessary.

Initiation and continued operation of torus sprays with these pumps is permitted only if continuous operation of the pumps to be used is not required to assure adequate core cooling, unless the H_2/O_2 combustible limits of Table 7 have been met. Maintaining adequate core cooling takes precedence over initiating torus sprays because catastrophic failure of primary containment is not expected at the primary containment pressures requiring action in these steps. If the Table 7 limits have been met, spraying takes precedence over adequate core cooling to mitigate the effects of a deflagration. This spray action would be applicable only if the Table 7 limits existed while taking step PC/P-2 actions. The wording of these steps purposely permits alternating the use of

Q#	*Question Description*	*Rev #*	*Rev Date*	*Topic Area*	*Diff*
RO 72	ILT	0	2/2001	MAIN STEAM	

Q Type	*Response Time*	*Max Point Value*	*Passing Point Value*	*Lesson #*
M/C				INT008-06-06

Objective #	*Reference*	*K/A #*	*10CFR 55 41/43/45*
2	EOP-6A	239001, K3.08	41(b)(5) 41(b)(10)

K/A Text:

K3.08 – Knowledge of the effect that a loss of the MAIN AND REHEAT STEAM SYSTEM will have on the following: Decay heat removal.

Question:

Following a scram from full power several control rods failed to insert. Plant conditions are:

- MSIVs are closed
- APRMs indicate 4% power
- SRVs have opened but are currently closed
- RPV level is being controlled using HPCI
- RPV pressure is 950 psig and slowly rising
- There are **NO** indications of fuel failure or a steam line break
- Suppression pool temperature has reached the BIIT
- Main condenser is available

Per the EOPs, which one of the following describes the action to be taken to control RPV pressure?

a. Initiate RCIC in the test mode with suction from the ECST.

b. Open the MSIVs and maintain pressure below 1050 psig using the turbine bypass valves.

c. Open SRVs in any order as needed to maintain pressure below the Heat Capacity Temperature Limit.

d. Open SRVs to maintain pressure below 940 psig. The opening sequence for the valves must be followed.

Answer:

ANSWER: b.

REFERENCE: EOP-6A

Tier: 2
Group: 2
K/A System: 239001
K/A Number: K3.08
K/A Value: 3.4
Cognitive Level: 2
Bank/Mod/New: Bank

Distracter a: Only used if the MSIVs are not available.
Distracter c: Not required until the suppression pool water temperature cannot be maintained below the HCTL.
Distracter d: Not required at this time since the SRVs are not cycling.

Proposed references to be provided to applicants during the examination: None

Q#	*Question Description*	*Rev #*	*Rev Date*	*Topic Area*	*Diff*
RO 73	ILT	0	2/2001	MAIN TURBINE AUXILIARIES	

Q Type	*Response Time*	*Max Point Value*	*Passing Point Value*	*Lesson #*
M/C				COR001-14-01

Objective #	*Reference*	*K/A #*	*10CFR 55 41/43/45*
	2.4.9.1.7, Section 3	245000, A2.02	41(b)(7)

K/A Text:

A2.02 - Ability to (a) predict the impacts of the following on MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of lube oil.

Question:

With the plant at 100% power, the Main Turbine Bearing Oil Pump fails. Per 2.4.9.1.7, "Main Turbine Bearing Oil Pump Failure," which one of the following describes the immediate operator action(s)?

a. Insert a manual reactor scram and then trip the turbine.

b. Trip the turbine and then insert a manual reactor scram.

c. Verify started or manually start the DC powered bearing oil pump.

d. Reduce recirculation flow to 35 mlbm/hr and monitor turbine vibration.

Answer:

ANSWER: c.
The immediate operator action is to verify the emergency bearing oil pump running.

REFERENCE: 2.4.9.1.7, Section 3

Tier: 2
Group: 2
K/A System: 245000
K/A Number: A2.02
K/A Value: 3.3
Cognitive Level: 1
Bank/Mod/New: New

Distracter a: If the emergency bearing oil pump cannot be started, then a turbine trip is required. The turbine is tripped first.
Distracter b: If the emergency bearing oil pump cannot be started, then a turbine trip is required. The turbine is tripped first.
Distracter d: There is no requirement in 2.4.9.1.7 to reduce recirculation flow. The immediate operator action is to verify the emergency bearing oil pump running.

Proposed references to be provided to applicants during the examination: None

2.1.4 The white light above the EMERG BEARING OIL PUMP control switch is on and bearing oil pressure as indicated on LOGT-PI-86 is > 20 psig.

2.1.5 Turning gear DRIVE MOTOR tripped and bearing oil pressure as indicated on LOGT-PI-86 is ≤ 3 psig.

3. IMMEDIATE OPERATOR ACTIONS

3.1 Verify the TURNING GEAR OIL PUMP, HP SEAL OIL BACKUP PUMP, and the EMERG BEARING OIL PUMP are running.

4. SUBSEQUENT OPERATOR ACTIONS

4.1 If the turbine trips at power:

4.1.1 Above 30% turbine first stage pressure:

4.1.1.1 Verify the reactor scrams.

4.1.1.2 Refer to Procedure 2.1.5.

4.1.2 Below 30% turbine first stage pressure, the reactor does not scram.

4.1.2.1 Refer to Procedure 2.4.9.1.10.

4.2 If the turning gear and emergency oil pumps fail:

4.2.1 During startup:

4.2.1.1 Secure steam admission to the turbine by simultaneously depressing the TURB TRIP 1 and TURB TRIP 2 pushbuttons.

4.2.1.2 Break vacuum by opening AR-MO-150, VACUUM BREAKER.

4.2.1.3 Check the following pump breakers CLOSED and try to get the pumps started:

a. MN TURB Turning Gear Oil Pump, MCC L - Breaker 1C.

b. Main Turbine Emergency Oil Pump, 250V DC Turbine Building Starter Rack.

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO 74	ILT	0	2/2001	CONDENSATE	

Q Type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C				COR002-02-02

Objective #	Reference	K/A #	10CFR 55 41/43/45
	2.3.2.2, Step 1.1, 2.5	256000, K1.06	41(b)(7)

K/A Text:

K1.06 – Knowledge of the physical connections and/or cause-effect relationships between REACTOR CONDENSATE SYSTEM and the following: Extraction steam system.

Question:

The plant is at 65% power with the following annunciators in alarm:

- A-2 / C-5, HEATER HIGH LEVEL
- A-2 / C-6, HEATER HIGH LEVEL TRIP

Which one of the following describes the state (open/closed/tripped) of the steam and condensate valves for the affected heater A3?

	CD-AO-LCV-62A Heater-To-Heater Valve	CD-AO-LCV-62B Heater-To-Condenser Valve	ES-AO-NRV5 / NRV6 Turbine-To-Heater Valves	ES-AO-DV1 / DV2 Steam Dump Valves
a.	Open	Open	Tripped	Open
b.	Closed	Closed	Open	Closed
c.	Closed	Open	Tripped	Open
d.	Open	Open	Tripped	Closed

Answer:

ANSWER: a.
The condensate valves (CD) remain open, the ES NRV is tripped (open or closed depending on steam flow), and Steam Dump Valve (DV) opens.

REFERENCE: 2.3.2.2, Step 1.1, 2.5

Tier: 2
Group: 2
K/A System: 256000
K/A Number: K1.06
K/A Value: 2.7
Cognitive Level: 2
Bank/Mod/New: New

Distracter b: See justification above.
Distracter c: See justification above.
Distracter d: See justification above.

Proposed references to be provided to applicants during the examination: None.

HEATER HIGH LEVEL TRIP

SETPOINT	CIC
Relay operation caused by:	
1. (3233) 16" below Heater A-5 centerline	1. CD-LS-60C
2. (3232) 15" below Heater A-4 centerline	2. CD-LS-61C
3. (3231) 15" below Heater A-3 centerline	3. CD-LS-62C
4. (3230) 12 7/8" below Heater A-2 centerline	4. CD-LS-63C
5. (3237) 16" below Heater B-5 centerline	5. CD-LS-65C
6. (3236) 13" below Heater B-4 centerline	6. CD-LS-66C
7. (3235) 15" below Heater B-3 centerline	7. CD-LS-67C
8. (3234) 12 1/4" below Heater B-2 centerline	8. CD-LS-68C

1. AUTOMATIC ACTIONS

1.1 Applicable turbine-to-heater valves (ES-AO-NRV) trip and act as check valves and steam dump valves (ES-AO-DV) open.

HEATER	VALVES	POSITION
A-2	ES-AO-NRV9 and ES-AO-NRV10	TRIPPED
	ES-AO-DV5 and ES-AO-DV6	OPEN
A-3	ES-AO-NRV5 and ES-AO-NRV6	TRIPPED
	ES-AO-DV1 and ES-AO-DV2	OPEN
A-4	ES-AO-NRV3	TRIPPED
A-5	ES-AO-NRV1	TRIPPED
B-2	ES-AO-NRV11 and ES-AO-NRV12	TRIPPED
	ES-AO-DV7 and ES-AO-DV8	OPEN
B-3	ES-AO-NRV7 and ES-AO-NRV8	TRIPPED
	ES-AO-DV3 and ES-AO-DV4	OPEN
B-4	ES-AO-NRV4	TRIPPED
B-5	ES-AO-NRV2	TRIPPED

2. OPERATOR OBSERVATION AND ACTION

2.1 Check associated CRT alarm messages to determine which input caused alarm.

2.2 If Heater A-5 and/or B-5 trip, direct main steam drain line discharge to condenser by use of main steam drain line selector switch on IR-1A.

(continued on next page)

2.3 Check affected heater level locally.

2.4 Check applicable heater-to-heater and heater-to-condenser valves (CD-AO-LCV), and steam dump valves (ES-AO-DV) are fully open.

2.5 Check applicable turbine-to-heater valves (ES-AO-NRV) are tripped (closed or intermediate).

HEATER	VALVES	POSITION
A-2	CD-AO-LCV63A and CD-AO-LCV63B	OPEN
	ES-AO-NRV9 and ES-AO-NRV10	TRIPPED
	ES-AO-DV5 and ES-AO-DV6	OPEN
A-3	CD-AO-LCV62A and CD-AO-LCV62B	OPEN
	ES-AO-NRV5 and ES-AO-NRV6	TRIPPED
	ES-AO-DV1 and ES-AO-DV2	OPEN
A-4	CD-AO-LCV61A and CD-AO-LCV61B	OPEN
	ES-AO-NRV3	TRIPPED
A-5	CD-AO-LCV60A and CD-AO-LCV60B	OPEN
	ES-AO-NRV1	TRIPPED
B-2	CD-AO-LCV68A and CD-AO-LCV68B	OPEN
	ES-AO-NRV11 and ES-AO-NRV12	TRIPPED
	ES-AO-DV7 and ES-AO-DV8	OPEN
B-3	CD-AO-LCV67A and CD-AO-LCV67B	OPEN
	ES-AO-NRV7 and ES-AO-NRV8	TRIPPED
	ES-AO-DV7 and ES-AO-DV8	OPEN
B-4	CD-AO-LCV66A and CD-AO-LCV66B	OPEN
	ES-AO-NRV4	TRIPPED
B-5	CD-AO-LCV65A and CD-AO-LCV65B	OPEN
	ES-AO-NRV2	TRIPPED

2.6 Check RF-TI-1, RFP DISCH HDR TEMP indicator, for a loss of feedwater heating and enter Procedure 2.4.9.4.7, if required.©

2.7 Determine cause of high level, correct, and reset the trip.

3. PROBABLE CAUSES

3.1 Main turbine trip.

3.2 Level control valve malfunction.

(continued on next page)

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO 75	ILT	0	2/2001	AC DISTRIBUTION	

Q Type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C				COR001-01-02

Objective #	Reference	K/A #	10CFR 55 41/43/45
6, 7, 13	2.2.13 COR001-01-02	262001, A1.05	41(b)(7)

K/A Text:

A1.05 – Ability to predict and monitor changes in parameters associated with operating the ELECTRICAL DISTRIBUTION controls including: Breaker lineups.

Question:

The reactor is operating at 100% power when the Auto-Transformer becomes de-energized.

Which one of the following will occur?

Power will be lost to ...

a. one (1) of the Reactor Recirculation pumps, requiring single loop operation.

b. the intake structure equipment, requiring a shutdown in accordance with GOP 2.1.5.

c. the 12.5 KV system, requiring the system to be restored from the Cornfield substation.

d. one (1) Condensate and one (1) Condensate Booster pump, resulting in a low RPV water level reactor scram.

