Facility: Columbia Generating Station	Task No: SRO-0066 RO-0559
Task Title: Comply with Conduct of Operations – Complete Shift Turnover forms for Simulator Scenario	Job Performance Measure No: BA.1-2JPM
K/A Reference: 2.1.3 3.0/3.4	
Examinee:	NRC Examiner:
Facility Evaluator:	Date: 02/22/01

Method of testing:

Admin - Perform

JPM SETUP INFORMATION

Initial Conditions:	A reactor startup is in progress. The following conditions exist: Control rods are being withdrawn to bring the reactor critical. IRM C is out of service and bypassed. SGT-V-2B was just found closed with no control power. The Plant Logging System is inoperable.
Task Standard:	Shift Turnover forms must be completed with the information shown on the attached copies. Information on the attached forms is the minimum required for credit.
Required Materials:	N/A
General References:	PPM 1.3.1 rev. 56, Operating Policies, Programs, and Practices, attachments 6.8.2 and 6.8.6.
Initiating Cue:	Given the Initial Conditions and the frozen simulator, complete a Shift Turnover Sheet for the oncoming Shift. When you are finished, hand the completed turnover sheet to the examiner.
Time Critical Task:	NO
Validation Time:	10 minutes
Simulator ICs:	N/A
Malfunctions/Remote Triggers:	N/A
Overrides:	N/A
Special Setup Instructions:	This JPM is to be performed in conjunction with and prior to Dynamic Scenario #1

PERFORMANCE INFORMATION

START TIME:

Critical Step: **YES**	
Performance Step: 1	Complete the appropriate Shift Turnover Sheet using the Initial Conditions given and the frozen simulator.
CUE:	
Standard:	Shift Turnover forms must be completed with the information shown on the attached copies. Information on the attached forms is the minimum required for credit.
Comment: SAT / UNSAT	

THE EXAMINEE SHOULD HAND THE JPM TO THE EXAMINER AT THIS POINT.

JPM TERMINATION TIME: JPM START TIME: - JPM COMPLETION TIME:	______________________

VERIFICATION OF COMPLETION

JPM Number: Ba.1-2.JPM

Examinee's Name:

Examiner's Name:

Date Performed:

Facility Evaluator:

Number of Attempts:

Time to Complete:

JPM INFORMATION CARD

HAND THE STUDENT INFORMATION CARD TO THE EXAMINEE

READ TO THE EXAMINEE:
I will explain the initial conditions, which steps to simulate or discuss, and provide initiation cues. When you complete the task successfully, the objective for this job performance measure will be satisfied.

Task Standard:	Shift Turnover forms must be completed with the information shown on the attached copies. Information on the attached forms is the minimum required for credit.
Required Materials:	N/A
Safety Equipment:	N/A
General References:	PPM 1.3.1 rev. 56, Operating Policies, Programs, and Practices, attachments 6.8.2 and 6.8.6.
Time Critical Task:	NO
Initial Conditions:	A reactor startup is in progress. The following conditions exist: Control rods are being withdrawn to bring the reactor critical. IRM C is out of service and bypassed. SGT-V-2B was just found closed with no control power. The Plant Logging System is inoperable.

INITIATING CUE

Given the Initial Conditions and the frozen simulator, complete a Shift Turnover Sheet for the oncoming Shift.

When you are finished, hand the completed turnover sheet to the examiner.

INFORMATION BELOW THIS LINE NOT SHARED WITH EXAMINEE

Task Number: SRO-0066 RO-0559	Validation Time: 10 minutes
NUREG 1123 Reference: 2.1.3 3.0/3.4	Time Critical: NO
Location: Simulator	Performance Method: Perform
Prepared/Revised by: S Hutchison	Revision Date: 7/23/02

STUDENT INFORMATION

Initial Conditions: A reactor startup is in progress. The following conditions exist:
Control rods are being withdrawn to bring the reactor critical.
IRM C is out of service and bypassed.
SGT-V-2B was just found closed with no control power.
The Plant Logging System is inoperable.

INITIATING CUE

Given the Initial Conditions and the frozen simulator, complete a Shift Turnover Sheet for the oncoming Shift.

When you are finished, hand the completed turnover sheet to the examiner.

Facility: Columbia Generating Station	Task No: N/A
Task Title: USE EWD TO EXPLAIN OVERRIDE SWITCH FOR LPCS-V-5	Job Performance Measure No: BA.2JPMr0
K/A Reference: 2.1.24 2.8/3.1	
Examinee:	NRC Examiner:
Facility Evaluator:	Date:

Method of testing:

Admin – Perform (Any location with reference material.)

JPM SETUP INFORMATION

Initial Conditions: N/A

Task Standard: The task is completed successfully when the candidate explains the effects of LPCS-RMS-S22 on the operation of LPCS-V-5.

Required Materials: N/A

General References: EWD 8E-006

Initiating Cue: Using EWD 8E-006 explain the purpose of LPCS-RMS-S21 Override switch for LPCS-V-5. Include in your discussion all the expected effects this switch has on the operation of LPCS-V-5 and an explanation of how this is accomplished.

Indicate when you are finished with your explanation.

Time Critical Task: N/A

Validation Time: 10 minutes

Simulator ICs: N/A

Malfunctions/Remote Triggers: N/A

Overrides: N/A

Special Setup Instructions: N/A

PERFORMANCE INFORMATION

START TIME:

Critical Step: **YES**	
Performance Step: 1	Must indicate each of the following: LPCS-RMS-S21 contacts 7 - 8 close to energize LPCS-RLY-K22. LPCS-RLY-K22 contacts 2 – 1, 12 – 11, and LPCS-RMS-S21 contact 2 – 1 , all open. This allows LPCS-RMS-V/5 to be throttled.
CUE:	
Standard:	Credit is given when the candidate has explained the above using EWD 8E-006
NOTE:	
Comment: SAT / UNSAT	

THE EXAIMNES SHOULD ANNOUNCE THE END OF THE JPM WHEN HE HAS COMPLETED THE EXPLANATION..

JPM TERMINATION TIME:	
JPM START TIME:	
-	____________________
JPM COMPLETION TIME:	

VERIFICATION OF COMPLETION

JPM Number:	BA.2JPMr0
Examinee's Name:	
Examiner's Name:	
Date Performed:	
Facility Evaluator:	
Number of Attempts:	
Time to Complete:	

JPM INFORMATION CARD

HAND THE STUDENT INFORMATION CARD TO THE EXAMINEE

READ TO THE EXAMINEE:
I will explain the initial conditions, which steps to simulate or discuss, and provide initiation cues. When you complete the task successfully, the objective for this job performance measure will be satisfied.

Task Standard:	The task is completed successfully when the candidate explains the effects of LPCS-RMS-S22 on the operation of LPCS-V-5.
Required Materials:	N/A
Safety Equipment:	N/A
General References:	EWD 8E-006
Time Critical Task:	NO
Initial Conditions:	N/A

INITIATING CUE

Using EWD 8E-006 explain the purpose of LPCS-RMS-S21 Override switch for LPCS-V-5. Include in your discussion all the expected effects this switch has on the operation of LPCS-V-5 and an explanation of how this is accomplished.

Indicate when you are finished with your explanation.

INFORMATION BELOW THIS LINE NOT SHARED WITH EXAMINEE

Task Number: N/A
NUREG 1123 Reference: 2.1.24 2.8/3.1
Location: Any location with references.
Prepared/Revised by: S Hutchison

Validation Time: 10 minutes
Time Critical: NO

Performance Method: Perform
Revision Date: 6/26/02

STUDENT INFORMATION

Initial Conditions: N/A

INITIATING CUE

Using EWD 8E-006 explain the purpose of LPCS-RMS-S21 Override switch for LPCS-V-5. Include in your discussion all the expected effects this switch has on the operation of LPCS-V-5 and an explanation of how this is accomplished.

Indicate when you are finished with your explanation.

Facility: COLUMBIA **Scenario No.:** 1 **Op-Test No.:** 1

Examiners: ___________________________ **Operators:** ___________________________

___________________________ ___________________________

___________________________ ___________________________

Initial conditions: IC-191 (batch file NRC02.1.txt). The reactor is approaching criticality. IRM "C" is out of service and bypassed. SGT-V-2B has lost control power and is shut.

Turnover: A plant startup is in progress. The reactor is approaching criticality. The off-going shift pulled rods up through RWM group 12. The "C" IRM failed downscale 4 hours ago and the associated bypass switch is caution tagged. As the startup continues, RWCU will need to be lined up for reactor water level control. The off-going shift just found SGT-V-2B in the closed position. Upon attempting to open the valve from the control room, the valve lost control power.

Event No.	Malf. No.	Event Type*	Event Description
1.	Initiated by turnover	R (RO)	Withdraw control rods to bring the reactor critical.
2.	Initiated by turnover	N (BOP)	Establish Reactor Water Cleanup blow-down flow for Reactor Water Level control.
3.	Trigger 3	C (BOP)	Loss of REA-FN-1B resulting in a high reactor building pressure and entry into EOP Secondary Containment Control, 5.3.1.
4.	Trigger 4	I (RO)	IRM 'A' fails upscale resulting in a half scram on the 'A' side of RPS.
5.	Trigger 5	M (All)	An earthquake results in a Loss of All Offsite Power and a LOCA. **(Columbia IPE)**
6.	Initiated as part of Trigger 5	C	The Division 1 emergency bus, SM-7, locks-out resulting in a loss of power to the bus and its loads.
7.	Initiated as part of Trigger 5	C	The output breaker of the HPCS diesel generator fails to auto close requiring the operator to manually close the breaker in order to operate HPCS.
8.	Initiated as part of Trigger 5	C	The injection valve for the 'C' loop of RHR, RHR-V-42C, fails to auto open on an injection signal.

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor

Event No. 1

Description: Withdraw control rods to bring the reactor critical.

This event is initiated by the turnover sheet and is terminated when criticality is achieved.

Time	Position	Applicants Actions or Behavior
T=0 when crew assumes shift	SRO	Directs RO to continue pulling control rods to achieve criticality.
	RO	Withdraws control rods to continue startup (PPM 3.1.2): - closely monitors flux levels during rod withdrawal - verifies prior to each rod withdrawal: ➢ correct rod selected ➢ correct start/stop position - for each rod that is fully withdrawn: ➢ checks coupling integrity - initials sequence sheet (PPM 9.3.9) ➢ ensures position 48 corresponds to FULL OUT light
	BOP	Monitors plant conditions

COMMENTS: Rod pull starts at RWM group 13.

Event No. 2

Description: Establish Reactor Water Cleanup blow-down flow for Reactor Water Level control.

This event is initiated by the turnover sheet. Should the crew hold off, there is enough heat addition occurring to cause reactor water level to rise slowly with a resultant Hi Reactor Water Level alarm. This event's termination point is when blow-down flow has been established and reactor water level is returning to it's normal setpoint.

Time	Position	Applicants Actions or Behavior
	RO	Reports that Reactor Water Level is on an increasing trend
	SRO	Directs BOP to establish level control using RWCU blowdown to the condenser
	BOP	Initiates RWCU reject flow to maintain reactor water level at setpoint per PPM 3.1.2, step 5.3.9.

COMMENTS:

Event No. 3		
Description: Loss of REA-FN-1B resulting in a high reactor building pressure and entry into EOP Secondary Containment Control, 5.3.1. *This event is* ***MANUALLY initiated with*** **<u>TRIGGER 3</u>** *after the RO brings the reactor to criticality, or by direction of the lead examiner. The event endpoint occurs when Standby Gas Treatment has been started and Reactor Building pressure is returning to a negative state.*		
Time	**Position**	**Applicants Actions or Behavior**
	BOP	Reports the receipt of the secondary containment high-pressure alarm and notes that it is a possible EOP entry. Goes to the back panel (P812) to investigate the cause of the abnormal condition in secondary containment. Reports that REA-FN-1B has tripped and that Reactor Bldg. pressure is positive.
	SRO	If Reactor Bldg. pressure is positive, enters EOP 5.3.1, Secondary Containment Control, based on Reactor Bldg. pressure at or above 0” H_2O
	BOP	Refers to the annunciator response procedures (PPM 4.812.R2,9-1) ➢ attempts to start REA-FN-1A ➢ if neither reactor bldg. exhaust fan can be started: ▪ immediately secures Rx Bldg. Inlet Fan ▪ closes ROA-V-1 & 2, REA-V-1 & 2 ▪ *****starts a train of SGT to maintain Rx Bldg. pressure negative ▪ refers to PPM 2.3.5, SGT System ▪ notifies Chemistry to monitor Rx Bldg ▪ refers to ODCM 6.1.2.1 and LCS 1.3.3.1 ▪ restores Rx Bldg. ventilation as soon as possible ➢ refers to ABN-HVAC ➢ ensures Rx Bldg. pressure is maintained satisfactorily

		May send an equipment operator to investigate the loss of fan 1B and check the start of fan 1A **<u>OPS-2 CUE</u>: If asked to investigate the loss of REA-FN-1B, report that there is no apparent cause identified in your visual inspection. If requested to do pre or post fan start checks on REA-FN-1A or SGT, report that the checks are satisfactory.** ***CRITICAL TASK**
	RO	Monitors plant Continues with plant startup
	SRO	Exits EOP 5.3.1 when Rx Bldg. Pressure is restored and with shift manager permission **<u>SHIFT MANAGER CUE</u>: When asked for permission to exit the EOPs, provide permission to exit the EOPs since the entry condition has cleared and no emergency exists.**

COMMENTS: REA-FN-1A will fail to start requiring the operator to start SGT.

Event No. 4

Description: IRM 'A' fails upscale resulting in a half scram on the 'A' side of RPS.

This event is ***MANUALLY initiated by*** **TRIGGER 4** *after the crew has exited EOP 5.3.1, or at the direction of the lead examiner. The event endpoint is when IRM 'A' has been bypassed and RPS 'A' has been reset.*

Time	Position	Applicants Actions or Behavior
	RO	Reports half scram on RPS 'A' Reports that IRM 'A' has failed upscale and caused the RPS actuation Refers to the Half Scram System A annunciator response procedure: ➢ checks full core display for individual controls rods that might have scrammed ➢ directs or performs a PA announcement to stop all maintenance or surveillance testing that has the potential for generating a trip in the unaffected RPS channel ➢ when half scram has been reset, ensures scram group solenoid lights for Groups 1,2,3 and 4 are energized and that backup scram system lights have extinguished Refers to IRM annunciator response procedures: ➢ determines which IRM is upscale ➢ ranges associated IRM up to bring on scale ➢ considers bypassing the inoperable IRM channel ➢ resets the half scram ➢ informs CRS that the procedure references tech specs and license control specs for instrument operability
	SRO	When IRM 'A' is determined to be the problem, directs bypassing of IRM 'A' Refers to Tech Specs and License Control Specs for IRM 'A': ➢ TS 3.3.1.1 ➢ LCS 1.3.2.1

		➢ LCS 1.3.3.1 ➢ Determines that none of the specs are applicable with just one IRM inoperable May make PA announcement regarding stopping maintenance/ surveillance that could result in the trip of the other division of RPS Directs that the half scram be reset
	BOP	Monitors plant
COMMENTS:		

<table>
<tr><td colspan="3">Event No. 5</td></tr>
<tr><td colspan="3">Description: An earthquake results in a Loss of All Offsite Power and a LOCA.
This event is MANUALLY initiated by <u>TRIGGER 5</u> after the SRO has competed the Tech Spec determination in event 4, or by direction of the lead examiner. This event endpoint is the scenario endpoint.</td></tr>
<tr><th>Time</th><th>Position</th><th>Applicants Actions or Behavior</th></tr>
<tr><td colspan="3">SEISMIC SIM: Preset Seismic CD player on track 4 with a volume level of 0. Start CD player approx. 3 seconds <u>before initiating TRIGGER 5</u>. Allow CD to play approx. 40 seconds before securing. After securing, set volume level to –10 and randomly run 10-15 second aftershocks over the remainder of the scenario.</td></tr>
<tr><td colspan="3">CUE: As OPS1, report that you felt seismic activity in the radwaste building; there is a lot of dust in the air.</td></tr>
<tr><td></td><td>SRO/RO/BOP</td><td>Recognize/report “Operating Basis Earthquake Exceeded” alarm
Directs/performs actions of ARP:
• verifies alarm on Board L, numerous red indicators are illuminated
• initiates a reactor shutdown
• announces OBE
• Monitors control room instrumentation for evidence of increases in:
    • Drywell leakage
    • Drywell pressure
    • Drywell gaseous or particulate activity
    • Leak detection temperature changes
• Directs a walk-down of the plant by equipment operators to determine damage caused by the seismic activity.</td></tr>
</table>

	RO	Performs immediate scram actions • Places mode switch to shutdown • Reports power/pressure/level (reports level below +13”-EOP entry condition) • Reports all rods in • Inserts SRMs/IRMs
	RO/BOP	Reports high drywell pressure GT 1.68 psig, EOP entry condition
	SRO	Enters PPM 5.1.1 on low RPV level and 5.2.1 on high DW pressure: • Directs RO/BOP to verify isolation, initiations, and DG starts • *Directs RO to maintain RPV level between –161” and +54” (will give a band within these limits) with Table 1 systems. • *Directs RO/BOP spray of WW before reaching 12 psig in the WW • Directs RO/BOP to confirm RRC pumps are stopped and stop DW cooling fans in prep for DW spray • *Directs RO/BOP to spray DW when WW press exceeds 12 psig and within DSIL • *Directs the securing of containment sprays when pressure drops LE 1.68 psig in respective area. • At –50” RPV level, directs verification of expected isolations and initiations • *At –129”, if ADS timer has started, direct inhibition of ADS • *At TAF, determines that LP ECCS is available and that current trend will drop level below –192”; determines Emergency RPV depressurization is required; enters PPM 5.1.3, Emerg. RPV Depress. • *Directs RO/BOP to open 7 SRVs, ADS preferred. ***CRITICAL TASK**
	BOP	Operates SRVs to maintain reactor pressure in the prescribed band. Verifies that Main Turbine and Main Generator have tripped