Answer:

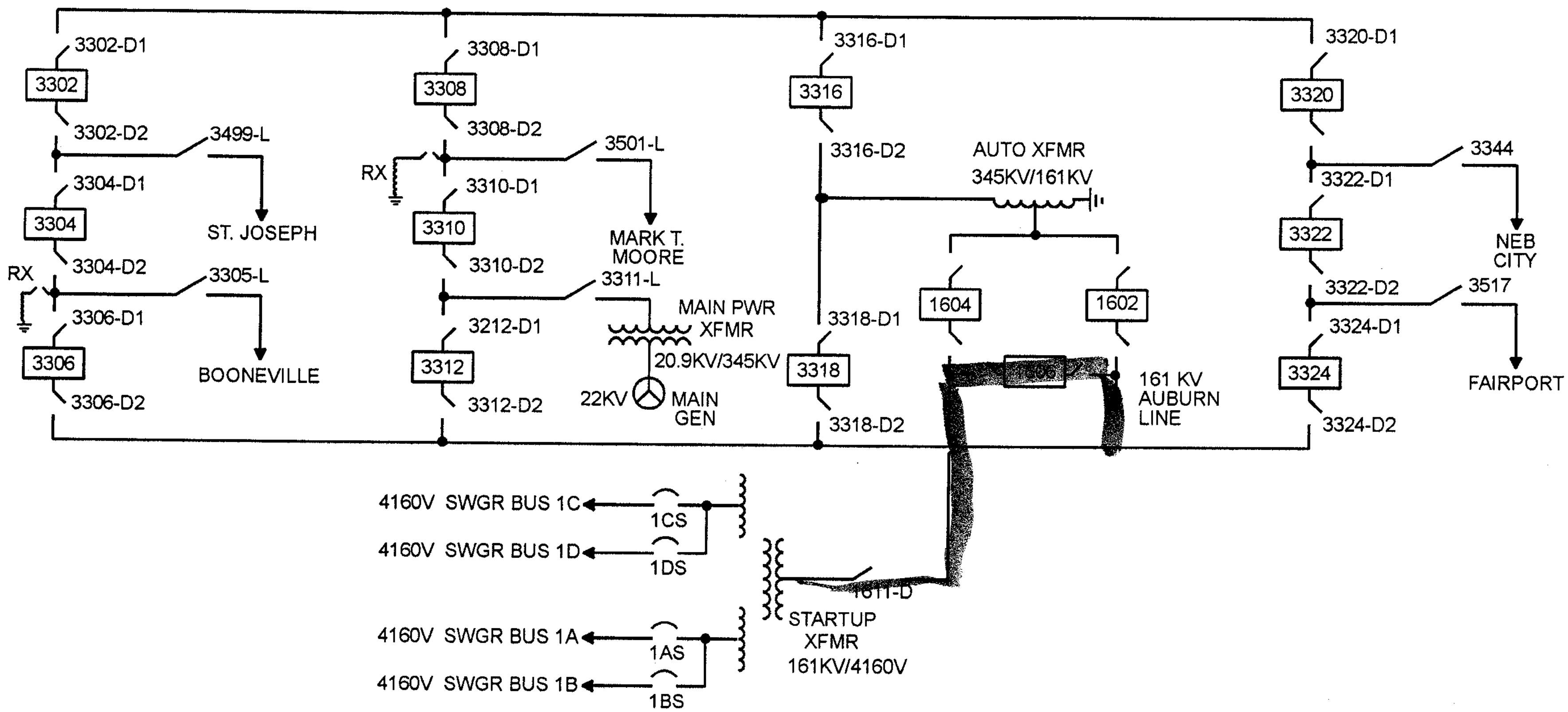
ANSWER: c.

REFERENCE: 2.2.13

Tier: 2
Group: 2
K/A System: 262001
K/A Number: A1.05
K/A Value: 3.2
Cognitive Level: 2
Bank/Mod/New: Bank

Distracter a: The startup transformer will be supplied by the 161KV Auburn line.
Distracter b: The intake structure is not effected.
Distracter d: The normal transformer is NOT effected.

Proposed references to be provided to applicants during the examination: None



345/161 KV DISTRIBUTION

Figure 2, Rev. 9

COR001-01-01

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CXA06847

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO 76	ILT	0	2/2001	UPS	

Q Type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C				

Objective #	Reference	K/A #	10CFR 55 41/43/45
	2.4.6.7, Section 6	262002, 2.1.31	41(b)(6) 41(b)(7)

K/A Text:

2.1.31 – Ability to locate control room switches/ controls and indications and to determine that they are correctly reflecting the desired plant lineup.

Question:

The Reactor Operator observes the following indicators and components have LOST power:

- Neutron monitoring recorders
- Reactor vessel level controllers
- Rod Select power
- Rod Position Information System

Which one of the following describes the single power source loss that caused these observations?

a. MCC-R

b. 250 VDC A

c. Vital Instrument Power.

d. No Break Power Panel.

Answer:

ANSWER: d.
NBPP is lost. 250 VDC A is the normal supply to the NBPP. If 250 VDC A is lost, the internal static switch will automatically transfer to MCC-R. MCC-R is the NBPP alternate supply and its loss will not result in a loss of the NNBPP.

REFERENCE: 2.4.6.7, Section 6

Tier: 2
Group: 2
K/A System: 262002, 2.1.31
K/A Number: A1.05
K/A Value: 4.2
Cognitive Level: 2
Bank/Mod/New: New

Distracter a: MCC-R is the NBPP alternate supply and its loss will not result in a loss of the NBPP.
Distracter b: NBPP is lost. 250 VDC A is the normal supply to the NBPP. If 250 VDC A is lost, the internal static switch will automatically transfer to MCC-R.
Distracter c: Vital instrument power does not power these instruments and components.

Proposed references to be provided to applicants during the examination: None

4.21 When fault has been determined and isolated, energize NBPP per Procedure 2.2.22.

5. PROBABLE CAUSE

5.1 Electrical fault on NBPP.

5.2 Blown fuse.

6. DISCUSSION

6.1 Power from the NBPP is obtained through an inverter fed from 250 VDC Bus A. The inverter feeds NBPP through a static switch inside the inverter or a manual bypass switch on the inverter. An emergency AC power feed is also provided from MCC-R. When inverter output voltage or frequency is abnormal, the internal static switch will automatically transfer to MCC-R. This static switch can also be transferred to MCC-R using the NBPP PWR TRANSFER switch on Panel C or by depressing the ALTERNATE SOURCE SUPPLYING LOAD button on the inverter. The NBPP power supply can also be transferred by placing the MANUAL BYPASS SWITCH on the inverter to ALTERNATE SOURCE TO LOAD. The NBPP is necessary for the operation of the station but is not critical to station safety. This procedure outlines the actions to be taken in the event of the failure of any of the supplies to the NBPP.

6.2 NBPP feeds the following major loads: reactor vessel level controllers and instrumentation, high off-gas activity isolation logic timers and valve control power, ERP flow indicating transmitter which sends process flow signal to ERP Kaman, Gaitronics, rod select power, rod position information system, NAWAS System, Ronan CRTs and printers, neutron monitoring recorders, condensate pump, condensate booster pump and reactor feed pump minimum flow valve control power and alarms, Reactor Building exhaust plenum and drywell high range radiation recorders, main generator voltage regulator alarms, fire protection manual pull stations and alarms, REC System low pressure alarms and low pressure non-essential isolation valve logic, and the SGT System low flow to stack alarm.

6.3 NBPP also supplies backup power to DEH and the RFPT speed controllers. NBPP can also supply power to drywell fan coil unit temperature recorders, drywell nitrogen purge controls, drywell temperature indicators and alarm units, SW rad monitor sample flow selector, main condenser hotwell level indicators and controls, Kaman RICs and recorders, and PC-TR-24, SUPPR POOL TEMP RECORDER, when the NORMAL/ALTERNATE POWER SUPPLY - DW TEMP RECORDERS & RECORDERS - TORUS TEMP RECORDER TR-24 switch is placed to ALT.

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO 77	ILT	0	2/2001	DC DISTRIBUTION	

Q Type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C				COR002-16-02

Objective #	Reference	K/A #	10CFR 55 41/43/45
2, 8	2.4.2.3.1, Section 4.8 2.3.2.21, 9-3-1/E-1, E-2 COR002-16-02	263000, K4.01	41(b)(7)

K/A Text:

K4.01 – Knowledge of D.C. ELECTRICAL DISTRIBUTION design feature(s) and/or interlocks which provide for the following: Manual/automatic transfers of control: Plant-Specific.

Question:

During normal operation at 100% power, 125 VDC panel "A" is lost. Which one of the following describes the effect on the Low-Low Set SRV solenoids?

a. Remain powered from their normal power supply.

b. Automatically transfer to their alternate power supply.

c. Are **de-energized** without any other power supply available.

d. Are **de-energized** and must be manually transferred to their alternate power supply.

Answer:

ANSWER: b.
Both LLS logic channels are normally powered from 125 VDC panel AA2, with an alternate supply from 125 VDC panel BB2. On a loss of power (panel AA2), both channels will automatically transfer to the alternate supply.

REFERENCE: 2.4.2.3.1, Section 4.8
2.3.2.21, 9-3-1/E-1, E-2

Tier: 2
Group: 2
K/A System: 263000
K/A Number: K4.01
K/A Value: 3.1
Cognitive Level: 1
Bank/Mod/New: Bank

Distracter a: See justification above.
Distracter c: See justification above.
Distracter d: See justification above.

Proposed references to be provided to applicants during the examination: None

CS system were in a test lineup when power is lost, for instance, flow would be diverted back to the suppression pool because the valves in the test flow path will be de-energized in the open position.

LO-08f 3. DC Power

Fig 4

Power for the pneumatic actuated solenoid valves for all SRVs is normally supplied from Panel AA2, with backup power available from Panel BB2. On a loss of power from AA2, power will automatically transfer to BB2.

Loss of 125V DC will affect ADS logic. Loss of 125V DC Panel AA2 results in loss of power to logic A as it has no backup power. Loss of 125V DC Panel BB2 results in loss of power to logic B, and part of the logic power will automatically transfer to 125V DC panel AA2 power. Logic B will not function with a loss of BB2, however, because the other part of the logic (the RPV level relays) is also powered from BB2 and has no backup power. Loss of one power supply will not prevent ADS from actuating as only one logic is required. Loss of both 125V DC power supplies will prevent actuation of the SRV for any reason other than high reactor pressure.

Fig 8

Both channels of LLS logic are normally powered from AA2 through the normal power supply fuses for RV-71D for logic A and RV-71F for logic B. On loss of power to AA2 , both LLS logic channels will automatically transfer to the alternate supply from BB2.

Loss of power supplies to the SRVs from both AA2 and BB2 will not, however, cause a loss of the ability to actuate SRVs E, F, and G. These valves are associated with the Alternate Shutdown system, and can also be powered from the 125V DC DIV II ASD power supply, by means of an isolation switch in the ASD Room.

LO-08i 4. Main Steam system

Closure of the MSIVs may force the NPR system to control reactor pressure.

V. SYSTEM INTERRELATIONSHIPS

LO-03d
SO-04a

A. Instrument Air/Nitrogen System

Provides the motive pressure (100 psig) to operate the relief valves in the ADS, LLS, or manual modes.

LO-02a,b
SO-04b

B. 125V DC power system

Provides the power supply to the ADS logic circuitry, the LLS logic circuitry, and the pilot solenoid air supply valves to the relief valves.

SO-04c

C. Main Steam Lines

Q#	*Question Description*	*Rev #*	*Rev Date*	*Topic Area*	*Diff*
RO 78	ILT	0	2/2001	OFFGAS	

Q Type	*Response Time*	*Max Point Value*	*Passing Point Value*	*Lesson #*
M/C				COR001-16-02

Objective #	*Reference*	*K/A #*	*10CFR 55 41/43/45*
8, 10	2.3.2.24, 9-4-1/C-4 2.4.7.1, Section 7.1	271000, A3.07	41(b)(7) 41(b)(13)

K/A Text:
A3.07 – Ability to monitor automatic operations of the OFFGAS SYSTEM including: Process radiation monitoring system indications.

Question:

The plant is operating at 75% power when the following indications are received:

- 9-4-1/C-5, OFFGAS HIGH RAD alarm
- 9-4-1/C-4, OFFGAS TIMER INITIATED alarm
- K-1/A-4, OFFGAS FILTER HIGH D/P alarm
- Off-gas flow indicates 100 cfm on Recorder AR-FR-47, SJAE AIR FLOW

If the above conditions are sustained for twenty (20) minutes, which one of the following automatic actions will occur?

a. AOG-AO-901 “AOG Supply valve” closes.

b. AOG-AO-902 “AOG Return valve” closes.

c. OG-AO-254 “Offgas System Isolation valve” opens.

d. AR-AO-12 “30 Minute Holdup Pipe Drain valve” opens.

Answer:

ANSWER: b.

REFERENCE: 2.3.2.24, 9-4-1/C-4
2.4.7.1, Section 7.1

Tier: 2
Group: 2
K/A System: 271000
K/A Number: A3.07
K/A Value: 3.4
Cognitive Level: 1
Bank/Mod/New: Bank

Distracter a: AOG-AO-901 remains open.
Distracter c: OG-AO-254 closes.
Distracter d: AR-AO-12 closes.

Proposed references to be provided to applicants during the examination: None

OFFGAS TIMER INITIATED	SETPOINT (1758) Any simultaneous combination of an A and B trip due to: 1. Channel inoperable 2. Downscale at 0 mR/hr 3. Hi-Hi trip at 1.58E3 for Channel A or Channel B (Tech Spec ≤ 1 Ci/sec)	CIC RMP-RM-150A and RMP-RM-150B

1. AUTOMATIC ACTIONS

   1.1 After 14 minutes (ODAM ≤ 15 minutes) of continuous alarm condition, following valves close:

      1.1.1 OG-AO-254, OFF/GAS SYSTEM ISOLATION.

      1.1.2 AOG-AO-902, AOG RETURN.

      1.1.3 AR-AO-12, 30 MINUTE HOLDUP LINE DRAIN.

      1.1.4 OG-AO-13, OG FILTERS A AND B DRAIN.

2. OPERATOR OBSERVATION AND ACTION

   CAUTION - Operation in instability region is prohibited.

   2.1 Reduce power, as necessary, to clear alarm.

   2.2 Enter Procedure 2.4.7.1.

3. PROBABLE CAUSES

   3.1 Shutdown conditions cause coincident downscales.

   3.2 Fuel cladding failure.

4. REFERENCES

   4.1 Technical Specification LCO 3.7.5, Air Ejector Off-gas.

   4.2 Abnormal Procedure 2.4.7.1, High Off-Gas Activity or Abnormal Off-Gas Flow.

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO 79	ILT	0	2/2001	FIRE PROTECTION	

Q Type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C				

Objective #	Reference	K/A #	10CFR 55 41/43/45
	2.2.2, Section 13, Note 1 2.2.2, Att. 2, 1.2.4, 2.2, 2.3	286000, K1.06	41(b)(10)

K/A Text:

K1.06 – Knowledge of the physical connections and/or cause-effect relationships between FIRE PROTECTION SYSTEM and the following: Emergency generator rooms: Plant-Specific.

Question:

Which one of the following describes how manual initiation of carbon dioxide (CO2) into the DG-1 room is affected if DG-1 is already running following a start on a LOCA signal?

a. CO2 will be discharged into the room and flood the room because the DG HVAC is interlocked off.

b. CO2 will be discharged into the room with some of it exhausted to atmosphere because the DG HVAC continues to run.

c. CO2 will be blocked from discharging into the room but will flood the room if a fire detection initiation signal is received.

d. CO2 will be blocked from discharging into the room and will remain blocked even if a fire detection initiation signal is received.

Answer:

ANSWER: b.
Following an emergency start of DG1, manual initiation of DG-1 CO2 system will result in CO2 being exhausted to atmosphere due to DG HVAC system interlock which continues to run because the CO2 discharge signal is bypassed.

REFERENCE: 2.2.2, Section 13, Note 1; 2.2.2, Att. 2, 1.2.4, 2.2, 2.3

Tier: 2
Group: 2
K/A System: 286000
K/A Number: K1.06
K/A Value: 3.2
Cognitive Level: 1
Bank/Mod/New: New

Distracter a: DG HVAC remains running. CO2 is exhausted to atmosphere.
Distracter c: CO2 discharge will occur.
Distracter d: CO2 discharge will occur.

Proposed references to be provided to applicants during the examination: None

12.10 Check solenoid power indicating panel indicates operational (green band).

12.11 Check Annunciator C-4/F-5, DIESEL GEN 2 CO2 SYSTEM ABNORMAL, is clear.

13. MANUAL INITIATION OF DG-1(2) CO_2 SYSTEM

[ ] **NOTE 1** - Following an emergency start of DG1(2) or placement of IS/DG1A-(1B) in ISOLATE, manual initiation of DG-1(2) CO_2 System will result in CO_2 being exhausted to atmosphere due to DG HVAC System interlock with CO_2 System being bypassed.

[ ] **NOTE 2** - Manual initiation results in immediate discharge of CO_2 into affected room.

[ ] **NOTE 3** - Pneumatic release bottles are located near DG-1 Room entrance outside of room, in Boiler Room near entrance to DG Building, and on west wall of each DG Room near double doors.

13.1 Ensure all personnel are evacuated from affected room.

13.2 Actuate pneumatic release bottles per posted instructions at bottles.

13.3 Ensuring CO_2-PS-CO1(2), DG 1(2) H&V TRIP/RESET (outside room near security door in Boiler Room near entrance to DG Building), has actuated (plunger extended).

14. RESETTING DG-1 CO_2 SYSTEM

14.1 Dispatch operator to DG Building Roof to ensure all personnel are clear of area while CO_2 is being exhausted from DG-1 Room.

14.2 Make gaitronics announcement for all personnel to stand clear of DG-1 Room and DG Building Roof while CO_2 is being exhausted from DG-1 Room.

14.3 Dispatch two operators wearing SCBAs to DG-1 Room.

14.4 Depress plunger on CO_2-PS-CO1, DG 1 H&V TRIP/RESET (outside room near security door).

14.5 Place #1 DG CO_2 SYSTEM ABORT SWITCH (west wall) to ABORT.

14.6 At LCP-HV-DG-1A (mezzanine level), place SS/1-HV-DG-1C switch to RUN.

14.7 Exit room and inform Radiological Protection personnel room is being ventilated.

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO 80	ILT	0	2/2001	SEC CONT	

Q Type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C				COR001-08-02

Objective #	Reference	K/A #	10CFR 55 41/43/45
13	2.2.47 COR001-08-02	290001, A4.10	41(b)(9)

K/A Text:

A4.10 – Ability to manually operate and/or monitor in the control room: System lineups.

Question:

Which one of the following describes how the Reactor Building Ventilation System maintains the required 0.25 inches of negative water pressure in the Reactor Building during normal operation of the system (aligned per 2.2.47, "HVAC Reactor Building")?

a. At least one (1) more exhaust fan than supply fan is operated.

b. A d/p controller regulates the operating supply fans vortex damper position.

c. A d/p controller regulates the operating exhaust fans vortex damper position.

d. The capacity of the exhaust fans is greater than the capacity of the supply fans.

Answer:

ANSWER: c.

REFERENCE: 2.2.47

Tier: 2
Group: 2
K/A System: 290001
K/A Number: A4.10
K/A Value: 3.4
Cognitive Level: 1
Bank/Mod/New: Bank

Distracter a: Capacity is not used to maintain d/p.
Distracter b: D/p controller on suction dampers maintain flow through the filters.
Distracter d: Not in accordance with system design.

Proposed references to be provided to applicants during the examination: None

1.2.2 When a fan is started, its respective discharge damper, AD-1005A or AD-1005B, and common inlet damper, AD-1005C, will open and control circuit will be energized. DPIC-1013 modulates the respective supply fan vortex damper to maintain it's DP setting. This controller senses the pressure difference between the H&V unit inlet and discharge pressures to provide a constant air flow.

1.2.3 A roughing filter removes the large particles of dust and is controlled manually by operator. When filter DP is 0.5" wg, as indicated on DPIS-1015, Annunciator R-2/E-2, HVAC FILTER HIGH D/P, alarms.

1.2.4 Supply air varying between design minimum temperature of 50°F in winter and a maximum of 82°F in summer is distributed through duct work to all floors, areas, and rooms of Reactor Building. A large duct supplies 13,000 cfm air flow to drywell and torus inlet ventilation headers.

1.2.5 Temperature Controller TC-1014 on discharge side of H&V unit is set at 75°F winter, 45°F summer, and modulates TCV-1014A and TCV-1014B to control the amount of steam supplied to heating coils.

1.2.6 A freezestat, TC-1014AA, will cause Annunciator R-2/E-1, HVAC FREEZESTAT, to alarm if heating coil air discharge temperature drops to 40°F or less.

1.2.7 Refueling floor (1001') heating is controlled by TC-1051 which actuates TCV-1051 in the steam supply to heating coil RHC-R-1A. This temperature controller is set at 65°F.

1.2.8 Reactor Building 976' southeast heating is controlled by TC-1052 which modulates TCV-1052 in the steam supply line to RHC-R-1B. This temperature controller is set at 70°F.

1.2.9 All of Reactor Building air is normally exhausted by one of two main exhaust fans, EF-R-1A or EF-R-1B; however, during building high temperatures two exhaust fans may be run if needed. The flow rate from these fans is controlled by a vortex damper.

1.2.10 Reactor Building pressure is controlled by exhaust fan vortex damper(s). The vortex damper is controlled by a circuit which averages outside air pressure as sensed on all four sides of Reactor Building. Vortex damper is operated to maintain Reactor Building pressure at least 0.25" wg below outside air pressure.

Q#	*Question Description*	*Rev #*	*Rev Date*	*Topic Area*	*Diff*
RO 81	ILT	0	2/2001	CONTROL ROOM HVAC	

Q Type	*Response Time*	*Max Point Value*	*Passing Point Value*	*Lesson #*
M/C				

Objective #	*Reference*	*K/A #*	*10CFR 55 41/43/45*
	2.2.38, 1.b.v	290003, K4.01	41(b)(7)

K/A Text:

K4.01 – Knowledge CONTROL ROOM HVAC design feature(s) and/or interlocks which provide for the following: System initiations/ reconfiguration: Plant-Specific.

Question:

Automatic initiation of BF-C-1A, EMER BSTR FAN, is a result of which of the following conditions?

a. Smoke in the cable spreading room.

b. Chlorine gas sensed near the control room ventilation intake louvers.

c. High radiation sensed near the control room ventilation intake louvers.

d. Low differential pressure between the control room and control building.

Answer:

ANSWER: c.

REFERENCE: 2.2.38, 1.b.v

Tier: 2
Group: 2
K/A System: 290003
K/A Number: K4.01
K/A Value: 3.1
Cognitive Level: 1
Bank/Mod/New: New

Distracter a: Smoke in the cable spreading room will trip the ventilation dampers but will not start the fan..
Distracter b: Fan initiation results from detecting anhydrous ammonia, not chlorine.
Distracter d: Low control room to control building differential pressure is an alarm function only.

Proposed references to be provided to applicants during the examination: None

1.2.5 The Control Room System has an Emergency Bypass System consisting of a Prefilter PF-C-1A, High Efficiency Filter HEF-C-1A, Carbon Filter CF-C-1A, and EMER BSTR FAN BF-C-1A which can be supplied from either MCC-LX or MCC-TX via a manual transfer switch in the Auxiliary Relay Room. Upon high radiation, this Bypass System is energized and allows outside air to pass through it to the AC unit. During Bypass System operation, one AC unit supply fan is required to run in order to maintain positive Control Room pressure. Additionally, the exhaust booster fan is required to run to provide backpressure which prevents inlet air flow rates from exceeding the Tech Spec limit. To ensure system OPERABILITY, all components must be aligned to an OPERABLE DG. This system is designed to maintain the Control Room environment for 200 man-days under the above conditions. Testing has shown that, for maximum radiological protection during a radiological event, one of two system lineups is recommended. The first is the Bypass System lineup with the emergency booster fan, exhaust booster fan, and one supply fan operating. This is the designed emergency operating configuration. The second lineup is a lineup where no fans operate. This configuration is assured during a loss of power only if all fans are aligned to the same divisional power source.

1.2.6 Reheat Coil RHC-C-2A is installed in the duct work to the Cable Spreading Room controlled by TC-1041.

1.2.7 Reheat Coil RHC-C-1A is supplied in the duct work to the Control Room controlled by TC-1039. TC-1039 also controls the refrigeration compressors.

1.2.8 TC-1037 controls the heating steam to the heating coil of the AC unit to maintain winter heating.

1.2.9 Humidistat MC-1040 controls steam to Humidifier MCV-1040 to maintain Control Room humidity at about 45%.

1.2.10 Control Room pressure equalization Dampers HV-AD-AD1581 & HV-AD-AD1582 are used for Control Room pressurization and are normally in the open position except in the event of fire in the Cable Spreading Room.

2. INTERLOCKS AND SETPOINTS

2.1 On a high alarm signal from ventilation monitor, following actions to occur:

2.1.1 BF-C-1A, EMER BSTR FAN, starts.

NOTE - BF-C-1A starting also causes the actions in Steps 2.1.2 through 2.1.5 to occur.

2.1.2 HV-270AV, CONTROL ROOM HVAC INLET VALVE, closes.

2.1.3 HV-271AV, CONTROL ROOM HVAC EMERGENCY BYPASS SYSTEM INLET VALVE, opens.

2.1.4 EF-C-1B, TOILET EXH FAN, stops.

2.1.5 HV-272AV, CONTROL ROOM PANTRY EXHAUST FAN ISOLATION VALVE, closes.

2.2 When smoke is detected by SD-1001 (Cable Spreading Room return duct), SUPPLY FANs SF-C-1A and SF-C-1B receive trip signals and fire/smoke Dampers HV-AD-AD1544, HV-AD-AD1545, HV-AD-AD1546, HV-AD-AD1547, HV-AD-AD1581, and HV-AD-AD1582 close.

2.3 Fire/smoke Dampers HV-AD-AD1544, HV-AD-AD1545, HV-AD-AD1546, HV-AD-AD1547, HV-AD-AD1581, and HV-AD-AD1582 will close when fire or smoke is detected locally at the damper or when power to BF-C-1A, EMER BSTR FAN, is lost.

3. REFERENCES

3.1 TECHNICAL SPECIFICATION

3.1.1 LCO 3.7.4, Control Room Emergency Filter (CREF) System.

3.2 UPDATED SAFETY ANALYSIS REPORT

3.2.1 Volume IV, Section X, Subsection 10.3.6.6, Main Control Room Air Conditioning System.

3.3 DRAWINGS

3.3.1 B&R Drawing 2019, Main Control Room, Cable Spreading Room, and Computer Room HVAC.

Q#	*Question Description*	*Rev #*	*Rev Date*	*Topic Area*	*Diff*
RO 82	ILT	0	2/2001	INSTRUMENT AIR	

Q Type	*Response Time*	*Max Point Value*	*Passing Point Value*	*Lesson #*
M/C				COR001-17-01

Objective #	*Reference*	*K/A #*	*10CFR 55 41/43/45*
5, 6, 11	5.2.8, Step 4.2.3	300000, 2.4.11	41(b)(7) 41(b)(10)

K/A Text:

2.4.11 – Knowledge of abnormal condition procedures.

Question:

The plant is at 100% power when an Instrument Air pipe in the turbine building breaks.

Per 5.2.8, "Loss of Instrument Air," which one of the following actions is required if air pressure decreases to 75 psig?

a. Immediately insert a manual reactor scram.

b. Perform a rapid shutdown per 2.1.4.1, "RAPID SHUTDOWN."

c. Confirm that the scram air header pressure rises above 80 psig.

d. Ensure SA-PCV-609, SERVICE AIR SYSTEM ISOLATION, automatically opens.

Answer:

ANSWER: a.
When instrument air pressure lowers to the point that IA-MO-80 is required to be closed (77 psig), then a reactor scram is also required.

REFERENCE: 5.2.8, Step 4.2.3

Tier: 2
Group: 2
K/A System: 300000
K/A Number: 2.4.11
K/A Value: 3.4
Cognitive Level: 1
Bank/Mod/New: Bank

Distracter b: A reactor scram is required.
Distracter c: There is no procedural requirement to perform this action at this time.
Distracter d: SA-PCV-609 auto closes below 77 psig service air pressure, not instrument air pressure.