		Starts RCIC to aid in pressure/level control
	BOP	Reports electric plant status: ➢ loss of the start-up and back-up transformers (loss of off-site power) ➢ lockout on SM-7 ➢ SM-8 is powered from DG #2 ➢ SW-P-1B auto starts ➢ SW-P-1A not running - trips DG1
	RO/BOP	Starts RCC-P-1C and trips RCC-P-1A (which lost power when SM-7 was lost).
	SRO	May enter ABN-ELEC-LOOP for guidance on restoration of electrical power
	RO/BOP	When directed to carry out actions of ABN-ELEC-LOOP: ➢ Ensures fire protection headers are pressurized from diesel fire pumps ➢ Directs an EO to reset local trips on CAS compressors ➢ Starts CAS compressors ➢ Ensures that Main and Feedpump Turbine DC lube oil systems are operating
	RO/BOP	Reports that expected initiations, isolations, and DG starts have occurred except that HPCS DG output breaker failed to close automatically and DG #1 was tripped due to no service water available (lockout on SM-7) *Uses RCIC, CRD, and HPCS systems to maintain RPV level *Sprays the WW when directed Confirms that RCC pumps and DW fans have been secured *Sprays the DW when directed

		*Secures WW/DW sprays when if or when LT 1.68 psig in each area *Opens 7 SRVs (ADS preferred) to emergency depressurize the RPV. ***CRITICAL TASK**
	RO/BOP	Contacts the load dispatcher (Munro Control Center) to find out status of off-site power supplies **CUE: As the load dispatcher, if asked status of off-site power supplies, inform the control room that crews are out inspecting the transmission lines; time for power return is unknown.**
	SRO	Directs restoration of RPV level to band of +13” to +54”
	RO/BOP	Controls injection systems to restore RPV level to new band.
COMMENTS:		

Event No. 6		
Description: The Division 1 emergency bus, SM-7, locks-out resulting in a loss of power to the bus and its loads. *This event is automatically initiated 5 seconds after the earthquake in Event 5 is initiated. The event termination point occurs upon completion of the applicable subsequent actions (listed below) for ABN-ELEC-SM1/SM7 or when the scenario is terminated, whichever occurs first.*		
Time	**Position**	**Applicants Actions or Behavior**
	BOP	Reports lockout on SM-7 Reports start of DG-1
	SRO	Refers to ABN-ELEC-SM1/SM7 for guidance on restoration of power and loads on SM-7 Requests work group assistance in determining problem with SM-7
	RO/BOP	When directed, carries out actions of ABN-ELEC-SM1/SM7: ➢ restores CRD Hydraulics by starting CRD-P-1B ➢ emergency trips DG-1 due to no service water cooling available ➢ restores TSW by placing TSW-P-1A in PTL and starting TSW-P-1B ➢ ensures that RCC-P-1B & 1C are operating ➢ starts SGT-FN-1B2 to maintain reactor building pressure negative ➢ ensures power is available to the selected Hotwell Level Controller ➢ places CB-B7 in PTL ➢ places CB-1/7 in PTL
COMMENTS: This event runs during the Earthquake/LOCA/LOOP (event 5)		

Event No. 7		
Description: The output breaker of the HPCS diesel generator fails to auto close requiring the operator to manually close the breaker in order to operate HPCS. *This event is self-initiated upon the receipt of a HPCS initiation signal. The event endpoint occurs when the operator manually closes the output breaker of the HPCS DG.*		
Time	**Position**	**Applicants Actions or Behavior**
	RO/BOP	Reports that the HPCS DG output breaker failed to close automatically *****Manually closes the output breaker of the HPCS DG to SM-4: ➢ takes synch selector switch to D GEN./BUS position ➢ shuts CB4 DG3 Breaker ***CRITICAL TASK**
	SRO	*****If the operator has not attempted to manually shut the DG output breaker, directs the RO/BOP to manually shut the breaker. ***CRITICAL TASK**
COMMENTS: This event runs during the Earthquake/LOCA/LOOP (event 5)		

Event No. 8		
Description: The injection valve for the 'C' loop of RHR, RHR-V-42C, fails to auto open on an injection signal. *This event is preset and will be evident when RHR has an initiation signal and reactor pressure is LT 470 psig. The event endpoint will occur when the operator manually opens RHR-V-2C with the previously mentioned conditions existing.*		
Time	**Position**	**Applicants Actions or Behavior**
	RO/BOP	*Reports the failure of RHR-V-2C to open on an injection signal Manually opens RHR-V-2C ***CRITICAL TASK**
	SRO	*If operator fails to attempt to manually open RHR-V-2C, directs RO/BOP to open the valve ***CRITICAL TASK**
COMMENTS: This event runs during the Earthquake/LOCA/LOOP (event 5) **<u>SCENARIO ENDPOINT</u> – When the reactor has been emergency depressurized and reactor water level has been returned to the band of +13" to +54", the scenario may be terminated.**		

SRO TURNOVER INFORMATION

A plant startup is in progress. The reactor is approaching criticality. The off-going shift pulled rods up through RWM group 12.

The “C” IRM failed downscale 4 hours ago and the associated bypass switch is caution tagged.

As the startup continues, RWCU will need to be lined up for reactor water level control.

The off-going shift just found SGT-V-2B in the closed position. Upon attempting to open the valve from the control room, the valve lost control power.

Facility: COLUMBIA **Scenario No.:** 2 **Op-Test No.:** 1

Examiners: ______________________ **Operators:** ______________________

______________________ ______________________

______________________ ______________________

Initial conditions: IC-194 (batch file NRC02.2.txt). The reactor is at 100% power on a beginning of life core. Diesel Generator #1 is operating at full load for its monthly operability check. RHR-V-24B is tagged out while the motor operator is being replaced.

Turnover: The plant is at 100% power. DG-1 is fully loaded for OSP-ELEC-M701 (currently at step 7.5.62). There are 20 minutes left on the one-hour diesel run. Wetwell temperature is slowly rising due to two SRVs that are leaking by. The off-going shift recommends that suppression pool cooling be initiated as soon as you take the shift. RHR-V-24B is tagged out while the motor operator is being replaced (job completion is expected in two hours).

Event No.	Malf. No.	Event Type*	Event Description
1.	Initiated by turnover	N (BOP)	Place RHR loop 'A' into the suppression pool cooling mode.
2.	Trigger 2	C (BOP)	High-pressure feedwater heater '6A' level controller fails high resulting in the trip of feedwater heater '6A'.
3.	Initiated by turnover	R (RO)	Reduces reactor power with recirc flow in accordance with the subsequent actions of ABN-POWER.
4.	Trigger 4	C (RO)	The 'A' recirc pump fails to follow the automatic controller and must be taken to manual for reduction and balance of recirc flow.
5.	AUTO Trigger 5 at 95% power	I (RO)	APRM 'A' gain drifts during the power reduction resulting in APRM indication reading out of specification for Tech Spec tolerance.
6.	Trigger 6	N (BOP) C (BOP)	Reduces load on DG-1 at completion of OSP-ELEC-M701. DG-1 Governor begins oscillating requiring the emergency trip of the diesel from the control room. **(Columbia LER 98-014)**
7.	Trigger 7	C (All)	DEH oil leak resulting in a Main Turbine trip and a loss of Bypass Valves.
8.	Initiated by event 7	M (All)	Reactor scrams due to the Main Turbine trip. A 100% ATWS prevents

	event 7 actions		inward rod movement by scram **(Columbia IPE)**
9.	Initiated manually by disconnect of GDS computers	C	The Graphical Display System (GDS) locks up during the Main Turbine trip transient.
10.	AUTO Trigger 10 on SLC initiation	C	The SLC common discharge header ruptures in the reactor building preventing boron from reaching the core.

(N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor

Event No. 1

Description: Place RHR loop ‘A’ into the Suppression Pool Cooling mode.

This even is initiated by the turnover sheet. The event endpoint occurs when the ‘A’ loop of RHR is in Suppression Pool Cooling mode.

Time	Position	Applicants Actions or Behavior
	SRO	Directs the BOP to place the ‘A’ loop of RHR into the Suppression Pool Cooling mode per SOP-RHR-SPC.
	BOP	Carries out actions of SOP-RHR-SPC: ➢ starts SW-P-1A (or allows to auto start upon start of RHR-P-2A) ➢ starts RHR-P-2A ➢ verifies RHR-FCV-64A opens during low flow conditions ➢ throttles open RHR-FCV-24A ➢ verifies RHR-FCV-64A closes at approximately 800 gpm flow ➢ closes RHR-V-48A ➢ monitors Suppression Pool temperature ➢ reports to CRS that the ‘A’ Loop of RHR is in Suppression Pool Cooling mode

COMMENTS: 8/6/02 Added RHR-V-2B tagged out to initial conditions and turnover sheet to force the crew to place Loop ‘A’ of RHR into suppression pool cooling to accommodate recent procedure change.

Event No. 2		
Description: High-pressure feedwater heater '6A' level controller fails high resulting in the trip of feedwater heater '6A'. *This event is **MANUALLY initiated by** <u>**TRIGGER 2**</u> after Loop 'A' of RHR is placed into Suppression Pool Cooling mode. The event endpoint occurs when core flow is LE 80Mlbm/hr.*		
Time	**Position**	**Applicants Actions or Behavior**
	BOP	Reports Heater 6A Level High Refers to Annunciator Response Procedure (4.820.A2, 8-2) ➢ check setpoint on HD-LIC-6A and adjust as required to clear alarm ➢ if necessary, takes manual control of HD-LIC-6A or HD-LIC-6A2 and restores normal level.
	RO	Monitors plant for loss of feedwater heating affects.
	BOP	Reports Heater 6A Level High Trip and MSR Drain Tank Level High Refers to Annunciator Response Procedures (4.820.A2, 7-2 & 4.840.A2, 1-2; 4-1) ➢ checks automatic actions ➢ informs CRS that the procedure refers to ABN-POWER ➢ places HD-LIC-6A in MANUAL and opens drain valve HD-LCV-6A1 to 100% ➢ places HD-LIC-6A2 in MANUAL and opens dump valve HD-LCV-6A2 TO approximately 50%. ➢ Verifies proper operation of MSR Drain tank level controllers and takes to MANUAL as necessary to control level.
	SRO	Refers to ABN-POWER for actions associated with Loss of FW Heating: ➢ Directs power to be maintained at LE 3486 MWt ➢ *Directs reactor power reduction with core flow to be reduced

		to LE 80Mlbm/hr ➢ Directs insertion of control rods per the fast shutdown sequence to maintain rod line below 100% ➢ Directs the reduction of power per PPM 3.2.4 to stay within the acceptable feedwater temperature-to-power operating region of Att. 7.1 ***CRITICAL TASK**
	RO/BOP	Carries out actions of ABN-POWER: ➢ *Maintains power LE 3486 MWt ➢ Monitors for thermal hydraulic instabilities ➢ *When directed, reduces core flow to LE 80 Mlbm/hr at a rate not to cause a Level 8 trip of the Main and FW turbines. (see event 3) ➢ When directed, inserts control rods to maintain rod line below 100% ➢ Ensures power is within the acceptable feedwater temperature-to-power operating region of Attachment 7.1 of PPM 3.2.4 ***CRITICAL TASK**
COMMENTS:		

Event No. 3		
Description: Reduces reactor power with recirc flow in accordance with the subsequent actions of ABN-POWER. *This event is initiated by the procedure addressed in Event #2. The event endpoint occurs when core flow is LE 80Mlbm/hr (this event is a subset of event 2, thus having the same endpoint)*		
Time	**Position**	**Applicants Actions or Behavior**
	SRO	Directs the RO to reduce power with recirc flow to LE 80 Mlbm/hr core flow Cautions the RO to reduce power at a rate to prevent a Level 8 trip of the main and feedwater turbines.
	RO	Reduces reactor power by reducing core flow with reactor recirc system at a rate as to prevent a Level 8 trip. Informs the CRS when he has reached LE 80 Mlbm/hr
	BOP	Monitors plant
COMMENTS: Event 3 actions are a subset of Event 2		

Event No. 4		
Description: The ‘A’ recirc pump fails to follow the automatic controller and must be taken to manual for reduction and balance of recirc flow. *This event is initiated by the power reduction requirement of ABN-POWER addressed in Event 2 & 3. The malfunctions are setup to automatically clear themselves to allow manual control of RRC when the Manual button is depressed on the ‘A’ RRC pump controller. The event endpoint occurs when the ‘A’ RRC pump controller has been taken to MANUAL and pump speed reduced while in manual.*		
Time	**Position**	**Applicants Actions or Behavior**
	RO	Realizes that the ‘A’ RRC pump speed is not changing when a change is commanded with the automatic controller. Reports the equipment problem to the SRO Takes manual control of the ‘A’ RRC controller and maintains pump speeds matched during the power reduction.
	SRO	If manual control is not taken by the RO, directs the RO to place the ‘A’ RRC pump controller in Manual and maintain pump speeds matched during the power reduction.
COMMENTS:		

Event No. 5

Description: APRM 'A' gain drifts during the power reduction resulting in APRM indication reading out of specification for Tech Spec tolerance.

*This event **AUTOMATICALLY initiates*** **TRIGGER 5** *when reactor power drops below 95% on APRM 'B'. The GAF malfunction ramps to its new value over a 6-minute period. The event endpoint occurs when APRM 'A' has been bypassed.*

Time	Position	Applicants Actions or Behavior
	RO	Reports that APRM 'A' is not tracking with the other APRMs during the power reduction. References the PPCRS overview screen or L-4 screen to determine APRM GAF's and determines that APRM 'A' is out of spec.
	SRO	Directs the STA to perform a GAF adjustment on APRM 'A' per TSP-APRM-C301. Directs RO to maintain reactor power constant while gain adjustment is made on the 'A' APRM Directs the RO to bypass APRM 'A'
STA CUE: When directed to perform TSP-APRM-C301, inform the CRS that you will proceed to adjust the gain after action to reduce power per ABN-POWER. Five minutes after completion of power reduction, report back that the gain adjust is not working.		
	RO	Bypasses APRM 'A' for gain adjustment
	SRO	After STA informs that the APRM 'A' gain adjust is not working, references Tech Specs and determines that no action is required. If APRM 'A' is not already bypassed, directs APRM 'A' to be bypassed. Prepares a tracking LCO for the inoperable APRM

COMMENTS: 8/6/02 – Changed statement regarding output of APRM from "non-conservative" to out of spec for Tech Spec tolerance.

Event No. 6

Description: Reduces load on DG-1 at completion of OSP-ELEC-M701. DG-1 Governor begins oscillating requiring the emergency trip of the diesel from the control room.

This event is ***MANUALLY initiated with*** **TRIGGER 6** *when the BOP begins to reduce load on DG-1, or as directed by the lead examiner. The event endpoint occurs when DG-1 has been emergency tripped from Board C.*

Time	Position	Applicants Actions or Behavior
	BOP	Informs the CRS that the run time is completed for the DG-1 surveillance and that he will be reducing DG-1 output. While reducing DG-1 load, determines that the DG-1 governor is oscillating and reports this to the CRS.
	SRO	Directs the BOP to manually trip DG-1
	BOP	Emergency trips DG-1 from Board C.

COMMENTS:

Event No. 7		
Description: DEH oil leak resulting in a Main Turbine trip and a loss of Bypass Valves. *This event is **MANUALLY initiated with <u>TRIGGER 7</u>** after the BOP operator trips DG-1 or as directed by the lead examiner. The event endpoint occurs when the reactor is scrammed.*		
Time	**Position**	**Applicants Actions or Behavior**
	BOP	Reports DEH reservoir low and low-low Refers to the annunciator response procedure (4.820.B1, 6-7) ➢ Has an equipment operator check local level indication for the DEH reservoir ➢ Notifies CRS of reference to ABN-DEH-LEAK
	SRO	Directs actions of ABN-DEH-LEAK: ➢ *Directs manual scram of the reactor and entry into PPM 3.3.1 ➢ Directs tripping of the Main Turbine ➢ Directs tripping of the Main Generator ***CRITICAL TASK**
	RO	*Manually scrams the reactor and carries out immediate scram actions: ➢ Takes the mode switch to SHUTDOWN ➢ Reports reactor power, reactor pressure, and reactor level ➢ Inserts SRMs and IRMs ➢ Reports control rod status ***CRITICAL TASK**
	BOP	Trips the Main Turbine and the Main Generator Verifies transfer of electric buses from the Normal transformer to the Startup transformer
COMMENTS:		

Event No. 8		
Description: Reactor scrams - A 100% ATWS prevents inward rod movement by scram *This event is initiated by actions taken in Event 7. The event endpoint occurs when all control rods are in their "full-in" position.*		
Time	**Position**	**Applicants Actions or Behavior**
	RO	Continues with reactor scram actions: ➢ Reports that all rods are not in – ATWS ➢ Depresses manual scram pushbuttons ➢ Initiates ARI ➢ Inserts SRMs and IRMs ➢ Reports that there is still no inward rod motion
	SRO	Enters PPM 5.1.2 due to ATWS condition • *directs BOP to inhibit ADS and take manual control of HPCS • *directs RO to trip both RRC pumps and initiate SLC • directs RO/BOP to ensure isolations and auto initiations have occurred • directs BOP to bypass MSIV isolations per PPM 5.5.6 and ECCS valve interlocks per PP 5.5.1 • *directs RO to stop and prevent FW injection and maintain RPV level –65" to –192" (or some band in between) • directs BOP to maintain RPV pressure 800-1000 psig using SRVs • *directs RO/BOP to attempt to insert control rods using PPM 5.5.10 and 5.5.11 ***CRITICAL TASK**
	RO/BOP	*Inhibits ADS *Takes manual control of HPCS • Manually initiates HPCS with ARM and DEPRESS • Secures HPCS pump and/or shuts HPCS-V-4

		*Trips RRC pumps and initiates SLC
		
		Verifies +13" isolations
		
		Bypasses MSIV isolations using PPM 5.5.6
		
		• Obtains procedure package and keys from EOP drawer
		
		• At P609, places MS-RMS-S84 to BYPASS position
		
		• At P611, places MS-RMS-S85 to BYPASS position
		
		• Reports completion of PPM 5.5.6 to the SRO
		
		Bypasses ECCS valve interlocks using PPM 5.5.1 and inserting keys:
		
		• At P625, takes HPCS-RMS-S25 to OVERRIDE
		
		• At P629, takes LPCS-RMS-S21 to OVERRIDE
		
		• At P629, takes RHR-RMS-S105 to OVERRIDE
		
		• At P618, takes RHR-RMS-S106 to OVERRIDE
		
		• At P618, takes RHR-RMS-S107 to OVERRIDE
		
		*Maintains water level using FW system
		
		Reports that control rods appear to be drifting into the core.
		