Proposed references to be provided to applicants during the examination: None

NOTE - Isolation of non-critical instrument air supply will cause a severe reduction in feedwater heating, resulting in unsafe plant operating conditions.©

4.2 If instrument air pressure drops to < 77 psig, perform following:

4.2.1 At Panel A, close IA-MO-80, NON CRIT INSTRUMENT AIR ISOLATION.

4.2.2 At Panel 9-5, place scram discharge volume ISOL TEST 29 VLV switch to ISOL.

4.2.3 At Panel 9-5, scram reactor.

4.3 If air compressor will not start (breaker closes and trips free) due to activation of undervoltage or over current device in breaker, breaker trip mechanism must be reset by performing following:

4.3.1 Press rectangular trip button on door of breaker cubicle to clear alarm and allow for breaker reclosure. Breaker can now be operated with switch (A - 480V F; B - 480V G; C - 480V B).

4.4 Check locally SA-PCV-609, SERVICE AIR SYSTEM ISOLATION, is closed <u>or</u> close SA-14, AIR RECEIVER 1A 6" OUTLET, and SA-15, AIR RECEIVER 1B OUTLET (C-882-S).

4.5 If Service Air header has been isolated by closure of SA-PCV-609 or SA-14 and SA-15, make following gaitronics announcement:

4.5.1 "Service Air header is isolated, personnel using forced air breathing equipment shall leave their work area and move to an area of clean atmosphere."

4.6 Dispatch an Operator to manually control hotwell level by performing following:

4.6.1 Close MC-37, FCV-17 INLET (T-882-N east of TEC HXs).

4.6.2 Close CM-11, LCV-2C NORMAL MAKEUP INLET (T-882-N).

4.6.3 Close MC-776, LCV-2D NORMAL DUMP INLET (T-882-N).

4.6.4 Throttle CM-16, LCV-2B, SURGE MAKEUP BYPASS, and/or MC-36, LCV-2A, SURGE DUMP BYPASS, as required to control hotwell level.

4.7 Declare DG1 and DG2 Cardox Systems inoperable due to DG HVAC dampers failing open until backup air supply is aligned. Dispatch an Operator to diesel generator rooms and perform following:

4.7.1 Close IA-V-678, DG-1 IA SUPPLY ROOT (DG-1 Room west of receivers).

4.7.2 <u>Slowly</u> open DGSA-V-37, DG-1 AC-DG-1A AND AC-DG-1C BACKUP AIR SUPPLY (DG-1 Room west of receivers).

4.7.3 Close IA-V-683, DG-2 IA SUPPLY ROOT (DG-2 Room west of receivers).

4.7.4 <u>Slowly</u> open DGSA-V-38, DG-2 AC-DG-1B AND AC-DG-1D BACKUP AIR SUPPLY (DG-2 Room west of receivers).

Q#	*Question Description*	*Rev #*	*Rev Date*	*Topic Area*	*Diff*
RO 83	ILT	0	2/2001	COMPONENT COOLING WATER	

Q Type	*Response Time*	*Max Point Value*	*Passing Point Value*	*Lesson #*
M/C				COR002-19-02

Objective #	*Reference*	*K/A #*	*10CFR 55 41/43/45*
2, 4, 5, 6	5.2.4	400000, K2.01	41(b)(7) 41(b)(8)

K/A Text:

K2.01 – Knowledge of electrical power supplies to the following: CCW pumps.

Question:

The plant is at 85% power with REC pumps "A", "B" and "C" operating. REC pump control switches are positioned as follows:

- "A" REC pump STANDBY
- "B" REC pump NORMAL
- "C" REC pump STANDBY
- "D" REC pump NORMAL

An operator mistakenly de-energizes MCC-K and ten (10) seconds later re-energizes MCC-K.

Twenty (20) seconds after MCC-K is re-energized, which one of the following will restore REC cooling with three (3) REC pumps in operation?

a. Manually start two (2) REC pumps only ("A," "B" and/or "D").

b. Manually start "A" <u>**or**</u> "B" REC pumps and verify REC pump "D" automatically starts.

c. Manually start two (2) REC pumps only ("A," "B" and/or "D") and then open the non-critical header supply, drywell supply isolation, HX outlet, and augmented radwaste supply.

d. Manually start "A" <u>**or**</u> "B" REC pumps, verify REC pump "D" automatically starts, and then open the non-critical header supply, drywell supply isolation, HX outlet, and augmented radwaste supply.

Answer:

ANSWER: a.
No pumps auto start. If two of "A," "B" and or "D" pumps are started within 40 seconds of the pump trips, no header isolation valves close.

REFERENCE: 5.2.4

Tier: 2
Group: 2
K/A System: 400000
K/A Number: K2.01
K/A Value: 2.9
Cognitive Level: 2
Bank/Mod/New: Bank

Distracter b: See justification above.
Distracter c: See justification above.
Distracter d: See justification above.

Proposed references to be provided to applicants during the examination: None

COOPER NUCLEAR STATION OPERATIONS MANUAL EMERGENCY PROCEDURE 5.2.4 LOSS OF ALL REACTOR EQUIPMENT COOLING (REC) WATER	CLASS: REFERENCE USE ⊛ EFFECTIVE: 4/2/00 APPROVAL: SORC OWNER: D. W. BREMER DEPARTMENT: OPS

1. SYMPTOMS

   1.1 Annunciator M-1/A-1, REC SYSTEM LOW PRESSURE, alarms.

   1.2 Annunciator M-1/A-3, REC SURGE TANK LOW LEVEL, alarms.

   1.3 Drywell temperature and pressure are rising.

   1.4 The temperature of equipment cooled by REC is rising.

   1.5 Low REC flow alarms on VBD-M.

   1.6 Pump failure alarms on VBD-M.

   1.7 Low REC System pressure.

2. AUTOMATIC ACTIONS

   > **CAUTION** - If pumps trip on loss of power and normal power is restored prior to emergency power energizing 4160V Bus 1F and 1G, REC pumps will not automatically start.

   2.1 REC pumps selected to standby will automatically start 20 seconds after 4160V Bus 1F and 1G are energized by emergency power.

   2.2 Following valves close when REC header pressure drops below specified pressure and a 40 second time delay has timed out:

      2.2.1 REC-MO-700, NON-CRITICAL HEADER SUPPLY (61.2 psig).

      2.2.2 REC-MO-702, DRYWELL SUPPLY ISOLATION (61.2 psig).

      2.2.3 REC-MO-712, HX A OUTLET (62.4 psig).

      2.2.4 REC-MO-713, HX B OUTLET (60.2 psig).

      2.2.5 REC-MO-1329, AUGMENTED RADWASTE SUPPLY (61.2 psig).

3. IMMEDIATE OPERATOR ACTIONS

   3.1 Ensure REC-MO-700, REC-MO-702, REC-MO-712, REC-MO-713, and REC-MO-1329 have closed.

   3.2 Immediately attempt to restore REC System by starting available pumps or isolating leak if possible.

   3.3 If annunciators M-1/A-1, REC SYSTEM LOW PRESSURE, and M-1/A-3, REC SURGE TANK LOW LEVEL, alarm simultaneously, perform following:

      3.3.1 Shut down all operating REC pumps.

      3.3.2 Close REC-MO-709, DRYWELL RETURN ISOLATION.

      3.3.3 Scram reactor.

Q#	*Question Description*	*Rev #*	*Rev Date*	*Topic Area*	*Diff*
RO 84	ILT	0	2/2001	TIP	

Q Type	*Response Time*	*Max Point Value*	*Passing Point Value*	*Lesson #*
M/C				COR002-31-02

Objective #	*Reference*	*K/A #*	*10CFR 55 41/43/45*
9	2.1.22, COR002-31-02	215001, A4.03	41(b)(9)

K/A Text:
A4.03 – Ability to manually operate and/or monitor in the control room: Isolation valves.

Question:

Which one of the following describes the design response of a TIP detector that is in the reactor core when a Group 2 and a Group 6 isolation signal is received?

a. **Group 2** Isolation will cause the TIP to withdraw. **Group 6** closes the ball valve.

b. **Group 6** Isolation will cause the TIP to withdraw. **Group 6** closes the ball valve.

c. **Group 2** <u>or</u> **Group 6** Isolation will cause the TIP to withdraw and close the ball valve.

d. **Group 2** Isolation will cause the TIP to withdraw. Ball valve closure is initiated by the withdraw signal.

Answer:

ANSWER: d.
The group 2 isolation will cause a group 6 isolation, however, a group 6 isolation has no effect on TIPs. Ball valve closure is automatic when the withdraw signal is initiated.

REFERENCE: 2.1.22, COR002-31-02

Tier: 2
Group: 3
K/A System: 215001
K/A Number: A4.03
K/A Value: 3.3
Cognitive Level: 1
Bank/Mod/New: Bank

Distracter a: See justification above.
Distracter b: See justification above.
Distracter d: See justification above.

Proposed references to be provided to applicants during the examination: None

LO-09g 3. MCC-S feeds the following via a lighting panel:

a. Drive Mechanisms

b. Indexers
c. Ball valves

LO-14b Loss of this power supply will cause the drive mechanisms and indexers to remain in whatever position they are in. The ball valves will close.

LO-09c,11b D. Nitrogen Purging

1. Nitrogen is supplied from the torus inerting makeup supply line to maintain the relative humidity in the drive control units and indexing mechanisms to minimize corrosion.

LO-14f 2. The Instrument Air system provides a backup to the Nitrogen Supply system. On a loss of the Nitrogen Supply system, a check valve will automatically cause the Purge system to shift to the Air system.

LO-09e E. Primary Containment Isolation System (PCIS)

LO-11a,14d

LO-14g,h

LO-16a,b

Upon receipt of a Group II Isolation signal, as initiated by ≤ +1.84 psig drywell pressure or
≥ +3.0 in. vessel water level, or loss or malfunction of the PCIS system, the following actions occur:

1. Any TIP detector not in its shield chamber is transferred automatically to the "manual reverse" mode of operation, as a result of the relay logic in the drive control unit;

2. When the detector is in its shield chamber as indicated by the limit switch, PCIS deenergizes the 125 VDC power to the Ball Valve Solenoid causing it to close. This will happen regardless of the Ball Valve switch position.

3. The indexer purge supply solenoid valve closes.

LO-09j F. Area Radiation Monitoring System

An Area Radiation Monitor is mounted in the TIP room adjacent to the shield chambers on the South wall. This detector has a range of 1000 mr/hr. Another monitor is located in the TIP tent to monitor the drive mechanisms. It has a range up to 100 mr/hr.

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO 85	ILT	0	2/2001	FUEL POOL COOLING	

Q Type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C				

Objective #	Reference	K/A #	10CFR 55 41/43/45
9	2.4.8.6	233000, K1.02	41(b)(9)

K/A Text:

K1.02 – Knowledge of the physical connections and/or cause-effect relationships between FUEL POOL COOLING AND CLEAN-UP and the following: Residual heat removal system: Plant-Specific.

Question:

The plant is in a refueling outage. Conditions are:

- Time since shutdown is 25 hours
- "B" Fuel Pool Cooling pump is out of service

The "A" Fuel Pool Cooling pump trips and **CANNOT** be started. The fuel pool temperature is currently at 110°F.

How much time is available to re-start a fuel pool cooling pump before boiling starts in the fuel pool?

a. 3 hours

b. 4 hours

c. 5 hours

d. 6 hours

Answer:

ANSWER: c
Using the Time to Boiling curves from 2.4.8.6, with an initial pool temperature of 90 degrees, boiling will be reached in just over 5 hours.

REFERENCE: 2.4.8.6 Att. 4 and 5

Tier: 2
Group: 3
K/A System: 233000
K/A Number: K1.02
K/A Value: 2.9
Cognitive Level: 3
Bank/Mod/New: New

Distracter a, b, d: Using the Time to Boiling curves from 2.4.8.6, with an initial pool temperature of 90 degrees, boiling will be reached in just over 5 hours.

Proposed references to be provided to applicants during the examination:
Time to boil curves from 2.4.8.6 (Att. 4 and Att. 5)