		*Overrides ARI logic using PPM 5.5.10: **(see note 1)**
		
		• Obtains procedure package and fuse pullers from EOP drawer
		
		• At P650, pulls one of the following fuses on TB1:
		
		    • F01; F02; F03; F04
		
		• At P650, pulls one of the following fuses on TB2:
		
		    • F01; F02; F03; F04
		
		• Reports to SRO that PPM 5.5.10 is complete
		
		*Performs actions of PPM 5.5.11: **(see note 1)**
		
		• Obtains procedure package and tools from EOP drawer
		
		• Determines that the appropriate sections of the procedure for the existing conditions are Tabs B and F.
		
		TAB B:

		• Places SDV HIGH LEVEL TRIP control switch to BYPASS • Overrides RPS trip signals: • Installs jumper between terminal stud 2 on RPS-RLY-K9B and terminal stud 4 on PRS-RLY-K12F in P611 • Installs jumper between terminal stud 2 on RPS-RLY-K9D and terminal stud 4 on PRS-RLY-K12H in P611 • Installs jumper between terminal stud 2 on RPS-RLY-K9A and terminal stud 4 on PRS-RLY-K12E in P609 • Installs jumper between terminal stud 2 on RPS-RLY-K9C and terminal stud 4 on PRS-RLY-K12G in P609 • Resets the scram on P603 • When SDV has drained for more than 2 minutes, checks rod density and initiates a manual scram • Reports any rod movement, or lack thereof, to the SRO **SIM CUE: When Tab B is completed and after the RO has attempted his first "re-scram", remove the Hydraulic ATWS malfunctions to allow inward rod motion on the next "re-scram".** TAB F: • Starts second CRD pump if available • Places SDV HIGH LEVEL TRIP control switch to BYPASS • Resets scram if possible • Bypasses all RSCS rod blocks: • Installs a jumper from terminal 7 to terminal 8 on the following two Bailey Alarm Cards on P613 • AHH (MS-PS-654A) • AGG (MS-PS-654B) • Places RWM bypass switch to BYPASS

		Manually drives rods and informs SRO of results ***CRITICAL TASK**
	RO	Reports when all rods are in
	SRO	Exits PPM 5.1.2 and re-enters 5.1.1 • directs RO to stop SLC and restore RPV level to +13” to +54” • may direct BOP to remove RPS jumpers
COMMENTS: NOTE 1: These tasks may not be necessary depending on how the crew prioritizes their actions.		

Event No. 9		
Description: The Graphical Display System (GDS) locks up during the Main Turbine trip transient. *This event is initiated manually (by disconnecting the GDS monitors in the panels) as soon as the verbal scram report is completed by the RO. The event endpoint occurs when the scenario is terminated.*		
Time	**Position**	**Applicants Actions or Behavior**
	RO/BOP	Reports the loss of the Graphical Display System
	SRO	Directs RO/SRO to monitor parameters on hardwired instrumentation
COMMENTS:		

Event No. 10		
Description: The SLC common discharge header ruptures in the reactor building preventing boron from reaching the core. *This event is **automatically initiated by** **<u>TRIGGER 10</u>** as soon as the SLC system is taken to ON. The event endpoint occurs when the scenario is terminated.*		
Time	**Position**	**Applicants Actions or Behavior**
	RO	Reports that SLC system discharge pressure has dropped approximately 50 psig and that flow rate had dropped from 84 psig to 14 psig with both pumps indicating energized.
	SRO	Dispatches OPS-2 to investigate the SLC system
OPS-2 CUE: If requested to investigate the SLC system, wait several minutes and report back that there is a leak at the combined outlet header downstream of the Squib Valves.		
	SRO	May direct SLC to be secured due to the leak, or may decide to leave it running since there is 14 gpm of borated solution being injected into the reactor.
	RO	If directed, secures both SLC pumps due to the SLC system leak.
COMMENTS: **<u>SCENARIO ENDPOINT</u>: When all control rods are in and reactor water level has been restored to its normal band and SLC is secured, the scenario may be terminated.**		

SRO TURNOVER INFORMATION

The plant is at 100% power.

DG-1 is fully loaded for OSP-ELEC-M701 (currently at step 7.5.62). There are 20 minutes left on the one-hour diesel run.

Wetwell temperature is slowly rising due to two SRVs that are leaking by. The off-going shift recommends that suppression pool cooling be initiated as soon as you take the shift. RHR-V-24B is tagged out for replacement of the motor operator. The expected time to job completion is two hours.

Facility: COLUMBIA **Scenario No.:** 3 **Op-Test No.:** 1

Examiners: ____________________ **Operators:** ____________________

____________________ ____________________

____________________ ____________________

Initial conditions: IC-195 (batch file NRC02.3.txt). Reactor power is at 22% on a beginning of life core. The feedwater system is in a "10 Valve" lineup with 2 reactor feed pumps in operation.

Turnover: The reactor is at 22% power with a reactor shutdown in progress. The feedwater system is in a "10 Valve" lineup with 2 reactor feed pumps in operation. A power reduction to 15% has been directed, at which point, the 'B' reactor feedwater pump will be taken out of service. You are to hold the plant at 15% with the main turbine on line while the Feedwater system engineer gathers data on the feedwater system.

Event No.	Malf. No.	Event Type*	Event Description
1.	Initiated by turnover	R (RO)	Reactor power reduction to 15% by inserting control rods
2.	Trigger 2	I (RO)	The "C" Recirc Flow Unit fails downscale resulting in a rod block requiring the RO to bypass the unit.
3.	Trigger 3	I (RO)	The RWM fails, requiring the RO to bypass the RWM.
4.	Initiated by turnover	N (BOP)	The "B" RFP is removed from service
5.	Trigger 5	C (BOP)	The running plant service water pump trips. The standby plant service water pump fails to auto start requiring the BOP to manually start it.
6.	Trigger 6	C (BOP)	The hotwell level controller fails causing a low condenser hotwell level, requiring the BOP to manually restore level and transfer control to the standby controller.
7.	Trigger 7	C (All)	The shaft of the running plant service water pump shears, resulting in a total loss of plant service water, requiring a manual reactor scram.
8.	Trigger 8	M (All)	A large LOCA occurs on the "B" reactor recirc loop.
9.	Preset	C	The HPCS pump experiences reduced head resulting in the loss of injection capability.
10.	Preset	C	RHR pump 2A fails to auto start on its initiation signal.

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor

Event No. 1		
Description: Reactor power reduction to 15% by inserting control rods *This event is initiated by the turnover sheet. The event endpoint occurs when power has been reduced by 5% from its original value.*		
Time	**Position**	**Applicants Actions or Behavior**
	SRO	Directs a power reduction to 15%
	RO	Verifies that the title on the pull sheet is the same as the title on the RWM display Inserts control rods to reduce power using the rod pull sequence sheets in the reverse order. Monitors reactor power Monitors RWM tracking while in the LPAP Notes the time when RSCS and RWM LPSP indication is received. Informs the CRS when power is at 15%
COMMENTS: 8/6/02 Changed initial power level and desired power level from 25 and 20 to 22 and 15 respectively to provide the necessary drop below the LPSP for RWM for event 3. Event 4 was moved to the event 2 position with the remaining events remaining in progression in order to better test the operator's reaction to this event.		

Event No. 2		
Description: The “C” Recirc Flow Unit fails downscale resulting in a rod block requiring the RO to bypass the unit. *This event is **MANUALLY initiated by** <u>**TRIGGER 2**</u> after the RO begins inserting control rods, or by the direction of the lead examiner. The event endpoint occurs when the ‘C’ Recirc Flow Unit has been bypassed.*		
Time	**Position**	**Applicants Actions or Behavior**
	RO	Reports Flow Reference Off Normal and Rod Out Block annunciators Refers to annunciator response procedures (4.603.A8, 3-6 & 4.603.A7, 2-7) ➢ Determines which flow comparater channel is causing the alarm by checking lights at P608 or P603 ➢ Informs the CRS that the procedure says to consider bypassing the failed channel
	SRO	Directs the RO to bypass the failed channel Refers to LCS 1.3.2.1 and TS 3.3.1.1 (will have to go to TS bases for inoperable flow unit)
	RO	Bypasses the ‘C’ recirc flow unit
COMMENTS:		

Event No. 3		
Description: The RWM fails, requiring the RO to bypass the RWM. *This event is **MANUALLY initiated by** <u>**TRIGGER 3**</u> after the RO bypasses the failed recirc flow unit in Event 2, or by the direction of the lead examiner. The event endpoint occurs when the RWM has been bypassed.*		
Time	**Position**	**Applicants Actions or Behavior**
	RO	Reports that the RWM is not following the rod insertion sequence. *NOTE: The operator may not notice that there is a problem with the RWM until he reaches the LPSP (approx. 17%).* Stops rod movement and awaits further direction from the CRS.
	SRO	Refers to ABN-RWM: ➢ Determines if the reactor is within the LPSP ➢ If within the LPSP, directs the RO to stop any further rod motion, except my manual scram if necessary. ➢ Refers to Tech Specs 3.3.2.1 and determines that the RWM is not required until 10% RTP. ➢ Directs the RWM to be reinitialized per PPM 2.1.4 ➢ After determining that the RWM will not return to service, directs the RWM be manually bypassed per PPM 2.1.4 and complies with Tech Spec 3.3.2.1, *requiring that rod movement be verified in compliance with the BPWS by a second licensed operator or member of the technical staff. *NOTE: All rod movements (except during emergency situations) are required to be verified by a second licensed operator or tech staff by Columbia Generating Station procedures.* ***CRITICAL TASK**
COMMENTS:		

Event No. 4

Description: The “B” RFP is removed from service

This event is initiated by the turnover sheet. The event endpoint occurs when the feedpump turbine has been placed on the turning gear.

Time	Position	Applicants Actions or Behavior
	SRO	Directs the shutdown of the ‘B’ RFP per PPM 2.2.4, section 5.17
	BOP	Performs PPM 2.2.4, section 5.17: ➢ informs HP of potential change in radiological conditions ➢ opens COND-V-149 ➢ lowers speed of RFP turbine ‘B’ until the MIN lamp illuminates ➢ ensures that the startup valve is responding as necessary to control the Reactor Water Level ➢ Places RFW-FIC-2B in Manual. ➢ Trips RFP turbine ‘B’ ➢ Ensures that RFW-FCV-2B closed. Adjust RFW-FIC-2B controller output to zero ➢ Closes RFW-V-102B, pump discharge ➢ Ensures MS-V-142B, BS-V-44B, and BS-V-45B auto open ➢ Closes MS-V-105B ➢ Closes BS-V-17B ➢ As the turbine slows to LT 1 RPM, places the turbine turning gear control switch to AUTO ENGAGE and ensures turning gear engagement. ➢ Starts the Aux. Oil Pump for the feedwater turbine ➢ Stops the main oil pump for the feedwater turbine ➢ Informs the CRS that the ‘B’ RFP is removed from service

COMMENTS:

Event No. 5

Description: The running plant service water pump trips. The standby Plant Service Water pump fails to auto start requiring the BOP to manually start it

*This event is **MANUALLY initiated by** __TRIGGER 5__ after the BOP completes the RFP removal event or at the direction of the lead examiner. The event endpoint occurs when the standby Plant Service Water pump has been manually started.*

Time	Position	Applicants Actions or Behavior
	BOP	Responds to TSW-P-1B motor trip annunciator Carries out actions of the ARP: ➢ Notifies the CRS of the loss of the operating TSW pump and the failure of the standby pump to auto start. ➢ Manually starts the standby TSW pump ➢ Ensures the discharge valve for the running pump opens ➢ Ensures the discharge valve for the tripped pump closes ➢ Checks the TSW discharge header pressure increasing on TSW-PI-28 ➢ Informs CRS of reference to ABN-TSW
	SRO	Refers to ABN-TSW and verifies that ARP actions meet the requirements of the ABN.
	RO	Monitors: ➢ reactor level, pressure, and power ➢ equipment for temperature increases ➢ drywell pressure for increase

COMMENTS:

Event No. 6		
Description: The hotwell level controller fails causing a low condenser hotwell level, requiring the BOP to manually restore level and transfer control to the standby controller. *This event is **MANUALLY initiated by** __**TRIGGER 6**__ after the completion of event 5 or at the direction of the lead examiner. The event endpoint occurs when Hotwell Level Control has been transferred to the alternate controller and hotwell level is recovering.*		
Time	**Position**	**Applicants Actions or Behavior**
	RO/BOP	Reports Main Condenser Hotwell Level Low annunciator.
	SRO	Gives BOP direction to carry out actions of the ARP (4.840.A3, 7-4).
	BOP	Carries out actions of the ARP: ➢ Checks in-service hotwell level controller for level indication ➢ Reports that the in-service hotwell level controller indicates low hotwell level and that the controller output is demanding the low level rather than responding to raise hotwell level. ➢ Checks hotwell LCV's aligned for raising hotwell level and notes that they are not due to the apparent controller failure ➢ Checks COND-V-17 is open ➢ Checks CST levels are normal ➢ Recommends that the Hotwell Level Control be swapped to the alternate controller
	SRO	Directs that the alternate Hotwell Level Controller be placed into service May direct EO to investigate hotwell level control system.

	BOP	Continues with the ARP: ➢ Places both Hotwell Level Controllers into MANUAL ➢ Places the on-coming controller at 50% ➢ Shifts Hotwell Level Control by placing the Condenser Hotwell Level Select switch to COND-LIC-1 ➢ If desired, places the on-coming controller in AUTO
COMMENTS: 8/6/02 Added clarification to the event title to show the direction of the hotwell level failure		

Event No. 7

Description: The shaft of the running plant service water pump shears, resulting in a total loss of plant service water, requiring a manual reactor scram.

*This event is **MANUALLY initiated by** __**TRIGGER 7**__ after the BOP completes the swap of hotwell level controllers, or at the direction of the lead examiner. The event endpoint occurs when the immediate operator actions for a scram have been completed.*

Time	Position	Applicants Actions or Behavior
	BOP	Reports Plant Service Water Header Pressure Low annunciator References and carries out actions of ARP (4.840.A5, 5-7): ➢ Checks discharge pressure on P840 ➢ Checks TSW-PCV-20 operating properly ➢ Sends equipment operators to check for TSW leakage and system operation ➢ Determines that running pump amps are low and may deduce that there is a problem with the running TSW pump. ➢ Informs the CRS that the ARP refers to ABN-TSW
	SRO	Refers to ABN-TSW: ➢ Determines that neither TSW pump is running and declares a total loss of TSW ➢ Directs the RO to reduce RRC flow and SCRAM the reactor
	RO	Carries out immediate SCRAM actions: ➢ *Takes mode switch to SHUTDOWN ➢ Reports reactor power, pressure, and level (reports level below +13” and states that this is an EOP entry condition) ➢ Inserts IRMs and SRMs ➢ Reports that all control rods are in ➢ Refers to PPM 3.3.1 to verify that all immediate actions have been carried out. ***CRITICAL TASK**

	SRO	Enters EOP 5.1.1 (RPV Control) based on low reactor water level ➢ Directs the BOP or RO to verify isolations associated with reactor water level below Level 2. ➢ Directs the RO to maintain reactor water level in a band of +13" to +54" using table one injection systems (feed, RCIC, CRD, Condensate, HPCS) ➢ Directs the BOP to maintain reactor pressure in a band of 800 to 1000 psig using BPVs (and alternate pressure control systems if necessary) ➢ Directs RO to enter and carry out actions of PPM 3.3.1
	RO	Lines up the feedwater system for startup level control with the "10 valves" in automatic to maintain the prescribed water level band Verifies that RRC pumps are at 15 Hz, or trips RRC pumps if directed Swaps at least four IRM/APRM recorders to IRM and ranges IRMs down to monitor power Verifies that the SCRAM has been announced over the Plant PA
	BOP	Monitors BPV operation to ensure reactor pressure remains in the prescribed band. Trips the Main Turbine when output is LT 50 Mwe and verifies Main Generator output breakers have opened Verifies power transfer to TR-S May start RCIC to aid in pressure/level control
	SRO	Directs alignment of Fire Water to the CAS and SA compressors per ABN-TSW Directs monitoring of TSW cooled equipment
COMMENTS:		

Event No. 8		
Description: A large LOCA occurs on the "B" reactor recirc loop. *This event is **MANUALLY initiated by** <u>**TRIGGER 8**</u> after the immediate scram actions have been completed, or at the direction of the lead examiner. The event endpoint occurs when the scenario is terminated.*		
Time	**Position**	**Applicants Actions or Behavior**
	SRO	Re-enters PPM 5.1.1 on low RPV level and 5.2.1 on high DW pressure: • Directs RO/BOP to verify isolation, initiations, and DG starts • *Directs RO to maintain RPV level between –161" and +54" (will give a band within these limits) with Table 1 systems. • *Directs RO/BOP spray of WW before reaching 12 psig in the WW • Directs RO/BOP to confirm RRC pumps are stopped and stop DW cooling fans in prep for DW spray • *Directs RO/BOP to spray DW when WW press exceeds 12 psig and within DSIL • *Directs the securing of containment sprays when pressure drops LE 1.68 psig in respective area. • At –50" RPV level, directs verification of expected isolations and initiations • *At –129", if ADS timer has started, direct inhibition of ADS • *At TAF, determines that LP ECCS is available and that current trend will drop level below –192"; determines Emergency RPV depressurization is required; enters PPM 5.1.3, Emerg. RPV Depress. • *Directs RO/BOP to open 7 SRVs, ADS preferred. ***CRITICAL TASK**
	RO/BOP	Reports that expected initiations, isolations, and DG starts have occurred except that HPCS-V-4 has failed closed (see event 11). *Uses RCIC and FW systems to maintain RPV level *Sprays the WW when directed Confirms that RCC pumps and DW fans have been secured *Sprays the DW when directed

		*Secures WW/DW sprays when if or when LT 1.68 psig in each area *Opens 7 SRVs (ADS preferred) to emergency depressurize the RPV. ***CRITICAL TASK**
	SRO	Directs restoration of RPV level to band of +13” to +54”
	RO/BOP	Controls injection systems to restore RPV level to new band.
COMMENTS:		

Event No. 9		
Description: The HPCS pump experiences reduced head resulting in the loss of injection capability. *This event is self-initiating when the HPCS pump starts. The event endpoint occurs when HPCS has been secured.*		
Time	**Position**	**Applicants Actions or Behavior**
	RO/BOP	Reports that HPCS is not injecting even though it is running
	SRO	Directs HPCS secured
	RO/BOP	Secures HPCS
COMMENTS:		

Event No. 10

Description: RHR pump 2A fails to auto start on its auto initiation signal.

This event is self-initiating when the RHR pump receives an initiation signal. The event endpoint occurs when RHR-P-2A has been manually started.

Time	Position	Applicants Actions or Behavior
	RO/BOP	Reports that RHR-P-2A failed to auto start *Manually starts RHR-P-2A ***CRITICAL TASK**
	SRO	If the operator fails to attempt the manual start of RHR-P-2A, then directs the operator to start RHR-P-2A

COMMENTS:

SRO TURNOVER INFORMATION

The reactor is at 22% power with a reactor shutdown in progress.

The feedwater system is in a “10 Valve” lineup with 2 reactor feed pumps in operation.

A power reduction to 15% has been directed, at which point, the ‘B’ reactor feedwater pump will be taken out of service. You are to hold the plant at 15% power with the main turbine on line while the Feedwater system engineer gathers data on the feedwater system.

The plant is operating at 100% power.

Which of the following describes the effect of pushing the STOP pushbutton for RRC-P-1B?

A. Shuts down the both of the associated ASD channels and opens the source and load breakers causing the pump to trip. RRC-P-1A runs back to 51 hertz.

B. Shuts down the both of the associated ASD channels and opens the source and load breakers causing the pump to trip. RRC-P-1A continues to run at 60 hertz.

C. Trips the slave channel only, RRC-P-1B runs back to 51 hertz, RRC-P-1A continues to run at 60 hertz.

D. Trips the slave channel only, RRC-P-1B runs back to 30 hertz, RRC-P-1A runs back to 30 hertz.

ANSWER: B

QUESTION TYPE:	SRO/RO
KA # & KA VALUE:	202002A1.01 3.2/3.2 10CFR55.41 & 45 - Ability to predict and/or monitor changes in parameters associated with operating the Recirculation System controls including: Recirculation Pump Flow
REFERENCE:	LO000184 rev 13, pages 15, 16, 21, and 22
SOURCE:	**NEW QUESTION –** SRO T2, GP1, #1 RO T2, GP1, #2
LO:	9681 – Describe the operation of the following ASD controls on P602, including expected indications and system response: STOP pushbutton.
RATING:	L3
ATTACHMENT:	NONE
JUSTIFICATION:	Pushing the STOP pushbutton causes both ASD channels to trip and open both the supply and load breakers. This causes the pump to trip. The trip has no effect on the remaining loop. B is correct.
COMMENTS:	

The reactor was operating at 99% power when the reactor scrammed. The following conditions exist:

TR-S has tripped on overcurrent
HPCS and RCIC auto started, RCIC is maintaining level in the normal band
Drywell pressure is .92 psig
Both Recirc pumps have tripped CB-RPT-3A/4A and CB-RPT-3B/4B are open
Both feed pumps are off

Which of the following caused the scram?

A. Reactor level + 8 inches

B. MSIV isolation

C. Reactor pressure 1069 psig

D. Main turbine trip

ANSWER: D

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 295005AK2.03 3.2/3.3 10CFR55.41 & 45 – Knowledge of the interrelations between MAIN TURBINE GENERATOR TRIP and the following: Recirculation System.

REFERENCE: LO000196 rev. 12, pages 13 and 14

SOURCE: **BANK QUESTION – 1998 NRC EXAM -** SRO T1,G2, #4 RO T1, G1, #5

LO: 5023 – Predict the impact on the RRC System of each of the following conditions or events: e. EOC-RPT logic.

RATING: H3 H4

ATTACHMENT: NONE

JUSTIFICATION: The only trip that causes <u>BOTH</u> CB-RPT-3A (3B) and 4A (4B) to open is the EOC-RPT trip coming from a turbine trip.

QUESTION # 3 EXAM KEY

10/04/2002

COMMENTS:

An inboard and outboard NS4 isolation has just occurred. The plant continues to operate at 99% power.

Which of the following is the cause of this isolation?

A. 86 lockout on Bkr 7-1

B. 86 lockout on Bkr 8-3

C. Loss of SL-71

D. Loss of SL-81

ANSWER: B

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 295003AK3.06 3.7/3.7 10CFR55.41 & 45 – Knowledge of the reason for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: Containment Isolation.

REFERENCE: LO000182 pages 56, 57, and 88 LO000161 page 16, LO000173 page 13

SOURCE: **NEW QUESTION** SRO T1, GP1, #1 RO T1, GP2, #3

LO: 5604 – List the actions that would occur on a loss of one or both RPS power supplies to the NS4 Logic.

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: The cause for the inboard and outboard isolation is a loss of RPS B. B is the only selection that causes a loss of RPS B. A causes a loss of RPS A, which would result in the closure of the outboard valves only. C and D would not result in any closure of isolation valves.

COMMENTS:

QUESTION # 5 EXAM KEY
10/04/2002
ex98103

The plant is operating at 100% power when a jet pump nozzle separates from the rams head.

Which of the following describes expected plant indications?

RRC loop flow on the loop with the failed nozzle.....

A. increases, actual core flow decreases, indicated core flow increases, and core thermal power decreases.

B. decreases, actual core flow decreases, indicated core flow increases, and core thermal power decreases.

C. increases, actual core flow increases, indicated core flow decreases, and core thermal power increases.

D. decreases, actual core flow increases, indicated core flow increases, and core thermal power increases.

ANSWER: A

QUESTION TYPE:	SRO/RO
KA # & KA VALUE:	295001AK2.02 3.2/3.3 10CFR55.41 & 45 - Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION and the following: Nuclear Boiler Instrumentation
REFERENCE:	LO000196 rev/ 12. page 29
SOURCE:	**BANK QUSTION – 98 NRC EXAM –** SRO T1, GP2, #2 RO T1, GP2, #1
LO:	5023 – Predict the impact on the RRC System of each of the following conditions or events: a. Jet Pump Failure
RATING:	H3 H4
ATTACHMENT:	NONE
JUSTIFICATION:	A is correct because when the failure occurs, RRC flow increase due to lower resistance, actual core flow decreases because there are fewer jet pumps in operation. Indicated flow increases because there is more flow through the failed jet pump, this is reverse but the flow indicators do not differentiate between forward and reverse flow. Core thermal power decreases due to the decrease in core flow.

The plant had been operating for an extended run at full power when an accident occurred. Primary Containment pressure is approaching the Primary Containment Pressure Limit (PCPL).

Which of the following is correct for these conditions?

A. The Drywell must be vented to prevent a failure of the wetwell to drywell interface.

B. The Wetwell must be vented to prevent a failure of the wetwell to drywell interface.

C. Primary containment must be vented irrespective of offsite release to prevent failure of the containment and subsequent loss of ECCS Systems.

D. Primary containment can be vented to prevent failure of the containment and subsequent loss of ECCS Systems when the projected offsite release rate is within limits.

ANSWER: C

QUESTION TYPE: SRO
KA # & KA VALUE: 295010 2.4.6 3.1/4.0 10CFR55.41, 43.5, & 45 – Knowledge of symptom base EOP mitigation strategies during High Drywell Pressure.
REFERENCE: PPM 5.0.10 rev 6, pages 93, 94, and 260
SOURCE: **NEW QUESTION –** SRO T1, GP1, #7
LO: 8364 – Given a list, identify the two reasons for venting the primary containment irrespective of offsite release rates when the Primary Containment Pressure Limit is reached.
RATING: L2
ATTACHMENT: NONE
JUSTIFICATION: PPM 5.0.10 states that the containment must be vented prior to exceeding the PCPL to prevent potential containment failure and the potential loss of ECCS Systems. C is correct. D is incorrect because containment integrity is maintained at this point irrespective of offsite release. A and B are incorrect because PCPL has no effect on the WW to DW interface.
COMMENTS:

QUESTION # 7 EXAM KEY
10/04/2002
ex00012

The plant was operating at 100% power when a fire caused the abandonment of the Control Room. The CRO1, at the Remote Shutdown Panel is attempting to contact the CRS at the Alternate Remote Shutdown Panel. The door between the Remote and Alternate Remote Shutdown rooms is jammed closed and cannot be opened.

Which of the following describes the communication systems permanently installed at both of these panels for communication with the other room?

A. Plant page and plant radio

B. Plant page and plant phones

C. Sound powered phones and plant radio

D. Sound powered phones and plant phones

ANSWER: D

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 295016AK2.02 4.0/4.1 10CFR55.41 - Knowledge of the interrelations between CONTROL ROOM ABANDONMENT and the following: Local control stations.

REFERENCE: LO000210 rev. 5, pages 3 & 4

SOURCE: **BANK QUESTION – 2000 NRC EXAM– Direct –** SR0 T1, GP1, #6 RO T1, GP2, #8

LO: 7739 – State the communications systems available at the Remote Shutdown room.

RATING: L3

ATTACHMENT: NONE

JUSTIFICATION: Sound powered phones and Plant phones are the only systems available at both the RSD Panel and ARSD Panel. D is the correct answer. The other 3 are all incorrect because one or both are not available in either the RSD or the ARSD Panels.

COMMENTS: This question has been slightly modified. It is still counted as direct from the bank.

The EOPs direct entering PPM 5.1.1 RPV Control when suppression pool level cannot be maintained greater than 19 feet 2 inches.

Which of the following is the basis for this direction?

A. Specific directions for reactor level control with low suppression pool level are given in PPM 5.1.1 RPV Control.

B. Steam condensation during a LOCA cannot be assured with suppression pool level less than 19 feet 2 inches.

C. Suppression pool volume below 19 feet 2 inches is not adequate for steam condensation during a 100% power ATWS

D. The conditions for entry into PPM 5.2.1 Primary containment Control may not have caused an entry into PPM 5.1.1 RPV Control.

ANSWER: D

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 295030EK3.06 3.6/3.8 10CFR55.41 & 45 - Knowledge of the reasons for the following responses as they apply to LOW SUPPRESSION POOL WATER LEVEL: Reactor SCRAM

REFERENCE: PPM 5.0.10 rev 6 page 263

SOURCE: **NEW QUESTION** SRO T1, GP1, #19 RO T, GP2, #14

LO: 8383 – Given a list, identify the statement that describes the reason for entering PPM 5.1.1 RPV Control if wetwell level cannot be maintained below the SRVTPLL

RATING: L2
ATTACHMENT: NONE
JUSTIFICATION: PPM 5.0.10 gives the reason as stated in D. A is incorrect because there are no specific directions for low suppression pool level in 5.1.1. B is incorrect because during a LOCA, steam is exhausted through the DW floor downcomers, which exhaust below the SRV tailpipes. C is incorrect suppression pool volume below 19 feet 2 inches is not considered for ATWS.
COMMENTS:

The plant is in MODE 5 with the full core offloaded following an extended run at rated power. The normal cooling water supply to Fuel Pool Cooling Heat Exchangers has been lost.

Which of the following systems can be used as a backup cooling supply?

A. RCC Reactor Closed Cooling Water

B. CST Condensate Storage and Transfer

C. TSW Plant Service Water

D. SSW Standby Service Water

ANSWER: D

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 295018AA1.01 3.3/3.4 10CFR55.41 & 45 - Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: Backup Systems

REFERENCE: LO000202 rev. 10, page 16

SOURCE: **BANK QUESTION – 2000 NRC EXAM –** SRO T1, GP2, #6 RO T1, GP2, #11

LO: 5371 – Given Fuel Pool Cooling and Cleanup System operating mode and various plant conditions, predict how key FPC/plant parameters will respond to the failure of the following support systems: a. Reactor Building Closed Cooling Water System

RATING: L3

ATTACHMENT: NONE

JUSTIFICATION: RCC is the primary source of cooling for FPC and not the backup. TSW and CST are not hard piped into the FPC heat exchangers. Standby Service Water is the system hard piped into the FPC heat exchangers for a backup system. D is correct.

COMMENTS:

The plant was operating at 98% power when a transient occurred. The following indications were received:

- Rod groups 1, 2, 3, and 4, power available lights (white) are out on P603 and P609
- Both RWCU Pumps have tripped
- AR-EX-1A trips
- NS4 Groups 2, 5, 6, and 7 (outboard) isolated

Which of the following is the reason for these indications?

A. Reference leg leak on MS-LT-26A (Wide Range MS-LR/PR-623A)

B. Variable leg leak on MS-LT-26D (Wide Range MS-LR/PR-623B)

C. Failed (low) undervoltage relay (27) on the EPA Breakers for RPS A.

D. Failed (low) undervoltage relay (27) on the EPA Breakers for RPS B.

ANSWER: C

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 212000K6.05 3.5/3.8 10CFR55.41 & 45 - Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR PROTECTION SYSTEM: RPS sensor inputs

REFERENCE: LO000161 rev. 11, pages 7 and 16, LO000126 rev 8, pages 8 and 10

SOURCE: **NEW QUESTION –** SRO T2, GP1, #5 RO T2, GP1, #8

LO: 5957 – Describe the function of these RPS EPA breaker components: d. Undervoltage light

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: The indications given are for a loss of RPS making A and B incorrect. The loss of group 1, 2, 3, and 4 white lights and the outboard isolation indicate the loss of RPS A. C is correct.

QUESTION # 12 EXAM KEY

10/04/2002

COMMENTS:

QUESTION # 13 EXAM KEY
10/04/2002
ex02004

An accident has occurred causing a Site Area Emergency due to offsite release rate. Rad Waste building ventilation has tripped off.

Which of the following is correct for these conditions?

Attempt to restart Rad Waste Building Ventilation to …

A. assure any release from the Radwaste Building is discharged through an elevated and monitored release point.

B. provide for Radwaste Building atmosphere recirculation and reduce the amount of radioactivity present.

C. cause a ground level release and limit the dispersion of radioactive material.

D. limit the intrusion of radioactive material from the Reactor Building.

ANSWER: A

QUESTION TYPE:	SRO/RO
KA # & KA VALUE:	295017AK3.02 3.3/3.5 10CFR55.41 & 45 - Knowledge of the reasons for the following responses as they apply to HIGH OFF-SITE RELEASE RATE: Plant Ventilation
REFERENCE:	PPM 5.0.10 rev. 6 page 302
SOURCE:	**BANK QUESTION LR00967– MODIFIED -** SRO T1, GP1, #15 RO T1, GP2, #9
LO:	8477 – Given a list, identify the statement that describes the purpose of restarting turbine building and Radwaste building HVAC during attempts to control offsite radioactivity release rates above the ALERT level.
RATING:	L3
ATTACHMENT:	NONE
JUSTIFICATION:	B & C are incorrect because operation of RW Ventilation provides for an elevated exhaust, not recirculation of atmosphere. D is incorrect because RW building Ventilation is not designed to limit intrusion of radioactive material from the RB. A is correct as stated in the basis for PPM 5.4.1.
COMMENTS:	

The plant is in MODE 5 with fuel movement underway. The CRO notes both EDR-V-394 and 395, EDR-P-5 Discharge to Waste Collector Tank in Radwaste, have closed.

Which of the following caused these indications?

A. EDR-5 sump level High High

B. Drywell pressure 1.59 psig

C. ARM-RIS-23, CRD Pump Room, 215 mr/hr

D. Rx Building Exhaust Plenum 16 mr/hr

ANSWER: D

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 295036EA1.04 3.1/3.4 10CFR55.41 - Ability to operate and/or monitor the following as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL: Radiation Monitoring

REFERENCE: LO000139 rev. 9, page 7

SOURCE: **BANK QUESTION – 2000 NRC EXAM**
SRO T1, GP, #15 RO T1, GP3, #4

LO: 5333 – List the isolation signal and setpoints for the following valves: EDR-V-394 & 395, ED-R-5 Sump Pump discharge isolation

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: EDR-V-394 and 395 isolate on an FAZ signal. D is correct.
COMMENTS:

The plant is operating at 99% power when an undervoltage occurs on SM-7 due to a spurious trip of BKR 1-7. CRA-FN-2A1 (MC-7B) has tripped from the loss of power.

Which of the following is correct concerning these conditions?

A. CRA-FN-2A1 must be manually restarted when MC-7B is repowered.

B. CRA-FN-2A1 auto starts 3 seconds following the undervoltage.

C. CRA-FN-2A1 auto starts 5.5 seconds after the undervoltage.

D. CRA-FN-2A1 auto starts 10 seconds after the undervoltage.

ANSWER: C

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 223001K2.09 2.7/2.9 10CFR55.41 - Knowledge of electrical power supplies to the following: Drywell Cooling Fans

REFERENCE: LO000127 rev. 10, pages 30 & 31 LO000182 rev. 12 pages 41 and 88

SOURCE: **NEW QUESTION –** SRO T2, GP1, #13 RO T2, GP1, #19

LO: NO LO

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: MC-7B loses power during the undervoltage on SM-7. Since it is a critical power supply, it repowers and the fan starts 5.5 seconds later when B-7 closes. C is correct.

COMMENTS:

A plant startup is in progress. you have been given the direction to start the A Reactor Feed Pump.

What are the requirements for procedure usage for this evolution?

A. The procedure need not be present for this evolution if the operator assigned is familiar with the task.

B. The procedure must be present but steps can be skipped for more efficient operation of the plant if they are noted administratively on the front page of the procedure.