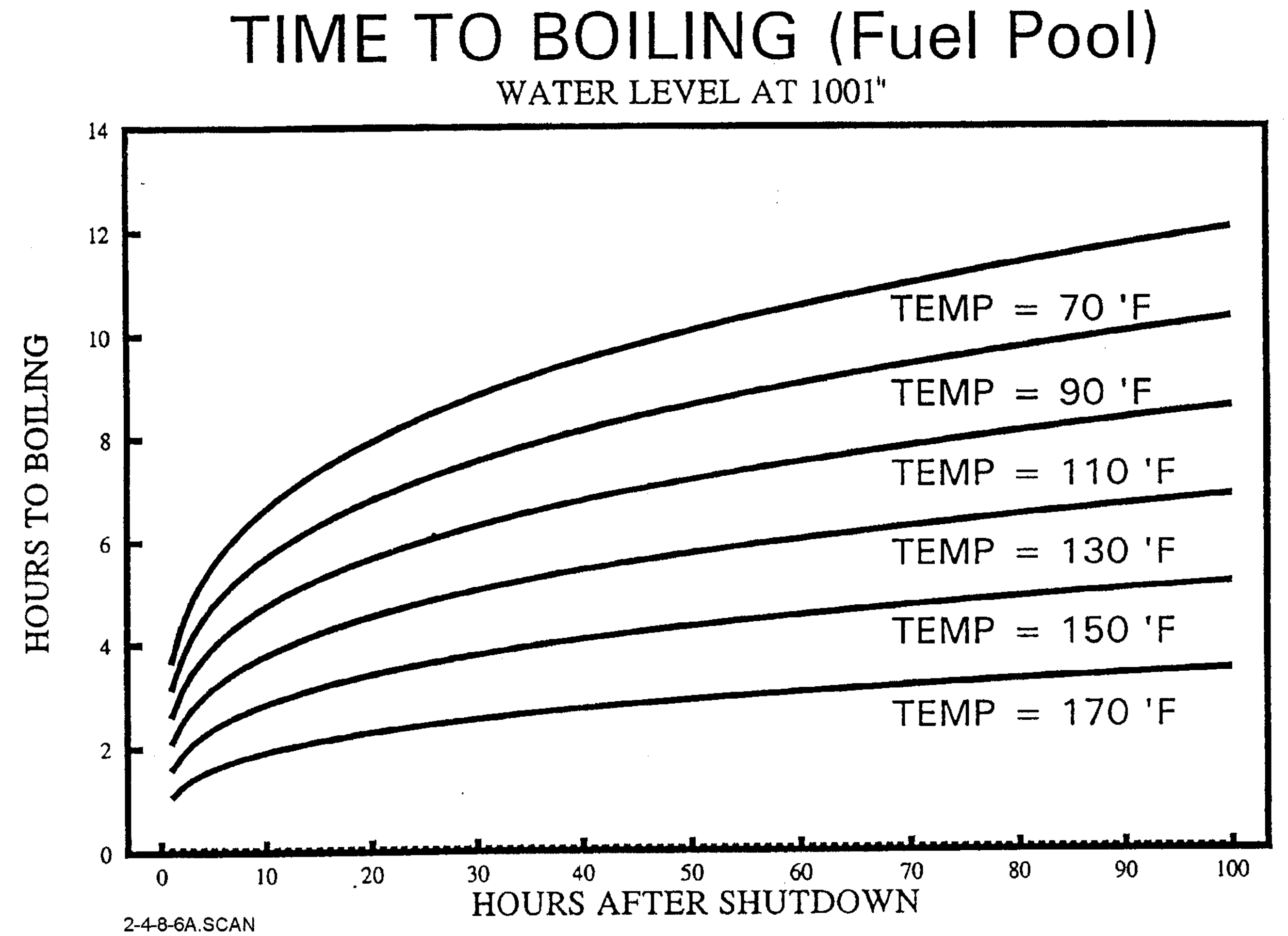
TIME TO BOILING (Fuel Pool)
WATER LEVEL AT 1001"
HOURS TO BOILING
0
2
4
6
8
10
12
14
HOURS AFTER SHUTDOWN
0
10
20
30
40
50
60
70
80
90
100
TEMP = 70 'F
TEMP = 90 'F
TEMP = 110 'F
TEMP = 130 'F
TEMP = 150 'F
TEMP = 170 'F
2-4-8-6A.SCAN

Figure 1

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO 86	ILT	0	2/28/01	PLANT HVAC	

Q Type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C				COR001-08-02

Objective #	Reference	K/A #	10CFR 55 41/43/45
11	2.2.47, 2.3.2.19	288000, A3.01	41(b)(7) 41(b)(9)

K/A Text:

A3.01 – Ability to monitor automatic operations of the PLANT VENTILATION SYSTEMS including: Isolation/initiation signals.

Question:

The unit is at 100% power with the following Reactor Building Ventilation Lineup:

- Exhaust Fan EF-R-1A and Supply Fan SF-R-1A are in AUTO
- Exhaust Fan EF-R-1B and Supply Fan SF-R-1B are in STBY

The reactor building differential pressure controller fails causing Annunciator R-2/A-4, REACTOR BLDG HIGH PRESSURE to alarm.

Which one of the following describes the Reactor Building Ventilation system response over the next five minutes?

a. EF-R-1A and SF-R-1A will continue to run.
EF-R-1B and SF-R-1B will **NOT** start.

b. EF-R-1A and SF-R-1A will continue to run.
EF-R-1B and SF-R-1B will start.

c. EF-R-1A and SF-R-1A will trip.
EF-R-1B and SF-R-1B will **NOT** start.

d. EF-R-1A and SF-R-1A will trip.
EF-R-1B and SF-R-1B will start.

Answer:

ANSWER: c.

REFERENCE: 2.2.47, 2.3.2.19

Tier: 2
Group: 3
K/A System: 288000
K/A Number: A3.01
K/A Value: 3.8
Cognitive Level: 2
Bank/Mod/New: Bank No. 5227

Distracter a: The fans will trip on high reactor building pressure and it will block starting any fans in standby.
Distracter b: The fans will trip on high reactor building pressure and it will block starting any fans in standby.
Distracter d: The fans will trip on high reactor building pressure and it will block starting any fans in standby.

Proposed references to be provided to applicants during the examination: None

2.6 An exhaust fan will trip when control switch is in STBY or AUTO if any of following conditions are met:

2.6.1 Reactor Building pressure is above -0.15" wg or below -0.35" wg and 45 second Reactor Building high/low pressure time delay has timed out.

2.6.2 Group 6 Isolation is received.

2.7 An exhaust fan will trip when control switch is in RUN if Group 6 Isolation is received.

2.8 An exhaust booster fan will start when control switch is placed in RUN if Group 6 Isolation is reset.

2.9 An exhaust booster fan will start when control switch is placed in AUTO if following conditions are met:

2.9.1 Exhaust fan is running.

2.9.2 Reactor Building pressure is below -0.15" wg and above -0.35" wg or 45 second Reactor Building high/low pressure time delay relay has not timed out.

2.9.3 Group 6 Isolation is reset.

2.10 An exhaust booster fan will automatically start when control switch is in STBY if following conditions are met:

2.10.1 Exhaust fan is running.

2.10.2 Reactor Building pressure is below -0.15" wg and above -0.35" wg or 45 second Reactor Building high/low pressure time delay relay has not timed out.

2.10.3 Breaker for exhaust booster fan with control switch in AUTO or RUN has tripped and 15 second time delay relay has timed out.

2.10.4 Group 6 Isolation is reset.

2.11 An exhaust booster fan will trip when control switch is in STBY or AUTO if any of following conditions are met:

2.11.1 Exhaust fan is not running.

2.11.2 Reactor Building pressure is above -0.15" wg or below -0.35" wg and 45 second Reactor Building high/low pressure time delay has timed out.

2.11.3 Group 6 Isolation is received.

2.12 An exhaust booster fan will trip when control switch is in RUN if a Group 6 Isolation is received.

2.13 A supply fan will start when control switch is placed in RUN if Group 6 Isolation is reset.

2.14 A supply fan will start when control switch is placed AUTO if following conditions are met:

2.14.1 Exhaust fan is running.

2.14.2 Reactor Building pressure is below -0.15 wg and above -0.35" wg or 45 second Reactor Building high/low pressure time delay relay has not timed out.

2.14.3 Group 6 Isolation is reset.

2.15 A supply fan will automatically start when control switch is in STBY if following conditions are met:

2.15.1 Exhaust fan is running.

2.15.2 Reactor Building pressure is below -0.15" wg and above -0.35" wg or 45 second Reactor Building high/low pressure time delay relay has not timed out.

2.15.3 Breaker for supply fan with control switch in AUTO or RUN has tripped and 15 second time delay relay has timed out.

2.15.4 Group 6 Isolation is reset.

2.16 Supply fan will trip when control switch is in STBY or AUTO if any of following conditions are met:

2.16.1 Exhaust fan is not running.

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO 87	ILT	0	2/2001	RPV INTERNALS	

Q Type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C				COR002-22-02

Objective #	Reference	K/A #	10CFR 55 41/43/45
6	2.4.1.7, Section 6.2	290002, K2.03	41(b)(1) 41(b)(2)

K/A Text:

K3.03 – Knowledge of the effect that a loss or malfunction of the REACTOR VESSEL INTERNALS will have on the following: Reactor power.

Question:

While operating steady state the following indications are observed:

- Reactor power lowers
- Narrow Range reactor water level rises

- Indicated core plate d/p lowers
- Indicated core flow rises
- "A" and "B" recirculation loop flows rise

Which one of the following failures caused the above conditions?

a. One (1) of the Jet pumps has failed.

b. A shroud support access hole cover has failed.

c. One (1) recirculation pump's speed has raised to maximum.

d. Flow through a control cell (four fuel bundles) has been blocked.

Answer:

ANSWER: b.

REFERENCE: 2.4.1.7, Section 6.2

Tier: 2
Group: 3
K/A System: 290002
K/A Number: K2.03
K/A Value: 3.3
Cognitive Level: 2
Bank/Mod/New: Bank

Distracter a: Loop flows will only rise in one loop and reactor water level change would not be discernible.
Distracter c: Would not provide these indications.
Distracter d: This would lower core flow.

Proposed references to be provided to applicants during the examination: None

4.7.2 Perform a normal shutdown per Procedure 2.1.4.

5. PROBABLE CAUSES

5.1 Shroud cracking and separation.

5.2 Separation of a shroud support access cover.

5.3 RPV shroud head lift during operation.

5.4 DEH pressure controller output fails high.

5.5 RR flow control failure.

5.6 Level control malfunction.

5.7 Pressure control malfunction.

6. DISCUSSION

6.1 Shroud cracks have been identified in several BWR reactor pressure vessels. Shroud cracks may be detected during normal operation by observing reactor core, RPV, and primary system parameters that are either directly or indirectly impacted by changes in core flow. When crack separation occurs, some of the water reaching the lower plenum of the RPV is discharged through the crack and into the downcomer annulus region bypassing the dryer-separators and perhaps the core itself. The magnitude of crack separation is dependent on crack location and reactor power and flow conditions in existence when the crack develops. The higher the core flow, the greater is the force to displace (or lift) the portion of the shroud that is above the crack. The potential for the shroud to be displaced and thereby cause significant leakage flow is most likely near rated flow. However, shroud separation may be experienced at core flow as low as 60% of rated.©

6.2 A shroud support access hole cover is ~ 19" in diameter. The effect to recirculation flow and core power by the separation of a cover could be significant. If a cover should separate, a flow path which bypasses the core is established which reduces the hydraulic resistance to flow through the core. This condition would indicate an increase in total core flow but actual flow through the core would drop and cause power to drop.

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO 88	ILT	0	2/2001	CONDUCT OF OPERATIONS	

Q Type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C				SKL010-10-01

Objective #	Reference	K/A #	10CFR 55 41/43/45
A3	OI-7	2.1.1	41(b)(10)

K/A Text:

2.1.1 - Knowledge of conduct of operations requirements.

Question:

During an ATWS, the Reactor Operator is directed to perform alternate control rod insertion. The Reactor Operator will be performing the actions to insert control rods by resetting RPS and inserting a manual reactor scram.

Assume the CRS has **NOT** suspended any peer check requirements.

Which one of the following describes the peer checking requirements to perform this task?

a. Required for all steps of the task.

b. Required for all steps except for panel 9-5 actions only.

c. Required for all steps except for jumper installation only.

d. Required for the jumper installation, and is waived for all other steps.

Answer:

ANSWER: c. Jumper installation is waived in accordance with OI-7 as it is a back panel action.
Peer check will be performed by operators in the Control Room for front panel manipulations prior to manipulating controls. This verification will be consistently performed during steady state manipulations and whenever reasonably possible during abnormal and transient conditions. Immediate operator actions shall not be delayed to wait for peer check. Peer check can be suspended for specific tasks during transients by the CRS as he deems reasonable and necessary.

REFERENCE: OI-7

Tier: 3
Group: -
K/A System: Generic
K/A Number: 2.1.1
K/A Value: 3.7
Cognitive Level: 2
Bank/Mod/New: Bank

Distracter a: See justification above.
Distracter b: See justification above.
Distracter d: See justification above.

Proposed references to be provided to applicants during the examination: None

PEER CHECK

Peer check shall be performed by operators in the Control Room for front panel manipulations prior to manipulating controls. This verification will be consistently performed during steady state manipulations and whenever reasonably possible during abnormal and transient conditions. Peer check can be waived without announcement by the Board Operators as deemed reasonable and necessary for the following conditions:

1. Immediate operator actions required by abnormal and emergency procedures.
2. Actions to stabilize plant parameters during or immediately following a major transient. This would include, but not be limited to Scrams and transients causing entry into EOPs.
3. Taking action prior to an automatic action or to take action to correct a failed automatic action.

The CRS may also waive peer checks for individual tasks as deemed necessary (Ie. Board Operator waiting for peer check and the CRS deems prompt action appropriate).

Peer check shall be performed using the following sequence:

1. The operator will be at the panel where the control is to be manipulated.
2. The operator will point at or touch the control he will manipulate and verbalize the action he will perform.
3. The verifier will be near enough to actually verify that the proper control is being manipulated, (able to read the component label except as described later) and must also ensure that the action is sequentially correct to achieve the desired outcome.
4. The peer checker will verbalize any questions or concerns.
5. Each action that is announced by the operator will be acknowledged by the verifier with a verbal response, such as, "That's correct".
6. After receiving acknowledgment from the verifier the operator will then perform the intended manipulation.

The following list identifies manipulations which are exempt from the requirements of this instruction:

1. Selecting indication.
2. Temperature recorders.
3. Annunciator Acknowledgment.
4. All back panel manipulations.
5. Selecting Control Rods.

The following list identifies peer checks which may be performed at a distance. The distance is defined as a reasonable space that allows the verifier to be able to competently perform the verification. These controls have been selected due to their ease of identification.

1. Changing recirculation pump speed with the manual/auto controller.
2. Pumping drywell sumps.

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO 89	ILT	0	2/2001	CONDUCT OF OPERATIONS	

Q Type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C				

Objective #	Reference	K/A #	10CFR 55 41/43/45
	2.0.4, step 3.2.2.6	2.1.3	41(b)(10)

K/A Text:

2.1.3 - Knowledge of shift turnover practices.

Question:

Given the following watch standing information for a 12-hour shift rotation:

- The most recent watch you stood was BOP on Monday dayshift (3/5/2001)
- You are preparing to assume the BOP watch on dayshift on Wednesday (3/7/2001)

Per 2.0.4, "Relief Personnel and Shift Turnover," which one of the following describes the log entries that you are required to review PRIOR to assuming the shift?