C. The procedure need not be present for this evolution but the steps of the procedure must be performed in order.

D. The procedure must be present and strict adherence to the procedure is required for operating the equipment.

ANSWER: D

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 2.1.20 4.3/4.2 10CFR55.41, 43.5, & 45 Ability to execute procedure steps.

REFERENCE: SWP-PRO-01 rev. 3, page 11

SOURCE: **BANK QUESTION – 99 NRC EXAM –** SRO T3, #4 RO, T3, #2

LO: 6063 – State the requirement concerning whether or not approved plant procedures shall be used in the performance of plant activities.

RATING: L2

ATTACHMENT: N/A

COMMENTS:

JUSTIFICATION: The Company standard for procedure usage is STRICT ADHERENCE to the procedure. If the evolution is a complex operation and memory cannot be relied on then the procedure must be present. D is correct.

The plant is operating at 96% power with a normal electrical plant lineup when breaker 1-11 trips, breaker 21-11 auto closes and repowers SL-11 from SL-21.

Which of the following conditions caused these indications?

A. Undervoltage on SL-11

B. 86 lockout trip of breaker N1-1

C. Undervoltage on SM-1

D. Overcurrent on SL-11

ANSWER: B

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 262001A3.02 3.2/3.3 10CFR55.41 & 45 - Ability to monitor automatic operations of the A.C. ELECTRICAL DISTRIBUTION including: Automatic bus transfer.

REFERENCE: LO000182 rev. 12, pages 62, 63, and 64

SOURCE: **BANK QUESTION –** SRO T2, GP1, #20 RO T2, GP2, #13

LO: 7767 – Describe the physical connection and/or cause and effect relationship between SL11 and : c. Breaker 21-11

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: A, C, and D are incorrect because neither undervoltage on SL-11 or SM-1 nor overcurrent on SL-11 cause closure of 21-11. Only a lockout associated with TR-N1 or TR-S cause breaker 11-1 to trip and close breaker 21-11. B is correct.

COMMENTS:

The plant is in MODE 3 and the CRS has directed you to flush RHR to Radwaste in preparation for starting Shutdown Cooling per PPM 2.4.2.

Which of the following is correct for this condition?

RHR can be flushed to …

A. MWR-TK-23A/B Chemical Waste Tank

B. FDR-TK-9 Floor Drain Sample Tank

C. EDR-TK-4A/B Waste Sample Tanks

D. EDR-TK-5 Waste Surge Tank

ANSWER: D

QUESTION TYPE:	SRO/RO
KA # & KA VALUE:	2.3.10 2.9/3.3 10CFR55.43 – Ability to perform procedures to reduce excess levels of radiation and guard against personnel exposure.
REFERENCE:	82-RSY-0200-T6 page 2 and figure 1
SOURCE:	**BANK QUESTION – 2000 NRC EXAM –** SRO T3, #12 RO T3, #9
LO:	5652 – Describe the normal flow path through the equipment drain processing system.
RATING:	L2
ATTACHMENT:	N/A
JUSTIFICATION:	Of the listed tanks, only EDR-TK-5 is capable of receiving RHR flush water. D is correct.
COMMENTS:	Removed section of answers/distracters that required knowledge of HP actions. Changed distracters to RW tanks and made the question a memory question.

The precautions for OSP-RSCS-C401 RSCS CFT Prior to Reactor Startup, prohibit selecting a bypassed control rod to verify RSCS Operability.

Which of the following describes the method used to verify that the selected control rod is available for RSCS Operability?

A. Depress the “RODS FULL IN/BYPASS” pushbutton to illuminate the BYPASS light. All bypassed control rods are indicated by a red LED.

B. Depress the “ALL RODS/FREE RODS” pushbutton to illuminate the FREE RODS light. All bypassed control rods are indicated by a flashing yellow LED.

C. If a bypassed control rod is selected on the rod select matrix, the RSCS INOP annunciator illuminates.

D. If a bypassed control rod is selected on the rod select matrix, the white position indication (XX-YY) does not illuminate on the full core display.

ANSWER: A

QUESTION TYPE:	SRO/RO
KA # & KA VALUE:	201004A4.01 3.4/3.5 10CFR41 & 45 - Ability to manually operate and/or monitor in the control room: System bypass switches
REFERENCE:	LO000160 rev. 9, page 7
SOURCE:	**BANK QUESTION – LO0898 –** SRO T2, GP2, #3 RO T2, GP2, #3
LO:	5807 – State the function of the following indications and controls on the RSCS operator console: RODS FULL IN/BYPASS.
RATING:	L2
ATTACHMENT:	NONE
JUSTIFICATION:	B is incorrect because a flashing yellow LED indicates the rod is selected on the rod select matrix. C is incorrect because selection of a bypassed control rod does not cause an INOP RSCS. D is incorrect because selection of a bypassed control rod does not cause the full core indication to change. A is correct because the purpose of the BYPASS position is to cause all bypassed control rods to indicate as stated.
COMMENTS:	This question has been reworded but has not changed enough for a modified question. It is the same as Bank Question LO00898.

An accident has occurred following an extended run at power. RCIC room temperature is approaching its maximum safe operating value.

Which of the following is correct concerning these conditions?

PPM 5.1.1 RPV Control must be entered to:

A. Ensure containment integrity.

B. Ensure adequate core cooling.

C. Reduce the dependence on RCIC for reactor level control.

D. Reduce the energy discharged to the Sec. Containment to decay heat level.

ANSWER: D

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 295032EK3.02 3.6/3.8 10CFR55.41 & 45 - Knowledge of the reasons for the following responses as they apply to HIGH SECONDARY CONTAINMENT AREA TEMPERATURE: Reactor SCRAM

REFERENCE: PPM 5.0.10 rev. 6, page 300

SOURCE: **NEW QUESTION –** SRO T1, GP2, #11 RO T1, GP3, #2

LO: 8457 – Given a list, identify the statement that describes the reason for entering PPM 5.1.1, RPV CONTROL if secondary containment parameters are approaching their Maximum Safe Operating Values and a primary system is discharging reactor coolant into secondary containment.

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: A and B are both incorrect because entering PPM 5.1.1 does not ensure either containment integrity or adequate core cooling. D is incorrect again because entry into PPM 5.1.1 does not reduce the dependence on RCIC. C is correct as stated in PPM 5.0.10.

COMMENTS:

A plant startup is in progress at 234 psig and a RWCU blowdown in progress to the main condenser. Several alarms are received and upon investigation, the RO notes RWCU-FCV-33 closed. The RWCU lineup is otherwise normal.

Which of the following is the cause of these indications?

A. RWCU pressure upstream of RWCU-FCV-33 is 17 psig.

B. RWCU pressure downstream of RWCU-FCV-33 is 157 psig.

C. Reactor level is +5 inches.

D. Reactor level is –62 inches.

ANSWER: B

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 204000A3.01 3.3/3.3 10CFR41 & 45 - Ability to monitor automatic operations of the REACTOR WATER CLEANUP SYSTEM including: Pressure downstream of pressure regulating valve

REFERENCE: LO000190 rev. 10, pages 8-11

SOURCE: **NEW QUESTION –** SRO T2, GP2, #6 RO T2, GP2, #6

LO: 5035 – List all RWCU System and filter Demineralizer isolations including setpoints and valves affected.

RATING: H2

ATTACHMENT: NONE

JUSTIFICATION: System pressure must be either less than 5# upstream of 33 or greater than 140 downstream of 33 to isolate the valve. B would cause the valve to close and is correct. C and D are incorrect because reactor level of +5 inches has no effect on RWCU, while any level less than –50 inches causes a complete system isolation (V-1 & 4 close also).

COMMENTS:

The plant was operating at 99% power when a transient occurred. The following conditions exist:

P603 A7 drop 2.2 RPV PRESS HIGH TRIP	Illuminated
P603 A8 drop 2.2 RPV PRESS HIGH TRIP	Illuminated
P603 A8 drop 3.4 ½ SCRAM SYSTEM B	Illuminated
Reactor Pressure RFW-PI-605	1076 psig
Reactor Power	99%

Which of the following procedures is entered first?

A. ABN - PRESSURE

B. ABN - POWER

C. PPM 5.1.1 RPV Control

D. PPM 5.1.2 RPV Control - ATWS

ANSWER: C

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 2.4.45 3.3/3.6 10CFR55.43 – Ability to prioritize and interpret the significance of each annunciator or alarm

REFERENCE: PPM 1.3.1 rev 56, page 30

SOURCE: **BANK QUESTION – 2000 NRC EXAM –** SRO T3, #15 RO T3, #11

LO: 8044 – Given abnormal and annunciator response procedure steps that conflict with EOP steps being performed, determine which procedural steps take precedence.

RATING: H2

ATTACHMENT: NONE

JUSTIFICATION: PPM 1.3.1 directs the priority/precedence of the Volume 5 procedures over the Vol. 4 Abnormals. With reactor pressure >1060 PPM 5.1.1 should be entered and the actions directed there taken. Even though this scenario is an ATWS, PPM 5.1.1 is the correct procedure. C is correct.

QUESTION # 24 EXAM KEY
10/04/2002
COMMENTS:
ex02012

A reactor startup is in progress with power at 60 on IRM range 2. Several alarms activate along with a Rod Out Block and a ½ Scram on RPS A.

Which of the following caused these indications?

A. Failure of DP-SO-A (24 VDC)

B. Failure of DP-S1-1A (125 VDC)

C. SRM A failed upscale.

D. IRM E ranged from range 2 to range 4.

ANSWER: A

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 215003K2.01 2.5/2.7 10CFR41 – Knowledge of electrical power supplies to the following: IRM channels/detectors

REFERENCE: LO000138 rev. 7, page 7 and LO000188 rev. 6, pages 27, 29, and 34

SOURCE: **BANK QUESTION – MODIFIED –** SRO T2, GP2, #10 RO T2, GP1, #10

LO: 7655 – Predict the effects a failure of 24VDC bus SOA will have on: c. IRM

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: A failure of DP-SO-A causes a rod block and additionally a ½ scram from The IRM Inop. None of the other malfunctions cause a ½ scram. A is correct.

COMMENTS: This question was modified from LR01007, LO00532, and LX00058 and given a new number for this exam.

The plant was operating at 98% power when a Main Turbine trip causes a reactor scram. The lights in the control room go out for approximately 5.5 seconds and then some of the lights come back on.

Which of the following is correct for these conditions?

A. BKRs S-1, S-2, and S-3 have closed and are providing power.

B. BKRs N-1, N-2, and N-3 have closed and are providing power.

C. SM-7 and SM-8 are powered from DG-1 and DG-2.

D. SM-7 and SM-8 are powered from TR-B.

ANSWER: D

QUESTION TYPE:	SRO/RO
KA # & KA VALUE:	295003AA2.05 3.9/4.2 10CFR55.41, 43 - Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: Whether a partial or complete loss of AC power has occurred
REFERENCE:	LO000182 rev. 12, page 30
SOURCE:	**BANK QUESTION – 2001 NRC EXAM –** SRO T1, GP1, #2 RO T1, GP2, #4
LO:	5047 – State the open/closed status of the N and S breakers for SM1, 2, 3, SH5, and SH6 for: MT trip and TRS loss.
RATING:	H3
ATTACHMENT:	NONE
JUSTIFICATION:	A is incorrect because the control room lights would not have gone out it the S BKRs had closed. B is incorrect be cause the N BKRs are the normal supply when the main turbine is on the line. C is incorrect because the light would have been out for at least 7 seconds if the DGs were powering the bus. D is correct because all control room lights, except those powered from MC-7C and 7E, and 8C and 8E would come back on when SM-7 and SM-8 were repowered.
COMMENTS:	

QUESTION # 26 EXAM KEY
10/04/2002
ex02013

Which of the following systems is designed to maintain control room temperature at 75°F ± 3°F for personnel habitability if the **normal** cooling supply system **fails**?

A. Radwaste Chilled Water (WCH)

B. Control Room Emergency Chilled Water (CCH)

C. Standby Service Water (SW)

D. Turbine Service Water (TSW)

ANSWER: B

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 290003K3.01 3.5/3.8 10CFR41 & 45 - Knowledge of the effect that a loss or malfunction of the CONTROL ROOM HVAC will have on following: Control room habitability.

REFERENCE: LO000201 rev. 9, pages 8, 19, and 20

SOURCE: **NEW QUESTION –** SRO T2, GP2, #13 RO T2, GP2, #18

LO: 5221 – State the purpose of each of the following system components: k. Control room emergency chillers

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: A is incorrect because Radwaste Chilled water is the normal source for cooling. C is incorrect because Service Water will only maintain control room temperature at 104 °F for equipment. D is incorrect because TSW by itself does not cool the control room. B, Control room Emergency Chilled Water is the backup source of cooling water for personnel habitability. B is correct.

COMMENTS:

A plant startup is underway with the A2 sequence selected. All rod withdrawals in RSCS groups 1-4 have been completed.

Which of the following is the correct control rod density for this condition?

A. 25%

B. 50%

C. 75%

D. 100%

ANSWER: B

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 201004K5.02 3.1/3.3 10CFR55.41.5/45.3 - Knowledge of the operational implications of the following concepts as they apply to ROD SEQUENCE CONTROL SYSTEM: Sequences and groups

REFERENCE: LO000160 rev 10, page 7

SOURCE: **BANK QUESTION – 2001 NRC EXAM –** SRO T2, GP2, #2 RO T2, GP2, #2

LO: 5806 – Explain the following term: Control rod density

RATING: H2

ATTACHMENT: NONE

JUSTIFICATION: Completion of the 1st 4 RSCS rod groups places 50% of the rods in the full out position. This is 50% rod density. B is correct.

COMMENTS:

The plant is operating at 97% power when a radioactive spill occurs in the Radwaste Building. WEA-RIS-14 indicates 1.39E5 cpm and is stable. It will take at least 90 minutes before a plan is in place to begin work on the cleanup.

Which of the following is correct for these conditions?

A. Declare an Alert.

B. Declare a Site Area Emergency.

C. A four hour notification to the NRC is required.

D. A one hour notification to the NRC is required.

ANSWER: D

QUESTION TYPE: SRO

KA # & KA VALUE: 288000 2.4.30 2/3.6 10CFR55.43.5 & 45: Plant Ventilation Systems – Knowledge of which events related to system operations/status should be reported to outside agencies.

REFERENCE: PPM 1.10.1 rev. 22, page 9, PPM 13.1.1 rev. 31, pages 19 & 35

SOURCE: **NEW QUESTION –** SRO T2, GP3, #3

LO: 6008 – State the notification requirements to State and Local Government agencies after declaring an Emergency Classification at Columbia.

RATING: H3

ATTACHMENT: **YES – PPM 13.1.1 rev. 31 pages 19 & 35 PPM 1.10.1 rev. 22, page 9**

JUSTIFICATION: The value given for WEA-RIS-14 is greater than the threshold for a UE if the release is expected to last more than an hour. Since it will take longer than 1 hour for a work plan, a UE must be declared. By declaring a UE, a 1 hour notification is required. D is correct.

COMMENTS:

A plant transient occurred approximately two minutes ago. The reactor had been operating at 99% power for the last six months. The following conditions exist:

- Reactor level is 56 inches on the Narrow Range
- Reactor pressure is cycling between 1080 psig and 1094 psig
- HPCS and RCIC have not initiated
- No operator actions have been taken except the immediate scram actions

Which of the following is correct concerning these conditions?

The transient was caused by …

A. an APRM A INOP with a coincident loss of RPS-B.

B. a main generator trip from phase differential current.

C. a coincident loss of both SM-7 and SM-8.

D. a loss of SM-1.

ANSWER: C

QUESTION TYPE:	SRO
KA # & KA VALUE:	295006AA2.04 4.1/4.1 10CFR55.41,. 43.5, and 45 - Ability to determine and/or interpret the following as they apply to SCRAM: Reactor Pressure
REFERENCE:	LO000128 rev.8, page 4
SOURCE:	**NEW QUESTION –** SRO T1, GP1, #3
LO:	7682 – Describe the physical connection and/or cause and effect relationship between RPS and the following: MSIVs
RATING:	H3
ATTACHMENT:	NONE
JUSTIFICATION:	From the information given it can be deduced that the MSIVs are isolated – pressure is at about 1090 psig and cycling. Reactor level is given at greater than +60 inches with neither HPCS nor RCIC having been initiated. A is incorrect because there would be no MSIV isolation at this point. B would cause a regular scram and no isolation. D would cause a loss of feedwater and an auto start of both HPCS and RCIC. A loss of both SM-7 and SM-8 would cause an immediate MSIV isolation and scram. C is correct.
COMMENTS:	After further research, the question stands as written. This is the response of the simulator under these conditions.

The plant has been operating at 99% power for the past 8 months. One SRV is leaking. Suppression pool temperature is 84°F and going up.

Which of the following is the reason to maintain the suppression pool temperature LT the Tech Spec LCO?

A. Ensures peak containment temperature and pressure are LT the maximum allowable limits during a DBA LOCA.

B. The wetwell to drywell interface integrity will not be challenged during the DBA LOCA..

C. Code allowable stresses on the Wetwell Floor will not be exceeded during an emergency depressurization.

D. SRV Tailpipe integrity will not be challenged during an emergency depressurization.

ANSWER: A

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 295013AK3.01 3.6/3.8 10CFR55.41 & 45 - Knowledge of the reasons for the following responses as they apply to HIGH SUPPRESSION POOL TEMPERATURE: Suppression pool cooling operation

REFERENCE: PPM 5.0.10 rev. 6, page 248 TS 3.6.2.1 and bases pages B 3.6.2.1-1 & 2

SOURCE: **NEW QUESTION –** SRO T1, GP1, #10 RO T1, GP2, #7

LO: 6925 - Identify the basis for any LCO.

RATING: L3

ATTACHMENT: NONE

JUSTIFICATION: A is correct as stated in the bases document. B, C, and D are all incorrect applications of other containment limits.

COMMENTS:

QUESTION # 31 EXAM KEY
10/04/2002
ex01062

The plant was operating at 75% power when a transient occurred causing reactor level to decrease to –7 inches.

Which of the following is correct concerning the initial direction from the CRS?

CRO-1 should be directed to…

A. P603, and Board A.
CRO-2 Board B and Board C
CRO-3 P601 and P602

B. P602, P603, and Board A.
CRO-2 Board B and Board C
CRO-3 P601

C. P602 and P603
CRO-2 Board A, Board B and Board C
CRO-3 P601

D. P603
CRO-2 Board A, Board B and Board C
CRO-3 P601, and P602

ANSWER: B

QUESTION TYPE: SRO

KA # & KA VALUE: 2.1.6 2.1/4.3 10CFR55.43.5/45.12/45.13 – Ability to supervise and assume a management role during plant transients and upset conditions.

REFERENCE: PPM 1.3.1 rev 56, page 28

SOURCE: **BANK QUESTION – 2001 NRC EXAM –** SRO T3, #1

LO: 6092 – State the responsibilities of the Control Room Supervisor.

RATING: L2
ATTACHMENT: NONE
JUSTIFICATION: PPM 1.3.1 states that the initial response for the CROs as those in B. B is correct.
COMMENTS:

The plant was operating at 99% power when an accident occurred. Core conditions have caused fuel damage. A release has been in progress for the last 23 minutes with TEA-RIS-13 indicating 4.6E4 cpm.

Which of the following is correct concerning these conditions?

A. Declare a General Emergency

B. Declare a Site Area Emergency

C. Enter PPM 5.1.1 RPV Control

D. Enter PPM 5.4.1 Radioactivity Release Control

ANSWER: D

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 295017AA2.03 3.1/3.9 10CFR55.41, 43.5, & 45.8 - Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE: Radiation levels

REFERENCE: PPM 13.1.1 rev. 31 pages 19 & 35 PPM 5.0.10 rev. 6 age 302

SOURCE: **NEW QUESTION –** SRO T1, GP1, #16 RO T1, GP2, #10

LO: 8017 – Given plant conditions, recognize an EOP entry condition and enter the appropriate flow chart

RATING: H3

ATTACHMENT: **YES -** PPM 13.1.1 rev. 31 pages 19 & 35

JUSTIFICATION: The indication given for TEA-RIS-13 requires that an Alert be declared. PPM 5.4.1 entry is required when the release rate exceeds the Alert level. D is the correct answer.

COMMENTS:

QUESTION # 33 EXAM KEY
10/04/2002
ex00121

A plant startup is underway. The following conditions exist:

APRM A, E, D	11%
APRM C and B	13%
APRM F	out of service – bypassed
IRM indications	25 to 35 on R10 for A, C, E, B, F, and H
IRM G	out of service – bypassed
IRM D	41 on R10
Reactor pressure	819 psig

Which of the following is the correct decision concerning these conditions?

A. Do not place the Mode Switch in RUN; a scram will occur from APRM C and D.

B. Do not place the Mode Switch in RUN; a MSL isolation will occur.

C. Place the Mode Switch in RUN; a mode change to RUN is allowed with at least 2 APRMs per trip system above 5%.

D. Place the Mode Switch in RUN; a mode change to RUN is allowed with at least 3 IRMs per trip system LE 40 on R10

ANSWER: B

QUESTION TYPE:	SRO/RO
KA # & KA VALUE:	2.2.2 4.0/3.5 10CFR55.45 – Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels.
REFERENCE:	LO000128 rev. 8, page21
SOURCE:	**BANK QUESTION – 2000 NRC EXAM –** SRO T3, #8 RO T3, #5 **WNP-2 LER 84-108**
LO:	NO LO
RATING:	H2
ATTACHMENT:	NO
JUSTIFICATION:	C and D are incorrect because the MS to run causes a full isolation and RX Scram. A is incorrect because the APRM scram setpoint is 15% with the MS in STARTUP. B is correct because MSL pressure of less than 831 psig in RUN causes a full MSIV isolation and reactor scram.

The plant is operating at 99% power with SM-4 de-energized and tagged out for maintenance. COND-P-2A trips due to a lockout.

Assuming no operator actions, which of the following is correct for these conditions?

A. RCIC initiates and injects until reactor level increases to +54 inches, RCIC-V-45 and 13 close until reactor level returns to -50 inches, where RCIC injects again.

B. HPCS initiates and injects until reactor level increases to +54 inches, HPCS-V-4 closes until reactor level returns to -50 inches, where HPCS injects again.

C. Recirc pumps runback to 30 hertz on low reactor water level and the feedpumps return reactor level to the normal operating band.

D. RHR injects and fills the RPV following an Automatic Depressurization System initiation.

ANSWER: A

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 295031EK2.04 4.0/4.1 10CFR55.41 & 45 - Knowledge of the interrelations between REACTOR LOW WATER LEVEL and the following: Reactor Core Isolation Cooling

REFERENCE: LO000180 rev. 11 pages 20 & 21

SOURCE: **NEW QUESTION –** SRO T1, GP1, #22 RO T1, GP1, #10

LO: 5714 – List the signal and setpoints that will cause a RCIC System initiation.

RATING: H2

ATTACHMENT: NONE

JUSTIFICATION: The loss of COND-P-2A causes a loss of feedwater (loss of feedpumps). Reactor level decreases to less than –50 inches and initiates RCIC. Because SM-4 is tagged out, HPCS does not start. RCIC returns level to +54 inches and RCIC-V-13 & 45 close. Reactor level does not decrease to the point of ADS/RHR initiation. A is correct.

COMMENTS:

You have assigned a maintenance person to repair a valve inside a valve room which is a posted RADIATION AREA. The job is projected to take four hours. The maintenance person has 1.985 rem TEDE this year.

Which of the following is correct for these conditions?

The maintenance person is allowed to work on this valve for a **maximum** of…

A. 1 hour

B. 2 hours

C. 3 hours

D. 4 hours

ANSWER: C

QUESTION TYPE: SRO

KA # & KA VALUE: 2.3.1 2.6/3.0 10CFR55.41.2/43.4/45.9/45.1 – Knowledge of 10 CFR 20 and related facility radiation control requirements

REFERENCE: GEN-RPP-07 rev. 3, page 7

SOURCE: **BANK QUESTION #98126 – MODIFIED –** SRO T3, #13

LO: 6013 – Define radiation area.

RATING: H2

ATTACHMENT: NONE

JUSTIFICATION: The Radiation area minimum threshold is 5 mr/hr. The **maximum** amount of time available to a worker for a 15 mr limit (2 rem admin limit) would be 3 hours in the area.

COMMENTS:

The plant is operating at 95% power with suppression pool level slowly and steadily going up. All systems are in the normal standby lineup.

Which of the following is the cause of the increasing suppression pool level?

A seat leak on …

A. RCIC-V-22 & 59 – Test Bypass

B. HPCS-V-23 – Full Flow Test

C. RHR-V-24B – Full Flow Test

D. LPCS-V-12 – Test Return

ANSWER: B

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 209002K1.03 3.0/3.0 10CFR55.41 & 45 - Knowledge of the physical connections and/or cause- effect relationships between HIGH PRESSURE CORE SPRAY SYSTEM (HPCS) and the following: water leg pump

REFERENCE: LO000174 rev. 9, fig. 1 LO000192 rev. 9, fig. 1 LO000198 rev. 10, fig. 1 LO000180 rev. 11, fig. 13

SOURCE: **NEW QUESTION –** SRO T2, GP1, #3 RO T2, GP1, #6

LO: NO LO

RATING: H2

ATTACHMENT: NONE

JUSTIFICATION: C and D are incorrect because both systems take the water leg pump suction from the suppression pool, and a leak would cause no net change in suppression pool level. A is incorrect because the RCIC is normally lined up with suction to the CSTs, but the Test Bypass line goes to the CSTs and not the suppression pool. B is correct because HPCS is normally lined up with the suction from the CSTs and a leak through the Full Flow Test valve would result in a net increase in suppression pool volume and level.

QUESTION # 38 EXAM KEY
10/04/2002
COMMENTS:

The plant is currently operating at 87% power. While performing a surveillance on Monday, MS-LIS-37A Level 1 input to ECCS Div 1 was determined to be inoperable at 1500. All required actions were taken. At 0300 on Tuesday, MS-LIS-37B, Level 1 input to ECCS Div 2 was found to be set at –144.5 inches.

Concerning these conditions, which of the following is correct?

A. No further action is required, MS-LIS-37B is set within tolerance.

B. Declare HPCS inoperable in 1 hour and place the channel in the trip condition within 24 hours.

C. Declare supported features inoperable within 1 hour, declare HPCS inoperable in 1 hour, and place the channel in the trip condition within 24 hours.

D. Declare supported features inoperable within 1 hour and place the channel in the trip condition within 24 hours.

ANSWER: D

QUESTION TYPE:	SRO
KA # & KA VALUE:	216000 2.2.25 2.5/3.7 10CFR55.43.2 – Knowledge of bases in Technical Specifications for limiting conditions for operations and safety limits: Concerning Nuclear Boiler Instrumentation.
REFERENCE:	TS 3.3.5.1 pages –1, -2, -8, -9, & -10 and bases pages B 3.3.5.1 –9, -10, -27, -28, & -29
SOURCE:	**NEW QUESTION –** SRO T2, GP1, #9
LO:	10306 – With Tech Specs provided, determine LCO compliance for a given operational condition.
RATING:	H4
ATTACHMENT:	**YES -** TS 3.3.5.1 pages –1, -2, -8, -9, & -10 and bases pages B 3.3.5.1 –9, -10, -27, -28, & -29
JUSTIFICATION:	MS-LIS-37B is out of spec and it is the redundant initiation capability per TS. Because of this, the supported features must be declared inop within 1 hour and the channel must be placed in the trip condition within 24 hours. The HPCS initiation function is not affected so HPCS does not have to be declared inop. D is correct.
COMMENTS:	

QUESTION # 40 EXAM KEY
10/04/2002
ex02021

The plant was operating at 99% power when loss of feedwater occurred. RCIC is in operation from its alternate source of water, and has restored reactor level to normal. The CRO reports RCIC discharge flow and discharge pressure are starting to oscillate and getting worse.

Which of the following describes the cause of these indications?

A. Low CST level

B. Low reactor water level

C. Low reactor pressure

D. Low suppression pool level.

ANSWER: D

QUESTION TYPE:	SRO/RO
KA # & KA VALUE:	217000K5.01 2.6/2.6 10CFR55.41 & 45 - Knowledge of the operational implications of the following concepts as they apply to REACTOR CORE ISOLATION COOLING SYSTEM (RCIC): Indications of pump cavitation
REFERENCE:	General Physics BWR Gen. Fundamentals Chapter 2: Pumps pages 21 & 22
SOURCE:	**BANK QUESTION – 2000 NRC EXAM -** SRO T2, GP1, #11 RO T2, GP1, #16
LO:	7145 – Describe cavitation, including symptoms, effects on centrifugal pump operation and methods of prevention.
RATING:	H2
ATTACHMENT:	NONE
JUSTIFICATION:	GP Gen. Fun. Chapter 4 describes the effect of cavitation as an oscillation of discharge pressure. Cavitation is caused by a loss of NPSH. NPSH is the difference between the pressure on the suction side of the pump and the saturation pressure of the liquid being pumped. D is correct. It is the only possibility that lowers the suction pressure of the RCIC pump.
COMMENTS:	This question is similar to ex00053 which was used on the 2000 ILC exam. The question stem has been only changed enough to use it for the RCIC system. This is not considered a MODIFIED BANK question, but it was given a new number.

QUESTION # 41 EXAM KEY
10/04/2002
ex02022

With the plant at 100% power, which of the following would indicate a leaking Main Steam Relief Valve?

A. Elevated SRV Acoustic Monitor indication.

B. Elevated SRV Tailpipe Temperature indication.

C. An SRV with a red indicating light on P628.

D. An SRV with a red indicating light on P631

ANSWER: B

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 239002K4.06 3.5/3.7 10CFR55.41 - Knowledge of RELIEF/SAFETY VALVES design feature(s) and/or interlocks which provide for the following: Detection of valve leakage

REFERENCE: LO000129 rev. 8, pages 10 - 12

SOURCE: **NEW QUESTION –** SRO T2, GP1, #17 RO T2, GP1, #21

LO: 5528 – Describe the physical connection and/or cause and effect relationship between the SRVs and : Tailpipe thermocouples.

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: C and D are incorrect because these lights only indicate the solenoids are energized and not actual valve position or leakage. A is incorrect because the Acoustic Monitoring system was removed and is no longer in service. B is correct because an elevated temperature indicates steam flow through the valve.

COMMENTS:

The plant is in MODE 4. Maintenance was performed on the breaker for ROA-FN-1A and it is now ready for a test start. The following conditions exist:

The breaker is racked in and the fuses are installed.

The CRO places the control switch to start on both ROA-FN-1A and REA-FN-1A. He notes there is no control room indication of a start on ROA-FN-1A.

He places both control switches in the stop position and notes the green light is out for ROA-FN-1A and illuminated for REA-FN-1A.

The panel lights are checked and are found to be OK.

Concerning these conditions, which of the following is true?

A. Reactor building pressure increases to 4.7 inches of H2O and stabilizes.

B. Reactor building pressure increases to 4.0 inches of H2O and stabilizes.

C. Reactor building pressure increases to 4.7 inches of H2O and all Reactor Building Ventilation fans trip. Pressure returns to 0 inches of H2O

D. Reactor building pressure increases to 4.0 inches of H2O and all Reactor Building Ventilation fans trip. Standby Gas Treatment auto starts and maintains reactor building pressure at -.25 inches of H2O.

ANSWER:	A
QUESTION TYPE:	SRO/RO
KA # & KA VALUE:	290001A1.01 3.1/3.1 10CFR55.41 & 45 - Ability to predict and/or monitor changes in parameters associated with operating the SECONDARY CONTAINMENT controls including: System Lineups
REFERENCE:	82-RSY-1000-T6 rev. 8, pages 5, 12, & 13 EWD-80E-001 WNP-2 LER 88-007
SOURCE:	**NEW QUESTION –** SRO T2, GP1, #23 RO T2, GP2, #17
LO:	5677 – State the purpose of the alternate relief path damper.
RATING:	H3
ATTACHMENT:	NONE
JUSTIFICATION:	The information given indicates a blown or improperly installed trip fuse. The result of starting a fan with no trip circuit is a running fan that cannot be stopped from the control switch. In this case pressure would increase until both Div 2 fans trip and continue to increase pressure until the suction damper opened fully and limited pressure to 4.7 inches of H2O. A is correct.

The plant is operating at 95% power when several alarms are received on P603 including a Rod Out Block. On closer inspection the CRO notes that the UPSC, INOP, and DNSC indicating lights are illuminated on P603 for the A Rod Block Monitor.

Which of the following caused these indications?

A. Loss of RPS B.

B. Loss of RPS A.

C. An edge rod was selected.

D. RBM A was bypassed with the joystick.

ANSWER: B

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 215002K6.01 3.0/3.2 10CFR55.41 & 45 - Knowledge of the effect that a loss or malfunction of the following will have on the ROD BLOCK MONITOR SYSTEM: RPS

REFERENCE: 82-RSY-0800-T1 rev. 8, pages 17 & 18

SOURCE: **NEW QUESTION –** SRO T2, GP2, #9 RO T2, GP2, #9

LO: 7667 – Predict the effects the following failures have on the RBM System: RPS

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: The RBM System is divisionally powered from RPS. A loss of RPS-A causes the given indications. Selecting an edge rod or bypassing with the joystick does not result in a rod block. B is correct.

COMMENTS:

The plant is in Mode 5 with the refuel bridge (unloaded) over the spent fuel pool and refueling operations in progress. The Mode Switch is placed in the START/HOT STANDBY Position.

Which of the following is the result of this action?

The Refuel Bridge automatically stops…

A. only when it is over the reactor cavity.

B. at any location over the Spent Fuel Pool or the Reactor Cavity.

C. as it approaches the Reactor Cavity from the Spent Fuel Pool.

D. only as it approaches the Reactor Cavity from the Spent Fuel Pool when loaded with a fuel bundle.

ANSWER: C

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 2.4.46 3.5/3.6 10CFR55.43 & 45 – Ability to verify alarms are consistent with the plant conditions

REFERENCE: LO000207 rev 8, page 25 & 26

SOURCE: **BANK QUESTION – 2001 NRC EXAM –** SRO T3, # 16 RO T3, #12

LO: 5360 – List and explain all the conditions pertaining to fuel handling equipment that can cause a control rod block when the reactor mode switch in either the REFUEL position or the START/HOT STBY position.

RATING: L3

ATTACHMENT: NONE

JUSTIFICATION: Placing the Mode Switch to START/HOT STANDBY causes the refuel bridge to stop when the bridge is on the refueling switch #2. This prevents the bridge from traveling over the core with the Mode Switch in START/HOT STANDBY. A and B are both incorrect because the bridge is not limited when over the core or in the Spent Fuel Pool. D is incorrect because the hoist does not have to be loaded for the bridge to stop. C is correct.

QUESTION # 45 EXAM KEY
10/04/2002
COMMENTS:

QUESTION # 46 EXAM KEY
10/04/2002
ex02025

Preparations are in progress for a plant startup. The following conditions exist:

- RCIC is out of service for repair for the next two hours
- MS-LIS-24C (reactor level 3) is set at 9.1 inches
- All other plant equipment and conditions are normal

You have been directed to begin the startup.

Which of the following is correct concerning these conditions?