	Dayshift Monday (0700 3/5 to 1900 3/5)	Nightshift Monday (1900 3/5 to 0700 3/6)	Dayshift Tuesday (0700 3/6 to 1900 3/6)	Nightshift Tuesday (1900 3/6 to 0700 3/7)
a.	**NOT** required	**NOT** required	**NOT** required	Must review
b.	**NOT** required	**NOT** required	Must review	Must review
c.	**NOT** required	Must review	Must review	Must review
d.	Must review	Must review	Must review	Must review

Answer:

ANSWER: c.
Review of logs for which the individual is responsible back to the entries of the last shift that the individual stood or 24 hours, whichever is longer. It is not necessary to review the logs on Monday dayshift which are entries the individual made.

REFERENCE: 2.0.4, 3.2.2.6

Tier: 3
Group: -
K/A System: Generic
K/A Number: 2.1.3
K/A Value: 3.0
Cognitive Level: 2
Bank/Mod/New: New

Distracter a: See justification above.
Distracter b: See justification above.
Distracter d: See justification above.

Proposed references to be provided to applicants during the examination: None

3. INFORMATION TRANSMITTAL

3.1 The continuous transmittal of operating information is most likely to suffer at the change of shift, particularly during times of unusual activity. Whenever practical, transient activities should be minimized until shift change is completed.

3.2 Prior to assuming the duty, the on-coming Station Operators shall:

3.2.1 Obtain from SAS and sign for vital area keys.©

3.2.2 Review and understand the following, as applicable to the duty position:

3.2.2.1 Operating procedures in progress.

3.2.2.2 Abnormal or emergency procedures.

3.2.2.3 Surveillance tests.

3.2.2.4 Any other testing in progress.

3.2.2.5 Any off-normal conditions (including equipment tagged out for maintenance, testing, repair, etc.).

3.2.2.6 Review of logs for which the individual is responsible back to the entries of the last shift that the individual stood or 24 hours, whichever is longer.

3.2.2.7 A review of any significant changes in routine operation which has occurred during the previous two shifts.

3.2.2.8 Any other pertinent information.

3.3 The Control Room Supervisor and Control Room Operators (RO, BOP, and Fifth License), shall perform a panel walkdown for equipment status and off normal conditions. They shall review and understand the following prior to assuming the duty:

3.3.1 Reactor power level, pressure, and temperature.

3.3.2 Rod line.

3.3.3 Safety system status panel.

3.3.4 Tech Spec Tracking Form Index.

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO 90	ILT	0	2/2001	TECH SPECS	

Q Type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C				INT007-05-06

Objective #	Reference	K/A #	10CFR 55 41/43/45
1, 3	TECH SPEC 3.5.1	2.1.12	41(b)(7) 43(b)(2)

K/A Text:

2.1.12 - Ability to apply technical specifications for a system.

Question:

The unit is operating at 100% power when the following Technical Specification conditions are discovered:

- March **1**, 2001 at **1200** the "A" RHR pump is declared inoperable.
- March **5**, 2001 at **0800** the HPCI system is declared inoperable.

Apply any extensions that are permitted by Technical Specifications.
Assume the inoperable equipment will **NOT** be restored to OPERABLE status.

Which one of the following describes the LATEST time and date when the unit shall be in MODE 3?

a. March 5 at 2100.

b. March 8 at 2000.

c. March 8 at 2400.

d. March 9 at 2000.

Answer:

ANSWER: b.
When HPCI is declared inoperable, entry into Condition D is required. After 72 hours (3/8 at 0800), entry into Condition G is required. The unit shall be in MODE 3 within the next 12 hours (3/8 at 2000).

REFERENCE: TECH SPEC 3.5.1

Tier: 3
Group: -
K/A System: Generic
K/A Number: 2.1.12
K/A Value: 2.9
Cognitive Level: 2
Bank/Mod/New: Modified. Changed the dates and times and deleted one of the inoperable components. This caused the answer (time and date) to change.

Distracter a: Assumes entry into Condition H and LCO 3.0.3 when HPCI is declared inoperable requiring MODE 3 within 13 hours (3/5 at 2100). This is not the latest time to be in MODE 3.
Distracter c: Assumes entry into Condition B following the 7-day allowed outage time for the first inoperable pump requiring MODE 3 in 12 hours (3/5 at 2400). This is longer than the permitted time to be in MODE 3.
Distracter d: Assumes an extension of 24 hours is applied to HPCI. The completion time extension will never apply to HPCI. This is longer than the permitted time to be in MODE 3.

Proposed references to be provided to applicants during the examination:
Provide Technical Specification 1.0, 3.0, and 3.5 .1 Do not provide the Bases.

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

3.5.1 ECCS – Operating

LCO 3.5.1 Each ECCS injection/spray subsystem and the Automatic Depressurization System (ADS) function of six safety/relief valves shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3, except high pressure coolant injection (HPCI) and ADS valves are not required to be OPERABLE with reactor steam dome pressure $\leq$ 150 psig.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One low pressure ECCS injection/spray subsystem inoperable.	A.1 Restore low pressure ECCS injection/spray subsystem to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 4.	12 hours 36 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. HPCI System inoperable.	C.1 Verify by administrative means RCIC System is OPERABLE. AND C.2 Restore HPCI System to OPERABLE status.	1 hour 14 days

(continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. HPCI System inoperable. AND One low pressure ECCS injection/spray subsystem is inoperable.	D.1 Restore HPCI System to OPERABLE status. OR D.2 Restore low pressure ECCS injection/spray subsystem to OPERABLE status.	72 hours 72 hours
E. One ADS valve inoperable.	E.1 Restore ADS valve to OPERABLE status.	14 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
F. One ADS valve inoperable. AND One low pressure ECCS injection/spray subsystem inoperable.	F.1 OR F.2	Restore ADS valve to OPERABLE status. Restore low pressure ECCS injection/spray subsystem to OPERABLE status.	72 hours 72 hours
G. Required Action and associated Completion Time of Condition C, D, E, or F not met. OR Two or more ADS valves inoperable.	G.1 AND G.2	Be in MODE 3. Reduce reactor steam dome pressure to ≤ 150 psig.	12 hours 36 hours

(continued)

H. Two or more low pressure ECCS injection/spray subsystems inoperable. OR HPCI System and one or more ADS valves inoperable.	H.1	Enter LCO 3.0.3.	Immediately

Q#	*Question Description*	*Rev #*	*Rev Date*	*Topic Area*	*Diff*
RO 91	ILT	0	2/2001	EQUIPMENT CONTROL	

Q Type	*Response Time*	*Max Point Value*	*Passing Point Value*	*Lesson #*
M/C				SKL008-01-02, 10 SKL010-01-02, A.4

Objective #	*Reference*	*K/A #*	*10CFR 55 41/43/45*
SKL008-01-02, 10 SKL010-01-02, A.4	0.31, Section 8.2	2.1.29	41(b)(10)

K/A Text:

2.1.29 - Knowledge of how to conduct and verify valve lineups.

Question:

In accordance with Administrative Procedure 0.31, "Equipment Status Control," which one of the following set of conditions permit the concurrent verification for a procedure step to be waived?

a. The valve requires the use of a ladder so that it is accessible.

b. The valve location makes egress difficult should the valve malfunction.

c. The valve is required to be locked and is locked in position by the performer.

d. The verification will result in a radiation exposure of 12 mrem to the verifier.

Answer:

ANSWER: d.

REFERENCE: 0.31, Section 8.2

Tier: 3
Group: -
K/A System: Generic
K/A Number: 2.1.29
K/A Value: 3.4
Cognitive Level: 1
Bank/Mod/New: Bank

Distracter a: Not a permitted waiver for procedure steps.
Distracter b: Not a permitted waiver for procedure steps.
Distracter c: Not a permitted waiver for procedure steps.

Proposed references to be provided to applicants during the examination: None

b. Racking out and racking in 4160V and 480V breakers.

5.1.3.4 The following are the only additional instances where Concurrent Verification may be used:

a. Opening/closing breaker.

b. Lifting/landing lead/jumper/boot.

c. Removing/installing fuse.

d. Positioning control switch.

e. Installing/removing 4160V breaker test block and extension arm.

5.1.3.5 When Concurrent Verification is used during surveillance testing, initialing/signature is required for both Performer and Verifier.

6. SYSTEM COMPONENT CHECKLIST REQUIREMENTS

NOTE - Independent/Concurrent Verification requirements may be waived by the Operations Manager if excessive radiation exposure would result. As a guideline, an exposure in excess of 10 mrem to Independently/Concurrently Verify the position of a single component would be excessive.

6.1 System Component Checklists provide guidance for the position of applicable components contained within the system. These checklists dictate position of the applicable component as follows:

6.1.1 As directed by General Operating, System Operating, and Instrumentation Operating Procedures (e.g., Procedure 2.1.1, 2.2.3, 4.6.1, etc.).

6.1.2 When a component has multiple Normal positions based on plant conditions, the Operator performing the checklist shall be fully cognizant of plant/component status and know the reason for the component's state.

6.1.3 As directed by the applicable system drawings including mechanical, electrical, or instrumentation unless the position/status conflicts with a System Component Checklist. When a drawing conflicts with a System Component Checklist, the System Component Checklist position shall be used.

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO 92	ILT	0	2/2001	RECIRCULATION	

Q Type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C				COR002-22-02

Objective #	Reference	K/A #	10CFR 55 41/43/45
	2.2.68, step 2.1.2 2.3.2.26, step 2.4	2.1.32, 3.4	41(b)(10)

K/A Text:

2.1.32 - Ability to explain and apply system limits and precautions.

Question:

The plant is at 100% power. Annunciator 9-4-3 / C-3, RECIRC A PUMP MOTOR HI VIBRATION, alarms. Initially one vibration point is in the ALERT status. An operator is dispatched to investigate. NO other actions have been taken. One (1) minute following receipt of the high vibration annunciator, RONAN CRTs display the following information:

- (1831) RECIRC A LOWER MOTOR VIBRATION ALERT
- (1832) RECIRC A UPPER MOTOR VELOCITY DANGER
- (1825) RECIRC A PUMP VIBRATION DANGER

Which one of the following describes the required action?

a. Perform a rapid shutdown per 2.1.4.1, 'Rapid Shutdown."

b. Trip Recirc Pump A and enter 2.4.2.2.1, "Trip of Reactor Recirculation Pumps."

c. Reduce Recirc Pump A and B speeds to 45% per 2.1.10, "Station Power Changes."

d. Perform an emergency shutdown per 2.1.5, "Emergency Shutdown and Scram Response."

Answer:

ANSWER: b.
If two or more RR pump/motor vibration monitors for a given pump exceed the danger setpoint, the affected recirc pump must be tripped.

REFERENCE: 2.2.68, step 2.1.2
2.3.2.26, step 2.4

Tier: 3
Group: -
K/A System: Generic
K/A Number: 2.1.32
K/A Value: 3.4
Cognitive Level: 2
Bank/Mod/New: New

Distracter a: A rapid shutdown is not required. The recirc pump must be tripped, then single loop operation entered.
Distracter c: The recirc pump must be tripped.
Distracter d: An emergency shutdown (scram) is not required. The recirc pump must be tripped, then single loop operation entered.

Proposed references to be provided to applicants during the examination: None

REVISION VERIFICATION: (initial use + every 7 days)

REV.	DATE	CHANGES
53	8/23/00	Revised description of DC emergency lube oil pump operation to be more accurate.
54	see above	Updated pump operating parameters.

1. PURPOSE

This procedure provides instructions for Operations personnel to start up the Reactor Recirculation (RR) System and place it in service.

2. PRECAUTIONS AND LIMITATIONS

2.1 GENERAL PRECAUTIONS

[ ] 2.1.1 Do not exceed RRMG Set bearing oil temperatures of 194°F.

[ ] **NOTE** - Vibration monitoring system will cause an alert annunciator to alarm prior to danger alarm.

[ ] 2.1.2 If two or more RR pump/motor vibration monitors for a given pump exceed danger setpoint, trip affected pump.

[ ] 2.1.3 To prevent cold water stratification when an RR pump has tripped, attempt to restart pump(s) as soon as allowable.

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO 93	ILT	0	2/2001	SURVEILLANCE PROCEDURES	

Q Type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C				

Objective #	Reference	K/A #	10CFR 55 41/43/45
	6.SC.201, Section 6 and Att. 1	2.2.12, 3.0	41(b)(10)

K/A Text:

2.2.12 – Knowledge of surveillance procedures.

Question:

The plant is at 100% power. Surveillance 6.SC.201, Secondary Containment (Reactor Building H&V) Valve Operability Test, is in progress. The first damper (HV-AO-257) is tested satisfactorily. The second damper (HV-AO-259) is tested as follows:

- First stroke time for HV-AO-259 is 8.8 seconds.
- HV-AO-259 is tested again and the retest stroke time is 7.1 seconds.

Which one of the following describes damper HV-AO-259 status (INOPERABLE <u>or</u> OPERABLE) and why?

a. INOPERABLE. Both stroke times were excessive.

b. INOPERABLE. Only the first stroke time was excessive.

c. OPERABLE. Both stroke times were acceptable.

d. OPERABLE. Only the retest stroke time was acceptable.

Answer:

ANSWER: d.
Per Step 6.4, if step 6.2 is not satisfied, immediately retest the valve and perform step 6.5 (so the valve can be retested). The retest time of 7.1 seconds is within the IST RETEST STROKE TIME limit. Per step 6.5, which is referenced from step 6.4, if the IST RETEST STROKE TIME is met, then the apparent cause of initial test failure is documented on the Discrepancy Sheet and the damper is OPERABLE.

REFERENCE: 6.SC.201, Section 6 and Att. 1

Tier: 3
Group: -
K/A System: Generic
K/A Number: 2.2.12
K/A Value: 3.0
Cognitive Level: 2
Bank/Mod/New: New

Distracter a: The retest stroke time was within the IST RETEST STROKE LIMIT, therefore the damper is still OPERABLE.
Distracter b: Because the retest stroke time was within the IST RETEST STROKE LIMIT, the damper is still OPERABLE.
Distracter c: The initial stroke time was excessive (above the operability limit). Only the retest stroke time was acceptable (wiithin the IST RETEST STROKE LIMIT).

Proposed references to be provided to applicants during the examination: **6.SC.201; Section 6 (all) and Attachment 1 (all)**

6. ACCEPTANCE CRITERIA

6.1 [SR 3.6.4.2.2] STROKE TIME recorded in shaded blocks on Attachment 1 are within OPERABILITY LIMIT.

6.2 [IST] STROKE TIME recorded in shaded blocks of Attachment 1 are within IST RETEST LIMIT.

6.3 [IST] RETEST STROKE TIME recorded in shaded blocks of Attachment 1 are within IST RETEST LIMIT.

6.4 If Step 6.2 is <u>not</u> satisfied, immediately retest valve <u>and</u> perform Step 6.5.

6.5 If retest was performed, perform following:

[ ] 6.5.1 If Step 6.3 is satisfied, document apparent cause of initial test failure on Discrepancy Sheet.

[ ] 6.5.2 If Step 6.3 is <u>not</u> satisfied, contact Engineering and initiate PIR for Engineering to evaluate the data within 96 hours to verify the new stroke time represents acceptable valve operation.

1. Record Stopwatch Calibration Due Date: ___________

2. Record Stopwatch Identification Number: ___________

3. Mark STROKE TIME N/A for valves not tested.

VALVE NUMBER	CLOSING STROKE TIME seconds	RETEST STROKE TIME seconds	IST RETEST LIMIT seconds	OPERABILITY LIMIT seconds
HV-AO-257			4.2 to 12.0	≤ 12
HV-AO-259			2.4 to 7.2	≤ 9
HV-AO-261			2.1 to 6.3	≤ 8
HV-MO-272			49.6 to 67.0	≤ 79
HV-MO-258			46.3 to 60	≤ 60
HV-MO-260			48.2 to 60	≤ 60

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO 94	ILT	0	2/2001	SAFETY LIMITS	

Q Type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C				

Objective #	Reference	K/A #	10CFR 55 41/43/45
	TS 2.0, 2.1.1.2	2.2.22	41(b)(10)

K/A Text:

2.2.22 – Knowledge of limiting conditions for operation and safety limits.

Question:

The plant is at 75% power following an inadvertent reactivity addition (cold water). The cause of the reactivity event has been identified and is being corrected. When checking thermal limits, MCPR is noted at 1.07.

Which one of the following is a consequence for the conditions above?

a. Transition boiling was experienced for several fuel assemblies in the reactor core.

b. Pellet-cladding interaction exceeded 1% strain for several fuel assemblies in the reactor core.

c. All control rods must be inserted and permission received from the commission before startup.

d. Thermal power must be derated 10% and permission received from the plant manager before raising power.

Answer:

ANSWER: c.
The MCPR safety limit has been exceeded. All control rods must be inserted within 2 hours. Per the code of federal regulations the NRC must authorize restart when a safety limit has been exceeded.

REFERENCE: TS 2.0, 2.1.1.2

Tier: 3
Group: -
K/A System: Generic
K/A Number: 2.2.22
K/A Value: 3.4
Cognitive Level: 1
Bank/Mod/New: New

Distracter a: The threshold for transition boiling has not been achieved.
Distracter b: This is a consequence of exceeding the LHGR thermal limit (MFLDP).
Distracter d: All control rods must be inserted within 2 hours.

Proposed references to be provided to applicants during the examination: None

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 785 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 25% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.08 for two recirculation loop operation or ≥ 1.09 for single recirculation loop operation.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1337 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO 95	ILT	0	2/2001	SAFETY LIMITS	

Q Type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C				

Objective #	Reference	K/A #	10CFR 55 41/43/45
	10.13, Att. 1	2.2.34	41(b)(5)

K/A Text:

2.2.34 - Knowledge of the process for determining the internal and external effects on core reactivity.

Question:

A reactor startup will be performed with Groups 1 and 2 control rods being withdrawn first. The initial reading for each SRM is 200 counts per second (cps).

Per 10.13, "Control Rod Sequence And Movement Control," which one of the following describes the criteria used to determine when the Estimated Critical Position (ECP) is to be compared to actual core conditions by Reactor Engineering?

a. When any SRM count rate has doubled five times.

b. When any SRM indication is 2000 counts per second.

c. After withdrawing the last control rod in Group 1 to position 48.

d. After withdrawing the last control rod in Group 2 to position 48.

Answer:

ANSWER: b.
Prior to initial control rod withdrawal the initial SRM readings are recorded. Then the continuous withdraw limits, which are 10 times the initial readings are calculated. This corresponds to approximately three doublings. When any SRM reaches 10 times its initial value, the ECP will be compared to core conditions.

REFERENCE: 10.13, Att. 1

Tier: 3
Group: -
K/A System: Generic
K/A Number: 2.2.34
K/A Value: 2.8
Cognitive Level: 1
Bank/Mod/New: New

Distracter a: Three doubles. Five doubles indicates when reactor criticality is expected.
Distracter c: Based on SRM count rate not rod withdrawal status.
Distracter d: Based on SRM count rate not rod withdrawal status. Reactor should be critical with Group 2 control rods.

Proposed references to be provided to applicants during the examination: None

3. APPROACH TO CRITICAL

3.1 GENERAL GUIDELINES

3.1.1 When high xenon conditions exist, extra caution should be taken when withdrawing peripheral control rods. These rods may have higher rod worth than during low or xenon-free startups.

3.1.2 The first control rods in each Rod Worth Minimizer (RWM) group should be treated with extra caution since they will be the highest worth rods of the rods in that group.

3.1.3 With reactor power less than the Low Power Setpoint (LPSP), a control rod which is bypassed shall not be withdrawn outside the restraints of Banked Position Withdrawal Sequence (BPWS). This is required to preclude operation outside the assumptions of the Control Rod Drop Accident (CRDA) analysis.

3.1.4 Control rod coupling shall be verified and documented each time a control rod is withdrawn to Position 48.

3.1.5 Criticality should be expected at any time. Notch control is appropriate when SRM period response becomes significant.

3.2 INSTRUCTIONS

3.2.1 Prior to the start of subcritical control rod withdrawals, perform the following:

3.2.1.1 Obtain an approved copy of the Control Rod Sequence Package for startup from Reactor Engineering.

3.2.1.2 Obtain Estimated Critical Position (ECP) and ±1% ECP band from a Reactor Engineer and record on Attachment 1.

3.2.1.3 Record initial SRM readings on Attachment 1.

3.2.1.4 Calculate continuous withdrawal SRM count rate limits (10 times the initial SRM readings) and record on Attachment 1.

ESTIMATED CRITICAL POSITION:

	Group	Rod	Notch
-1%Δk/k			
ECP			
+1%Δk/k			

Performed By: ______________________________ Date: __________

SRM READINGS:

SRM	A	B	C	D
INITIAL SRM READING				
CONTINUOUS W/D LIMIT (10 x INITIAL)				

Performed By: ______________________________ Date: __________

PRESENT CRITICALITY DATA:

DATE/TIME	ROD/POSITION	TEMPERATURE	PERIOD*	SEQUENCE/GROUP

* Period = 1.443 x Doubling Time.

[ ] RPIS display hard copy edit attached (Critical Control Rod Pattern).

Performed By: ______________________________ Date: __________

RECORDS

Attachment 1 is sent to CNS Records (quality record upon Performed By signature).

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO 96	ILT	0	2/2001	RADIATION PROTECTION	

Q Type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C				

Objective #	Reference	K/A #	10CFR 55 41/43/45
	9.ALARA.1, Section 6.2.1.3	2.3.4	41(b)(12)

K/A Text:

2.3.4 – Knowledge of radiation exposure limits and contamination control / including permissible levels in excess of those authorized.

Question:

A station operator has an accumulated TEDE of 1.5 rem for the year. Because of dose projections for assigned work in the outage, the operator is expected to receive an **additional** TEDE of 1.8 rem.

Per 9.ALARA.1, "Personnel Dosimetry and Occupational Radiation Exposure Program," which one of the following describes the authorization required for the worker to receive the expected dose?

a. Plant Manager.

b. Outage Manager.

c. Radiological Manager.

d. Site Vice President – Nuclear.

Answer:

ANSWER: d.
Approvals are required by the Radiological Manager above 2000 mrem and by the Site Vice President – Nuclear above 3000 mrem.

REFERENCE: 9.ALARA.1, Section 6.2.1.3

Tier: 3
Group: -
K/A System: Generic
K/A Number: 2.3.4
K/A Value: 2.5
Cognitive Level: 2
Bank/Mod/New: Modified. The previous question asked the approval for an accumulated TEDE of 2.4 rem. The question was changed to 1.5 rem TEDE and an additional 1.8 rem TEDE. The applicant must calculate the total TEDE of 3.3 rem. This changes the answer from the Radiological Manager to the Site Vice President – Nuclear. Since the answer changes, no distracters were changed since this meets the requirement for significantly modified.

Distracter a, b, c: See explanation above.

Proposed references to be provided to applicants during the examination: None

6. OCCUPATIONAL RADIATION EXPOSURE PROGRAM

6.1 MAXIMUM ALLOWABLE NON PSE OCCUPATIONAL DOSE LIMITS (10CFR20.1201)

6.1.1 An annual limit which is the more limiting of:

6.1.1.1 5 rem TEDE.

6.1.1.2 50 rem TODE to any individual organ or tissue other than the lens of the eye.

6.1.2 An annual limit of 15 rem LDE to the lens of the eye.

6.1.3 An annual limit of 50 rem SDE to the skin or to any extremity.

6.2 ADMINISTRATIVE GUIDELINES

6.2.1 TEDE, received at CNS (on-site), to any individual shall be controlled in any calendar year as follows:

6.2.1.1 Authorization to exceed 1,000 mrem on-site requires written approval of the individual's Department Supervisor, the ALARA Supervisor, and shall be documented on the CNS RP-9. Prior to authorization, Radiological Protection shall verify the requirements in Step 6.8.2 have been met. Non-CNS dose (off-site) determinations shall be based on estimated, recorded, or calculated dose. CNS external dose determinations may be based on TLD and DRD readings. CNS internal dose determinations may be based on bioassay data, DAC-hour data, and ALI data.

6.2.1.2 Authorization to exceed 2,000 mrem on-site requires written approval of individual's Department Manager, ALARA Supervisor, and the Radiological Manager which shall be documented on the CNSRP-9. Prior to authorization, Radiological Protection shall verify the requirements in Step 6.8.2 have been met. Non-CNS dose (off-site) determinations shall be based on estimated, recorded, or calculated dose, of which non-CNS (off-site) estimated dose shall not exceed 1,500 mrem. CNS external dose determinations may be based on TLD and DRD readings. CNS internal dose determinations may be based on bioassay data, DAC-hour data, and ALI data.

6.2.1.3 Authorization to exceed 3,000 mrem (on-site) requires written approval by the Site Vice President-Nuclear. In no case shall an individual's cumulative dose be allowed to exceed 4,000 mrem. Non-CNS dose (off-site) determinations shall be based on estimated, recorded, or calculated dose, of which non-CNS (off-site) estimated dose shall not exceed 1,000 mrem. CNS external dose (on-site) determinations may be based on TLD and DRD readings. CNS internal dose determinations may be based on bioassay data, DAC-hour data, and ALI data. Prior to authorization, Radiological Protection shall verify the requirements in Step 6.8.2 have been met.

6.2.2 SKIN SDE DOSE CONTROL LIMIT

6.2.2.1 45 rem/year.

6.2.3 EXTREMITIES SDE DOSE CONTROL LIMIT

6.2.3.1 45 rem/year.

6.2.4 LENS OF THE EYE LDE DOSE CONTROL LIMIT

6.2.4.1 12.5 rem/year.

6.2.5 Doses received in excess of the annual limits, including doses received during accidents, emergencies, and PSEs, shall be subtracted from the limits for PSEs that the individual may receive during the current year and during the individual's lifetime.

6.2.6 The assigned DDE and SDE shall be for the part of the body receiving the highest exposure. If the individual monitoring device (TLD and/or DRD) was not in the region of highest potential dose or TLD and/or DRD results are not available, DDE, LDE, and SDE may be assessed from surveys or other radiological measurements.

6.2.7 DAC and ALI values listed in 10CFR20, Appendix B, Table 1, may be used to determine an individual's dose.

6.2.8 NPPD shall reduce the dose that an individual may be allowed to receive in the current year by the amount of occupational dose received during the current year while employed elsewhere.

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO 97	ILT	0	2/2001	SUMPS	

Q Type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C				

Objective #	Reference	K/A #	10CFR 55 41/43/45
	2.2.27, Att. 3, 1.2.3.4	2.3.10. 2.9	41(b)(7)

K/A Text:

2.3.10 – Ability to control radiation releases.

Question:

Which one of the following is a consequence of an extremely high level in the "Z" sump?

a. The SGT discharge lines to the elevated release point can become blocked.

b. The OFFGAS discharge lines to the elevated release point can become blocked.

c. The SGT common discharge (outlet) valve will be interlocked closed until the condition is corrected.

d. The OFFGAS common discharge (outlet) valve will be interlocked closed until the condition is corrected.

Answer:

ANSWER: a.
Drains from the elevated release point (ERP) and other parts of the OFFGAS system drain to the Z sump. Excessively high water level in the Z sump can block the SGT discharge lines to the ERP.

REFERENCE: 2.2.27, Att. 3, 1.2.3.4

Tier: 3
Group: -
K/A System: Generic
K/A Number: 2.3.10
K/A Value: 2.9
Cognitive Level: 1
Bank/Mod/New: New

Distracter b: Offgas drains go to the Z sump. Offgas discharge in not affected.
Distracter c: There is no interlock with high Z sump level and the discharge valve.
Distracter d: Offgas drains go to the Z sump. There is no interlock with high Z sump level and the discharge valve.