A. The Mode Switch cannot be placed in STARTUP until the MS-LIS-24C setpoint is returned to an allowable value.

B. The Mode Switch cannot be placed in STARTUP until RCIC is returned to service.

C. Place the Mode Switch in STARTUP, MS-LIS-24C is not required to be within specifications in MODE 2.

D. Place the Mode Switch in STARTUP, RCIC is not required to be in service until steam dome pressure is GT 150 psig.

ANSWER: A

QUESTION TYPE:	SRO
KA # & KA VALUE:	295009 2.2.24 2.6/3.8 10cfr55.43.2 & 45 - Ability to analyze the affect of maintenance activities on LCO status.
REFERENCE:	PPM 4.603.A7 rev. 28, page 15 Tech Spec 3.3.1.1 pages –1 & -8 TS 3.0.4 pages 3.0-1 & -2
SOURCE:	**NEW QUESTION –** SRO T1, GP1, #6
LO:	10308 – With Tech Specs provided and given an operational condition, determine if a Mode change or other change in plant condition is allowed.
RATING:	H3
ATTACHMENT:	**YES -** PPM 4.603.A7 rev. 28, page 15 Tech Spec 3.3.1.1 pages –1 & -8 TS 3.0.4 pages 3.0-1 & -2
JUSTIFICATION:	A is correct because placing the Mode Switch in STARTUP at this point relies on an action statement for the out of spec level switch (MS-LIS-24C), which is not allowed by TS 3.0.4. B is incorrect because RCIC is not required to be operable until steam dome pressure is GT 150 psig. C and D are incorrect because the Mode Switch cannot be placed in STARTUP until MS-LIS-24C calibration is within spec.

Which of the following is designed to prevent secondary containment over-pressurization during a postulated piping break in the area between the Drywell and the Turbine Building?

A. Standby Gas Treatment.

B. Reactor Building Ventilation.

C. Reactor Building Blowout Panels.

D. Main Steam Tunnel Blowout Panels.

ANSWER: D

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 295035EK1.02 3.7/4.2 10CFR55.41 - Knowledge of the operational implications of the following concepts as they apply to SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE: Radiation release

REFERENCE: LO000139 rev. 9, page 6

SOURCE: **BANK QUESTION – 2001 NRC EXAM –** SRO T1, GP2. #14 RO T1, GP3, #3

LO: 7003 – State the actions that occur on a Main Steam tunnel High Pressure.

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: An over pressurization in the Main Steam Tunnel results pressure relief through the MST Blowout panels. This relief minimizes the damage to the Secondary Containment and limits radioactive releases from any other part of the Sec. Cont. D is correct.

COMMENTS:

A reactor scram just occurred. Several control rods do not indicate full-in

How are these control rods identified on the Rod Worth Minimizer?

A. A list containing each rod not full in by it's numerical position (i.e. 30-31).

B. Both a list containing each rod not-full-in by it's numerical position (i.e. 30-31) and a full core display with rods not-full-in indicated by • • (2 dots).

C. A full core display map with rods full-in indicated by ++ and rods not full in indicated by >>.

D. A full core display map with rods full-in indicated by >> and rods not full in indicated by ++.

ANSWER: A

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 295015AK2.05 2.6/2.8 10CFR55.41 & 45 - Knowledge of the interrelations between INCOMPLETE SCRAM and the following: Rod Worth Minimizer

REFERENCE: LO000154 rev. 10, pages 12, 13, 16, & fig. 10

SOURCE: **NEW QUESTION –** SRO T1, GP1, #12 RO T1, GP1, #6

LO: 5908 – Given a set of RWM displays, explain the following symbols:

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: A is correct because any control rod not full in following a scram automatically is indicated on the RWM by a numerical listing of it's position in the core. B, C, and D are all incorrect because ˙˙ indicated a rod inserted past position 02 in the scram mode, ++ indicates a rod fully withdrawn, and > > is not a RWM indication.

COMMENTS:

The plant is shutdown. RHR has been lined up for suppression pool cooling and RCIC is running for level control, when S2-1 is de-energized.

Which of the following describes the effect on plant operation?

A. CN-V-65 auto closes and isolates the N2 supply to the containment.

B. TO-P-BOP main turbine bearing oil pump trips and trips the MT turning gear.

C. RCC-V-6 auto closes and isolates the non-drywell RCC loads.

D. RCIC will continue to operate and provide level control.

ANSWER: D

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 295004AK2.03 3.3/3.3 10CFR55.41 & 45 - Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF D.C. POWER and the following: DC bus loads

REFERENCE: ABN-ELEC-250VDC rev. 0, page 5

SOURCE: **BANK QUESTION – 98 NRC EXAM** SRO T1, GP2, #3 RO T1, GP2, #5

LO: 7657 – Predict the effect a failure of 250 VDC bus S2-1 will have on RCIC.

RATING: H3 H4

ATTACHMENT: NONE

JUSTIFICATION: A is incorrect because CN-V-65 is blocked open. B and C are both incorrect because neither pierce of equipment is powdered from 250 VDC. D is correct because the loss of S2-1 will not prevent RCIC from operating.

COMMENTS:

The plant is operating at 78% power when a total loss of DP-S1-2 occurs. Shortly thereafter a DBA LOCA occurs.

Which of the following is correct concerning these conditions?

A. HPCS-P-1 does not start due to a loss of power to the initiation logic and breaker.

B. RHR-P-2B does not start due to a loss of power to the initiation logic and breaker.

C. RHR-P-2A starts but does not inject due to he loss of power to the injection valve.

D. LPCS-P-1 starts but does not inject due to the loss of power to the injection valve.

ANSWER: B

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 203000K2.03 2.7/2.9 10CFR55.41 - Knowledge of electrical power supplies to the following: Initiation logic

REFERENCE: EWD-9E-002, 47E-003, & 47E-003A

SOURCE: **NEW QUESTION –** SRO T2, GP1, #2 RO T2, GP1, #3

LO: 7653 – Predict the effects of a failure of 125VDC bus S1-21 will have on RHR.

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: A is incorrect because HPCS logic is from Div 3 power. C and D are both incorrect because the injection valve is AC powered and will open. B is correct because of the loss of control power.

COMMENTS:

The reactor is at rated power with TIP Channel C inserted for LPRM calibration when a loss of SM-1 occurs.

Assuming no operator action, which of the following is correct?

A. The TIP drive continues to insert the detector to the Core Top Limit and completes the Tip trace. The detector then withdraws into the shield chamber and the ball valve closes.

B. The squib valve for TIP channel C fires immediately isolating the drive mechanism.

C. The inserted TIP detector withdraws into the shield chamber, the ball valve closes, and TIP-V-15, Tip Purge Isolation Valve closes.

D. The inserted TIP detector stops until power is restored to SM-1 and then completes the TIP trace.

ANSWER: C

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 215001K4.01 2.9/3.3 10CFR55.41 - Knowledge of TRAVERSING IN-CORE PROBE design feature(s) and/or interlocks which provide for the following: Primary containment isolation.

REFERENCE: LO000155 page 13 and 16

SOURCE: **BANK QUESTION – 2000 NRC EXAM** - SRO T2, G3, #2 RO T2, G3, #1

LO: 6989 – Explain the TIP system response to a LOCA signal.

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: A is incorrect because the drive does not continue for a complete trace. B is incorrect because the squib does not fire on and isolation signal. D is incorrect because the loss of power has no direct affect on the TIP drive.

COMMENTS:

The plant is in MODE 2 with a control rod withdrawal in progress. An LPRM failure has caused an I&C technician to take the LPRM mode selector switch out of OPERATE. This results in 13 LPRMS assigned to APRM A.

Which of the following is correct for these conditions?

A. A rod block is applied by the RMCS. Rod withdrawal can continue following bypass of APRM A.

B. A rod block is applied by the RMCS. Rod withdrawal can continue following bypass of the failed LPRM.

C. No rod block is applied by the RMCS for this failure.

D. No rod block is applied by the RMCS until rod movement is attempted.

ANSWER: A

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 215005K3.03 3.3/3.3 10CFR55.41 & 45 - Knowledge of the effect that a loss or malfunction of the APRM/LPRM will have on following: Reactor Manual Control System

REFERENCE: 82-RSY-0500-T2 rev. 8, page18 LO000148 rev. 10, page 14

SOURCE: **NEW QUESTION –** SRO T2, GP1, #8 RO T2, GP1, #13

LO: 5795 – Identify the conditions that will cause rod blocks: d. RMCS Block

RATING: H2

ATTACHMENT: NONE

JUSTIFICATION: Less than 14 LPRMs in OPERATE to an APRM results in an APRM Inop and a control rod out block applied by the RMCS. Taking the MODE switch out of operate for the failed LPRM caused the APRM inop. A is correct.

COMMENTS:

QUESTION # 54 EXAM KEY
10/04/2002
ex02029

The plant was operating at 75% power when an ATWS with a full MSIV isolation occurred. Suppression chamber pressure is 27 psig and up, with drywell pressure trending up approximately 0.5 psig less than suppression chamber.

Which of the following is correct concerning these conditions?

A. The wetwell to drywell vacuum breakers are covered with water.

B. The wetwell to drywell vacuum breakers are working correctly.

C. A drywell downcomer has failed below the water level in the suppression pool.

D. The drywell to suppression chamber interface has ruptured.

ANSWER: B

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 223001A3.02 3.4/3.4 10CFR55.41 & 45 - Ability to monitor automatic operations of the PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES including: Vacuum breaker relief valve operation.

REFERENCE: LO000127 rev. 10, page 8

SOURCE: **NEW QUESTION –** SRO T2, GP1, #14 RO T2, GP1, #20

LO: 5636 – State the purpose of the following major components: wetwell to drywell vacuum breakers.

RATING: H2

ATTACHMENT: NONE

JUSTIFICATION: The information given indicates a leak in an SRV downcomer. Suppression chamber pressure is increasing above drywell pressure by 0.5 psig which indicates that the wetwell to drywell vacuum breakers are working correctly. B is correct.

COMMENTS:

Reactor building ventilation has failed. The CRO has started Standby Gas Treatment by placing the control switch for SGT-EHC-1B-2 in the ON position.

Which of the following is correct concerning these conditions?

A. Room discharge valve (SGT-V-4A2), auto opens.

B. The associated Elevated release discharge valve (SGT-V-5A1), auto closes.

C. The incoming gas temperature is raised by the electric heaters to reduce humidity and increase efficiency of the moisture separators.

D. The incoming gas temperature is raised by the electric heaters to reduce humidity and increase efficiency of the charcoal adsorbers.

ANSWER: D

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 261000A1.07 2.8/2.9 10CFR55.41 & 45 - Ability to predict and/or monitor changes in parameters associated with operating the STANDBY GAS TREATMENT SYSTEM controls including: SGTS train temperature;.

REFERENCE: 82-RSY-0400-T3 rev. 10, pages 3-5 & 10

SOURCE: **NEW QUESTION –** SRO T2, GP1, 19 RO T2, GP1, #25

LO: 5825 – State the purpose of the SGT electrical heating coils.
5829 – Describe the SGT System response to operation of the heater control switch.

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: A and B are incorrect because these valves act in opposite direction as given. D is incorrect because the moisture separators are upstream of the heaters and take moisture out prior to the heating coils. D is the stated purpose of the electric heating coils.

COMMENTS:

QUESTION # 56 EXAM KEY
10/04/2002
ex02030

What is the purpose of sequencing loads on DG-1 and 2?

A. Prevents the possibility of over speeding the diesel.

B. Prevents a generator loss of excitation trip.

C. Reduces the automatically connected loads.

D. Reduces the tendency for the diesel engine to run under “souping” conditions.

ANSWER: C

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 264000K5.06 3.4/3.4 10CFR55.41 & 45 - Knowledge of the operational implications of the following concepts as they apply to EMERGENCY GENERATORS (DIESEL/JET): Load sequencing

REFERENCE: LO000200 rev. 8, pages 40 & 41

SOURCE: **NEW QUESTION –** SRO T2, GP1, #21 RO T2, GP1, #27

LO: NO LO

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: A and B are both incorrect because load sequencing would have no effect on the possibility of over speeding the DG or preventing a loss of excitation trip. D is incorrect because souping is prevented by not operating the DG for long periods of time under lightly loaded conditions. C is correct as given in the systems text.

COMMENTS:

Which of the following design features acts first to limit Secondary Containment overpressure?

A. Opening of ROA-AD-5, ROA suction relief damper.

B. Trip of the operating Standby Gas Treatment fans.

C. Secondary Containment (606 elevation) blowout panels.

D. Trip of reactor building supply and exhaust fans.

ANSWER: A

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 290001K4.02 3.4/3.5 10CFR55.41 - Knowledge of SECONDARY CONTAINMENT design feature(s) and/or interlocks which provide for the following: Protection against over pressurization.

REFERENCE: 82-RSY-1000-T6 rev. 8, pages 5 & 12 – 13 LO000139 rev.9, page 4

SOURCE: **NEW QUESTION –** SRO T2, GP1, #22 RO T2, GP2, #16

LO: 5678 – Explain how Reactor Building pressure is maintained by the Reactor Building HVAC system including the controller setpoint.

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: B is incorrect because there is no SGT trip on high pressure. C and D are both incorrect because the blowout panels and the fan trips occur at a higher pressure than the suction damper actuations. A is correct.

COMMENTS:

The plant is operating at 99% power when both ROD ACCUMULATOR TROUBLE and SCRAM VALVE PILOT AIR HEADER PRESSURE LOW annuciators alarm simultaneously.

Which of the following is correct for this condition?

Annunciator…

A. ROD ACCUMULATOR TROUBLE should be responded to first because the setpoint for this alarm is an LCO entry.

B. ROD ACCUMULATOR TROUBLE should be responded to first because this alarm indicates an unscramable control rod.

C. SCRAM VALVE PILOT AIR HEADER PRESSURE LOW should be responded to first because a full reactor scram may occur without operator response.

D. SCRAM VALVE PILOT AIR HEADER PRESSURE LOW should be responded to first because control rods may not scram if scram air header pressure decreases too far.

ANSWER: C

QUESTION TYPE: SRO
KA # & KA VALUE: 201001 2.4.45 3.3/3.6 10CFR55.43.5 & 45 – Concerning the CRDH system, the ability to prioritize and interpret the significance of each annunciator or alarm.
REFERENCE: PPM 4.603.A7, drop6-7, rev. 28 page 70 PPM 4.603.A8, drop 6-4, rev. 18, page 58
SOURCE: **NEW QUESITON –** SRO T2, GP2, #1
LO: NO LO
RATING: H3
ATTACHMENT: NONE
JUSTIFICATION: A and B are both incorrect because this annunciator is neither an LCO entry nor is it an indication that the control rod will not scram. D is incorrect because without scram pilot valve air header pressure, the control rods will scram. C is correct: with decreasing scram pilot air header pressure, a scram will occur with no operator action and no change in the leakage.

QUESTION # 59 EXAM KEY
10/04/2002
ex02033

The plant is operating at 99% power when a fault occurs. The RWM will no longer display the current control rod position information.

Which of the following caused this indication?

A loss of …

A. RPS-A

B. US-PP

C. IN-2

D. IN-3

ANSWER: B

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 201006A2.01 2.5/2.8 10CFR55.41 & 45 - Ability to (a) predict the impacts of the following on the ROD WORTH MINIMIZER SYSTEM (RWM); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those: Power supply loss

REFERENCE: LO000154 rev. 10, page 25

SOURCE: **NEW QUESTION –** SRO T2, GP2, #4 RO T2, GP2, #4

LO: 5916 – Describe the physical connection and/or cause and effect relationship between RWM and: e. IN1

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: A loss of power causes the indications given. US-PP is the power supply to the RWM. B is the correct answer, none of the distracters have an effect on the RWM.

COMMENTS:

QUESTION # 60 EXAM KEY
10/04/2002
ex00041

The plant is at 25% power following a maintenance outage for work in the drywell. The Primary Containment is being inerted when the EO reports that the Liquid Nitrogen Storage Tank Level is at 49 inches and down slow on CN-LIS-1. ADS header pressure has been 149 psig for the last four minutes.

Which of the following is correct for these conditions?

A. The CIA programmers placed their respective banks in service but stopped at step 1 and CIA-V-39A and 39B remained open.

B. The CIA programmers placed their respective banks in service and CIA-V-39A and 39B remained open.

C. The CIA programmers placed their respective banks in service but stopped at step 1 and CIA-V-39A and 39B have isolated.

D. The CIA programmers placed their respective banks in service and CIA-V-39A and 39B have isolated.

ANSWER: D

QUESTION TYPE:	SRO/RO
KA # & KA VALUE:	2.3.9 2.5/3.4 10CFR55.43 – Knowledge of the process for performing a containment purge.
REFERENCE:	PPM 2.3.1 rev 39, page 31 LO000156 rev. 6, page 8
SOURCE:	**BANK QUESTION 2000 NRC EXAM – WNP-2 LER 90-022 –** SRO T3, #13 RO T3, #7
LO:	5150 – State the 3 signals that cause auto initiation of the nitrogen bottle bank programmer to maintain system pressure.
RATING:	H2
ATTACHMENT:	NONE
JUSTIFICATION:	LER 90-022 documents the problem of CIA header isolation and inoperability due to inerting the containment with low levels of nitrogen in the nitrogen storage tank. CIA header pressure drops and causes CIA-V-39A and 39B to isolate. D is correct.
COMMENTS:	Question rewritten to make it closed reference. Removed reference from handout. Changed question to ADS pressure.

QUESTION # 61 EXAM KEY
10/04/2002
ex02034

The plant is operating at 33% power when the CRO arms and depresses both MSIV ISOLATION LOGIC B and D pushbuttons

Which of the following is correct for these conditions?

A. All inboard and outboard isolation valves close; MSIVS stay open.

B. All inboard and outboard isolation valves close; MSIVS close.

C. The inboard and outboard MSIVs close.

D. There are **no** isolation valve closures.

ANSWER: A

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 2.4.21 3.7/4.3 10CFR55.43 & 45 – Knowledge of the parameters and logic used to assess the status of safety functions: Radioactivity Release control

REFERENCE: LO000173 rev. 9, page 9

SOURCE: **BANK QUESTION LX00747 – MODIFIED –** SRO T3, #17 RO T3, #13

LO: 5600 – Explain the response to arming and depressing the MSIV isolation logic pushbuttons on P601.

RATING: L3

ATTACHMENT: NONE

JUSTIFICATION: Depressing the B pushbutton, causes an Outboard isolation. Depressing D causes an Inboard isolation. The combination of the 2 pushbuttons does not cause anything other than the combined results of press each individually. A is correct. There is no MSIV motion.