Proposed references to be provided to applicants during the examination: None

1.2.3.3 Drains from the centrifuge hopper and conveyor areas in the Radwaste Building are directed to the waste sludge tank utilizing shielded pipe chases.

1.2.3.4 Drains from the elevated release point and other parts of the Off-Gas System drain to Z sump located at the foot of the ERP and are pumped to the Waste Collector Tank. Components needed for Z sump pump operation are essential in support of the SGT System. Excessively high water level in Z sump can potentially block the SGT discharge lines to the ERP.©

1.2.3.5 The ERP sump drain line is routed to the Waste Collector Tank using a three-way plug valve (RW-V-1308). The valve in normally lined up to the Waste Collector Tank and can also be positioned to line up to the Floor Drain Collector Tank, if necessary.

1.2.4 NON-RADIOACTIVE (NORMAL DRAINAGE SYSTEM)

1.2.4.1 Roof drains and non-radioactive area floor drains in the Turbine Building service areas are collected and discharged by gravity to the roof drain system.

1.2.4.2 Other low point drains in the Turbine Building and Diesel Generator Rooms are collected in sumps and then pumped to the roof drain system.

1.2.4.3 Additional sumps are provided in electrical manholes, two of which are located in the yard area and one in the Control Building basement. These sumps are equipped with a single pump and discharge to the roof drain system.

2. INTERLOCKS AND SETPOINTS

2.1 Drywell equipment and floor drain sump pump isolation valves isolate on:

2.1.1 Reactor low water level $\geq$ 3".

2.1.2 High drywell pressure $\leq$ 1.84 psig.

2.1.2.1 RW-AO-82.

2.1.2.2 RW-AO-83.

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO 98	ILT	0	2/2001	INSTRUMENTATION	

Q Type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C				COR002-15-02

Objective #	Reference	K/A #	10CFR 55 41/43/45
	3.18, Section 2.9	2.4.3, 3.5	41(b)(7)

K/A Text:

2.4.3 – Ability to determine post-accident instrumentation.

Question:

An RPV level instrument on the Control Room panels is marked with a black diamond. Which one of the following describes the significance of this marking?

The marking designates instrumentation:

A. calibrated for hot conditions at 1000 psig.

B. calibrated for cold conditions and depressurized.

C. qualified to the requirements of RegGuide 1.97.

D. that will be unavailable during accident conditions.

Answer:

ANSWER: c
Control Room indicators are marked with black diamonds on the panels to identify them as RegGuide 1.97 instrumentation.

REFERENCE: 3.18, Section 2.9

Tier: 3
Group: -
K/A System: Generic
K/A Number: 2.4.3
K/A Value: 3.5
Cognitive Level: 1
Bank/Mod/New: New

Distracter a: See justification above.
Distracter b: See justification above.
Distracter c: See justification above.

Proposed references to be provided to applicants during the examination: None

2.7.17 Deviations and Justifications - Any deviations from the RegGuide are listed along with supporting justification.

2.8 RegGuide 1.97 equipment is designated in the "SPECIFICATION" field in the EDF. Those items that have more than one RegGuide 1.97 function shall be listed with the most limiting Category (i.e., a variable having a C-3 and an E-1 function will be listed as "RG 1.97 E-1").

2.9 Control Room indicators for Categories 1 and 2, Types A, B, and C variables are marked with black diamonds on the panels to identify them as RegGuide 1.97 instruments.

2.10 RegGuide 1.97 instruments are classified per Procedure 3.4. In general, the following rules apply:

2.10.1 Category 1 variables are classified ESSENTIAL throughout the instrumentation channel. Instruments located in harsh environments as defined in Procedure 3.12.7 are classified EQ.

2.10.2 Category 2 variables are classified NON-ESSENTIAL. However, instruments located in harsh environments as defined in Procedure 3.12.7 are classified EQ.

2.10.2.1 Category 2 variables may have non-RegGuide 1.97 functions that require an ESSENTIAL or EQ classification.

2.10.3 Category 3 variables are classified NON-ESSENTIAL.

2.10.3.1 Category 3 variables may have non-RegGuide 1.97 functions that require an ESSENTIAL or EQ classification.

2.11 RegGuide 1.97 instruments shall be included in the CNS Calibration Program per the guidelines of Procedure 0.38.

2.12 DEFINITIONS

2.12.1 <u>Type A</u> - Those variables to be monitored that provide the primary information required to permit the Control Room Operators to take the specified manually controlled actions for which no automatic control is provided and that are required for safety systems to accomplish their safety function for design basis accident events.

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO 99	ILT	0	2/2001	SEC CONT	

Q Type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C				COR002-22-02 COR001-08-01

Objective #	Reference	K/A #	10CFR 55 41/43/45
	2.3.2.1, A-1 / A-2 2.4.3.1	2.4.10, 3.0	41(b)(10) 43(b)(2)

K/A Text:

2.4.10 – Knowledge of annunciator response procedures.

Question:

The plant is at 100% power with irradiated fuel movements in the fuel pool in preparation for a refueling outage. Annunciator A-2, RX BLDG DOOR OPEN, alarms on Panel A-1. RONAN CRTs display the following information:

- (4224) MG VENT INTAKE ROOM OUTER DOOR OPEN
- (4225) MG VENT INTAKE ROOM INNER DOOR OPEN

Which one of the following describes the concern and required actions if the conditions above **CANNOT** be immediately corrected?

a. Secondary containment is inoperable and must be corrected within 4 hours.

b. Secondary containment is inoperable and fuel movements must be terminated within 4 hours.

c. Conditions for an unmonitored release are present and an emergency shutdown must be performed.

d. Conditions for an unmonitored release are present and Chemistry must be notified to evaluate the situation.

Answer:

ANSWER: a.
For a loss of secondary containment operability, enter TS 3.6.4.1 and perform the required actions. Secondary containment must be restored to OPERABLE status within 4 hours.

REFERENCE: 2.3.2.1, A-1 / A-2
2.4.3.1

Tier: 3
Group: -
K/A System: Generic
K/A Number: 2.4.10
K/A Value: 3.0
Cognitive Level: 2
Bank/Mod/New: New

Distracter b: Fuel movements must be terminated immediately.
Distracter c: An unmonitored release is not present. This condition would be present if the open doors were for the MG VENT EXHAUST ROOM however an emergency shutdown would not be required.
Distracter d: An unmonitored release is not present. This condition would be present if the open doors were for the MG VENT EXHAUST ROOM requiring notification of Chemistry to evaluate.

Proposed references to be provided to applicants during the examination: **TS 3.6.4.1 and Bases**

	SETPOINT	CIC
RX BLDG DOOR OPEN	1. (4218) Pipe tunnel door open	1. BLDG-LMS-R104
	2. (4219) SE torus door open	2. BLDG-LMS-R6
	3. (4220) NW torus door open	3. BLDG-LMS-R7
	4. (4226) HPCI Room door open	4. BLDG-LMS-R3
	5. (4217) 903' Access both doors open	5. LS-R101 and LS-R102
	6. (4221) 958' H&V equip doors open	6. BLDG-LMS-R301 and BLDG-LMS-R302
	7. (4222) MG Vent Exh Room outer door open	7. BLDG-LMS-R408
	8. (4223) MG Vent Exh Room inner door open	8. BLDG-LMS-R409
	9. (4224) MG Vent Intake Room outer door open	9. BLDG-LMS-R406
	10. (4225) MG Vent Intake Room inner door open	10. BLDG-LMS-R407

1. AUTOMATIC ACTIONS

   1.1 None.

2. OPERATOR OBSERVATION AND ACTION

   2.1 Check associated CRT alarm messages to determine which input caused alarm.

      2.1.1 (4218) RX BLDG PIPE TUNNEL DOOR OPEN.

      2.1.2 (4219) RX BLDG SE TORUS DOOR OPEN.

      2.1.3 (4220) RX BLDG NW TORUS DOOR OPEN.

      2.1.4 (4226) RX BLDG HPCI ROOM DOOR OPEN.

      2.1.5 (4217) RX BLDG NORTH 903' ACCESS BOTH DOORS OPEN.

      2.1.6 (4221) RX BLDG 958' H&V EQUIP AIR DOORS OPEN.

      2.1.7 (4222) RX BLDG MG VENT EXH ROOM OUTER DOOR OPEN.

      2.1.8 (4223) RX BLDG MG VENT EXH ROOM INNER DOOR OPEN.

      2.1.9 (4224) RX BLDG MG VENT INTAKE ROOM OUTER DOOR OPEN.

(continued on next page)

2.1.10 (4225) RX BLDG MG VENT INTAKE ROOM INNER DOOR OPEN.

2.2 Notify SS of condition.

2.3 Enter Procedure 2.4.3.1 as dictated by plant conditions.

2.4 If both RX BLDG MG VENT EXH ROOM OUTER and INNER DOORS are open at same time <u>and</u> an RRMG Set exhaust fan is running, notify Chemistry to evaluate for an unmonitored radiological release.

3. PROBABLE CAUSE

3.1 Personnel entry for maintenance or inspection.

4. REFERENCES

4.1 Abnormal Procedure 2.4.3.1, Loss of Primary or Secondary Containment.

3. IMMEDIATE OPERATOR ACTIONS

3.1 None.

4. SUBSEQUENT OPERATOR ACTIONS

4.1 For a loss of Primary Containment OPERABILITY, enter the applicable Conditions and Required Actions for LCO 3.6.1.1.

4.2 For a loss of Primary Containment Isolation Valve OPERABILITY, enter the applicable Conditions and Required Actions for LCO 3.6.1.3.

4.3 For a loss of Primary Containment OPERABILITY, perform following:

4.3.1 Monitor drywell O_2 concentrations.

4.3.2 If Primary Containment is inerted and a breach of containment has been confirmed, make a gaitronics announcement to evacuate the Reactor Building.

4.3.2.1 Control access to the Reactor Building such that personnel must be wearing self-contained breathing apparatus to enter until an adequate oxygen atmosphere has been verified.

4.4 For a loss of Secondary Containment OPERABILITY, enter applicable Conditions and Required Actions of LCO 3.6.4.1.

4.5 For a loss of Secondary Containment Isolation Valve OPERABILITY, enter applicable Conditions and Required Actions of LCO 3.6.4.2.

4.6 For a loss of Secondary Containment OPERABILITY, perform following:

4.6.1 If loss of Secondary Containment OPERABILITY has occurred during activities on refueling floor (refueling, core alterations, objects being moved over the opened reactor vessel or fuel pool) or activities that could reduce the shutdown margin, make a gaitronics announcement to suspend those activities until Secondary Containment OPERABILITY has been restored.

5. PROBABLE CAUSE

5.1 Failure of a containment isolation line to isolate when required.

Q#	Question Description	Rev #	Rev Date	Topic Area	Diff
RO 100	ILT	0	2/2001	PMIS	

Q Type	Response Time	Max Point Value	Passing Point Value	Lesson #
M/C				

Objective #	Reference	K/A #	10CFR 55 41/43/45
COR002-17-02-11.a	COR002-17-02, PMIS	2.4.21	41(b)(7)

K/A Text:

Knowledge of the parameters and logic used to assess the status of safety functions including:
1. Reactivity Control
2. Core Cooling and heat removal
3. Reactor coolant system integrity
4. Containment conditions
5. Radioactivity release control

Question:

During a LOCA, a HPCI isolation signal is received and HPCI responds as designed. Which one of the following describes how this isolation affects the PMIS CONTAINMENT DISPLAY?

a. Group 4 box changes to RED.

b. Group 5 box changes to RED.

c. Group 4 box changes to GREEN.

d. Group 5 box changes to GREEN.

Answer:

ANSWER: a.
The group 4 box will change from green to red. Group 5 is RCIC isolation.

REFERENCE: COR002-17-02, PMIS

Tier: 3
Group: -
K/A System: Generic
K/A Number: 2.4.21
K/A Value: 3.7
Cognitive Level: 1
Bank/Mod/New: Bank

Distracter b: Group 4 not group 5.
Distracter c: Changes to red from green.
Distracter d: Group 4 not group 5. Changes to red from green.

Proposed references to be provided to applicants during the examination: None

HRL LRL. If calculation does not have enough healthy input points, then the bar chart, indicator, etc. that use the calculation are displayed in MAGENTA.

5. Not valid indicators are used by the SPDS to assist the operator in recognizing a "not valid" situation. The characters "NV" appear in MAGENTA near the affected bar chart, trend or multi parameter plot whenever the associated point fails to meet its respective validation criteria. With the exception of SRM flux, the reason a point becomes "not valid" is due to failure to pass its redundant point check or is not healthy. The SRM detector must be fully inserted for the SRM flux level to be valid.

6. In a bar chart, the current value is shown by means of an appropriate colored bar and by a digital display of the current value. When a data point is pegged high, the bar will be completely filled in and the color of the bar will change to MAGENTA because of the quality of the point driving the bar (i.e. a quality of BAD or NCAL). When a data point is pegged low, the characters "DNSC" are displayed in MAGENTA at the low end of the affected bar chart whenever the current value of the data point driving the bar reaches its engineering limit low.

7. SPDS trends will include a trend line showing the last 10 minutes of data. Multi parameter plots will generate a 20 second tail.

8. Status indicators - Status indicators are boxes on SPDS displays. Their color is determined by the status of input points to the indicator.

   Status indicators are set up to be green during normal operation. For Group Isolations, the following color convection is used:

   Green indicates normal.

   Red indicates tripped.

   Magenta indicates not healthy.

   For valve positions the following color convection is used:

   Green indicates closed.

   Red indicates open.

   Magenta indicates not healthy.

   SRV position is based on pressure switch and temperature status.

   For pumps or motors the following color convention is used:

   Green indicates OFF.

   Red indicates running.