COMMENTS:

QUESTION # 62 EXAM KEY
10/04/2002
ex00099

The reactor was operating at 78% power coming out of a refueling outage. A large steam leak in the drywell caused the following plant conditions:

Wetwell level	39 feet
Drywell pressure	30 psig
Reactor pressure	214 psig
Reactor level	-145 inches and stable

RCIC tripped several minutes ago.

Which of the following caused the RCIC trip?

A. Low reactor level.

B. Isolation from low reactor pressure.

C. Low suction pressure.

D. High exhaust pressure.

ANSWER: D

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 2.1.7 3.7/4.4 10CFR55.55.43 – Ability to evaluate plant performance and make operational judgments base on operating characteristics/ reactor behavior / and instrument interpretation.

REFERENCE: PPM 5.0.10 rev 6, page 70

SOURCE: **BANK QUESTION – #EX00099 – 2000 NRC EXAM–** SRO T3, #3 RO, T3, #1

LO: 5722 - List the RCIC isolation signals and setpoints.
RATING: H3
ATTACHMENT: NONE
JUSTIFICATION: A is incorrect because there is no trip on low reactor level. Low pressure isolation has not yet been reached for B. C is incorrect because suction pressure would be relatively high from the conditions given. D is correct based on the explanation of Caution 4 in the EOPs.
COMMENTS:

PPM 5.1.2 has been entered due to ATWS conditions. Level/power conditions exist.

Which of the following is true?

Reactor water level is lowered to .....

A. lower the height of the fluid columns, thereby reducing natural circulation and reactor power

B. reduce inlet sub-cooling, which reduces the void fraction and reactor power

C. reduce the rate of steam removal from the core, which reduces the void fraction and reactor power

D. increase natural circulation driving head and flow through the core to reduce reactor power

ANSWER: A

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 295031EK1.03 3.7/4.1 10CFR55.41 - Knowledge of the operational implications of the following concepts as they apply to REACTOR LOW WATER LEVEL: Water level effects on reactor power.

REFERENCE: PPM 5.0.10 Rev 6, page 155

SOURCE: **BANK QUESTION – ex98001 – 98 NRC EXAM –** SRO T1, GP1, #21 RO T1, GP1, #9

LO: 7498 – Explain the causes of natural circulation in BWRs.

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: B and C have correct initial conditions, inlet sub cooling and steam production both decrease, but the void fraction would INCREASE from these effects, not decrease. In D, natural circulation head would decrease, not increase.

COMMENTS:

A fire has occurred in the area of Reactor Narrow Range level instrumentation tubing in the reactor building.

Which of the following is correct concerning these conditions?

Reactor water level instrumentation operability must be determined because…

A. heating in the variable leg results in unstable and higher indicated level than actual.

B. heating in the reference leg results in unstable and higher indicated level than actual.

C. water spray from the fire suppression systems causes failures of reactor level instrumentation from operation in non-qualified atmosphere.

D. smoke from the fire causes failures of reactor level instrumentation from operation in non-qualified atmosphere.

ANSWER: B

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 600000AK3.04 2.8/3.4 - Knowledge of the reasons for the following responses as they apply to PLANT FIRE ON SITE: actions contained in the abnormal procedures for plant fire on sight.

REFERENCE: ABN-FIRE rev. 3 page 12

SOURCE: **BANK QUESTION – LX00017 –** SRO T1, GP2, #17 RO T1, GP2, #19
LO: 7166 – Given a potential failure mode for a differential pressure cell used for level indication, describe how indicated level will be affected.
RATING: L2
ATTACHMENT: NONE
JUSTIFICATION: Any heat source place on the reference leg of a level instrument can cause false hi level indications due to the reduction in density of the ref. leg. Heating in the variable leg would cause a false lower indicated level. Smoke and water in the atmosphere would have no impact on the level instrument. B is correct.
COMMENTS:

QUESTION # 65 EXAM KEY
10/04/2002
ex02036

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

- All IRMs are inserted and indicate downscale on Range 1
- All SRMs are inserted - the RETRACT PERMIT light is NOT illuminated for SRM A.
- The mode switch is in the STARTUP/HOT STANDBY position

Which of the following is correct?

If SRM A is selected and a withdraw signal is applied, the detector…

A. does not move and a rod block is generated.

B. does not move and RETRACT NOT PERMITED annunciator illuminates.

C. retracts and a ½ scram is generated on RPS A.

D. retracts and a rod block is generated.

ANSWER: D

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 215004A4.06 3.2-3.1 10CFE55.41 & 48 - Ability to manually operate and/or monitor in the control room: SRM back panel switches, meters, and indicating lights.

REFERENCE: LO000132 rev. 9, page 18 and 19

SOURCE: **NEW QUESTION –** SRO T2, GP1, #6 RO T2, GP1, #12

LO: 5942 – Explain what is meant by SRM RETRACT PERMIT light when energized.

RATING: H2
ATTACHMENT: NONE
JUSTIFICATION: A and B are both incorrect because the detector starts to withdraw as soon as the withdraw signal is initiated. C is incorrect because there is no ½ scram initiated. D is correct. At less than 100 counts, a rod block is generated if a detector is withdrawn.
COMMENTS:

The plant was operating at 99% power when a LOCA occurred. The following conditions exist:

- RPV pressure = 30 psig
- Drywell temperature = 300°F
- All level indicators are swinging 20 to 40 inches
- Control rod 30-31 is stuck full out

Which of the following is the correct procedure for these conditions?

A. 5.1.3 Emergency RPV Depressurization

B. 5.1.4 RPV Flooding

C. 5.1.5 Emergency RPV Depressurization - ATWS

D. 5.1.6 RPV Flooding - ATWS

ANSWER: B

QUESTION TYPE: SRO

KA # & KA VALUE: 216000 2.1.6 10CFR43.5 & 45 – Ability to supervise and assume a management role during plant transients and upset conditions (Nuclear Boiler Instrumentation)

REFERENCE: PPM 5.1.1 RPV Control

SOURCE: **NEW QUESTION –** SRO T2, GP1, #7

LO: 8017 – Given plant conditions, recognize an EOP entry conditions and enter the appropriate flow chart.

RATING: H3

ATTACHMENT: **YES** - PPM 5.1.1 RPV Control

JUSTIFICATION: C and D are both incorrect because one control rod full out does not meet the criteria for an ATWS. With erratic level indications and conditions outside the bounds of saturation conditions for level indications, the direction is given to exit both the level and pressure legs of PPM 5.1.1 and enter PPM 5.1.4. B is correct.

QUESTION # 68 EXAM KEY

10/04/2002

COMMENTS:

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QUESTION # 69 EXAM KEY
10/04/2002
ex00100

The plant is operating at 97% power with a discharge from the Waste Collector Tank to the Circ. Water Blowdown line underway. Process Rad Monitor FDR-RIS-606 (Radwaste Effluent) fails downscale.

Which of the following is correct concerning these conditions?

A. The discharge may continue for up to 30 days provided grab samples are collected and analyzed for radioactivity of at least 10^{-7} microcurie/ml, at least once every 12 hours.

B. The discharge may continue for up to 30 days provided that the discharge flowrate is estimated at least once every 4 hours during the release.

C. Stop the discharge. The discharge may continue when 2 independent samples have been analyzed and 2 technically qualified members of the plant staff have independently verified the release calculations and the discharge valve lineup.

D. Stop the discharge. The discharge may continue when a temporary monitor has been installed and the monitor calibration has been verified by analysis of 2 independent batch samples.

ANSWER: C

QUESTION TYPE: SRO
KA # & KA VALUE: 2.3.3 1.8/2.9 10CFR55.43.4 – Knowledge of SRO responsibilities for auxiliary system that are outside the control room.
REFERENCE: ODCM 6.1.1 table 6.1.1.1-1
SOURCE: **BANK QUESTION – 2000 NRC EXAM –** SRO T3, #9
LO: NO LO
RATING: H3
ATTACHMENT: YES - ODCM 6.1.1 table 6.1.1.1-1, PPM 4.602.A5.6-6
JUSTIFICATION: A and B are incorrect because they both allow the discharge to continue and the actions given are for the SW monitors and for the flowrate monitor of Rad Waste. D is incorrect because there is no action allowing the use of a temporary monitor in the place of FDR-RIS-606. C is correct. This is the action given in the ODCM.
COMMENTS: I need some more direction on this one from Ryan. The direction to stop the discharge is on one page and the actions are determined after referring to another table then referring to specific actions. Talk to Ryan. Made A and B action 101 and 102 of table 6.1.1.1-1.

The plant is in MODE 2 following a refuel outage. The following conditions exist:

Reactor power = 2%
RCIC is in Full Flow Test operation
Suppression pool temperature = 106°F

Which of the following is correct concerning these conditions?

A. Stop RCIC immediately to preserve the heat absorption capability of the suppression pool.

B. Place the Mode Switch in SHUTDOWN immediately to preserve the heat absorption capability of the suppression pool.

C. Reduce suppression pool temperature to less than 90°F within 1 hour to prevent exceeding the HCTL during a LOCA.

D. Reduce reactor power to less than 1% within 1 hour to prevent exceeding the HCTL during a LOCA.

ANSWER: A

QUESTION TYPE:	SRO
KA # & KA VALUE:	2.4.49 4.0/4.0 10CFR55.41, 43.2, & 45 – Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.
REFERENCE:	Tech Spec 3.6.2.1 and TS Bases B 3.6.2.1
SOURCE:	**NEW QUESTION** SRO T3, #13
LO:	6926 – Given a set of plant conditions, be able to identify and state from memory the Technical Specification actions required to be taken within 15 minutes or less.
RATING:	H3
ATTACHMENT:	NONE
JUSTIFICATION:	B is incorrect because at this temperature, there is no requirement to place the Mode Switch in SHUTDOWN. C is incorrect because there is no requirement to reduce suppression pool temperature to less than 90° within 1 hour for any reason. D is incorrect because at this temperature there is no requirement to reduce power to less than 1%. A is correct as state in the action statement and the Bases.
COMMENTS:	

QUESTION # 72 EXAM KEY
10/04/2002
ex00011

Which of the following describes the reactivity effect the End of Cycle Recirc Pump Trip (EOC RPT) is designed to minimize?

A. Control rod insertion may not initially add enough negative reactivity to overcome the positive reactivity added by the pressure increase from a turbine trip.

B. Control rod insertion initially adds positive reactivity late in core life that must be compensated for by the trip of both Recirc pumps.

C. Recirc Pumps must be tripped to reduce the positive reactivity addition from the turbine trip and prevent exceeding MAPRAT.

D. Recirc Pumps must be tripped late in core life to minimize the effect of all control rods withdrawn to the full out position and prevent exceeding the LHGR

ANSWER: A

QUESTION TYPE: SRO

KA # & KA VALUE: 295007AA2.02 4.1/4.1 10CFR55.41, 43.5 - Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE: Reactor Power

REFERENCE: TS Bases B3.3.4.1 EOC-RPT Instrumentation

SOURCE: **BANK QUESTION – 2000 NRC EXAM –** SRO T1, GP1, #5

LO: 6925 – Identify the basis for any LCO.

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: TS Bases states the EOC-RPT trip is designed to overcome the lack of negative reactivity in the first few feet of control rod insertion on a scram. This lack of negative reactivity is cause by EOC flux shape. The Recirc pumps are tripped to supplement the control rod negative reactivity on a turbine trip. A is correct.

COMMENTS:

QUESTION # 73 EXAM KEY
10/04/2002

ex00062

The reactor is operating at rated conditions.

Which of the following describes how CRDM graphitar seal embrittlement is prevented?

A. The CRD Mechanism is monitored for temperature by a thermocouple in the instrument tube and maintained less than 250°F.

B. The CRD Mechanism is monitored for temperature by a thermocouple in the outer tube and maintained less than 250°F.

C. Cooling water from CRDH is supplied to the P-over port at a high enough flow rate to ensure sufficient cooling of the CRD Mechanism.

D. Cooling water from CRDH is supplied to the outside of the thermal sleeve at a high enough flow rate to ensure sufficient cooling of the CRD Mechanism.

ANSWER: A

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 214000K4.02 2.5/2.5 10CFR55.41 - Knowledge of ROD POSITION INFORMATION SYSTEM design feature(s) and/or interlocks which provide for the following: Thermocouple

REFERENCE: LO000137 pages 10 and 13

SOURCE: **BANK QUESTION – 2000 NRC EXAM –** SRO T2, GP2, #8
RO T2, GP2, #8

LO: 5217 – Explain how rod temperatures are monitored.

RATING: H2

ATTACHMENT: NONE

JUSTIFICATION: C and D are both incorrect because they describe an incorrect flow path. B is incorrect because the thermocouple is located inside the instrument tube not the base of the stop piston.

COMMENTS: Changed the reference to the base of the stop piston to the outer tube to make B more clearly incorrect.

QUESTION # 75 EXAM KEY
10/04/2002
ex00115

The plant is in MODE 3 with a reactor level reduction occurring through the A RHR Heat Exchanger Vents to the Suppression Pool. The Suppression Pool high level alarm has been sealed in for a period of time when the HPCS Suction Switchover occurs.

Which of the following describes the required action for these conditions?

A. Be in MODE 3 in 36 hours.

B. Restore suppression pool water level to within limits and restore HPCS suction to the CSTs within 2 hours.

C. Enter PPM 5.2.1 Primary Containment Control and restore suppression pool water level to within limits in 12 hours.

D. Enter PPM 5.2.1 Primary Containment Control and restore suppression pool water level to within limits in 2 hours.

ANSWER: D

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 2.1.33 3.4/4.0 10CFR55.43 – Ability to recognize indications for system operating parameters which are entry level conditions of technical specifications.

REFERENCE: TS 3.6.2.2 PPM 5.0.10 rev. 6, page 246

SOURCE: **BANK QUESTION – 2000 NRC EXAM –** SRO T3, #5 RO T3, #3
WNP-2 LER 91-015

LO: 5429 – List the automatic initiations and interlocks associated with HPCS system components.

RATING: H3

ATTACHMENT: YES – TS 3.6.2.2

JUSTIFICATION: The high level suppression pool suction switchover for HPCS occurs at +3 inches in the suppression pool. This level requires entry into PPM 5.2.1 Primary Containment Control. This is also above the LCO for suppression pool water level. TS require that suppression pool level be restored to within limits in 2 hours. There is no requirement to return the HPCS suction valves to CST suction. D is correct.

COMMENTS: After review, Ryan said make distracter A “Be in MODE 4 in 36 hours”. This is clearly incorrect.

A plant startup is in progress following a refueling outage. Primary containment is being inerted with nitrogen supplied through the high flow inerting supply header. Drywell pressure has increased to 1.78 psig.

Which of the following is correct for these conditions?

A. Inerting continues as established.

B. CSP-V-106 and 107 (N2 tank isolations) close to isolate the high flow header.

C. CSP-V-93 and 96 (containment isolations) close to isolate the high flow header.

D. CSP-V-97 and 98 (containment isolations) close to isolate the high flow header

ANSWER: A

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 295010AK2.04 2.6/2.8 10CFR55.41 & 45 - Knowledge of the interrelations between HIGH DRYWELL PRESSURE and the following: Nitrogen makeup system.

REFERENCE: LO000162 rev. 6 pages 7 & 8

SOURCE: **NEW QUESTION –** SRO T1, GP1, #8 RO T1, GP1, #4
LO: 5159 – State the auto close signals for nitrogen makeup line containment isolation valves.
RATING: L2
ATTACHMENT: NONE
JUSTIFICATION: B is incorrect because these valves close on low header temperature. C and D are both incorrect because while these valves do close on hi drywell pressure, they are on the low flow header and have no effect on the hi flow header. A is correct.
COMMENTS:

According to PPM 5.0.10 Flowchart Training Manual, which of the following is a possible consequence of rapid injection of water into the RPV during ATWS conditions?

A. Thermal shock to feedwater nozzles.

B. Fuel damage.

C. Rapid pressure decrease exceeding 100°F/hour.

D. Feed pump trip on high RPV level.

ANSWER: B

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 295015AK1.03 3.8/3.9 10CFR55.41 - Knowledge of the operational implications of the following concepts as they apply to INCOMPLETE SCRAM: reactivity affects.

REFERENCE: PPM 5.0.10 rev. 6 page 71

SOURCE: **BANK QUESTION – DIRECT –** SRO T1, GP1, #11 RO T1, GP1, #5

LO: 8499 – Given a list, identify the statement that describes the plant response to rapid injection of water into the RPV during an ATWS.

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: PPM 5.0.10 states that rapid injection into the core during ATWS conditions may result in a large increase in reactivity large enough to cause fuel damage. B is the only correct answer.

COMMENTS:

The plant was operating at 86% power when a fire occurred in the main control room. As the CRS you have directed that the plant be scrammed from the control room and a shutdown be performed from the remote shutdown panel. The following conditions exist:

Reactor Pressure = 945 psig
Reactor Level = -72 inches
Drywell pressure = 1.75 psig

Which of the following is correct concerning level indication installed at the Remote Shutdown Panel?

A. The narrow range is accurate because the Recirc Pumps have tripped.

B. The narrow range is accurate because the Recirc Pumps are in operation.

C. The wide range is accurate because the Recirc Pumps have tripped.

D. The wide range is accurate because the Recirc Pumps are in operation.

ANSWER: C

QUESTION TYPE: SRO

KA # & KA VALUE: 295016AA2.02 4.3/4.4 10CFR5541, 43.5, & 45 - Ability to determine and/or interpret the following as they apply to CONTROL ROOM ABANDONMENT: Reactor Water Level

REFERENCE: LO000126 rev. 8 page 4 and LO000210 rev. 5 page 15

SOURCE: **NEW QUESTION –** SRO T1, GP1, #13

LO: 5885 – List the systems and alignments that can be performed from the RSD Panel

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: Only the Wide Range is installed at the Remote Shutdown panel for level monitoring, so A and B are incorrect. The wide range is calibrated for no jet pump flow and with level at –72 inches both Recirc Pumps have tripped. This makes C correct.

COMMENTS:

The plant was operating at 100% power when an accident occurred. The following conditions now exist:

RPV Level = -89 inches and stable
RPV Pressure = 997 psig and stable
Wetwell Level = 19 feet and down slow
Wetwell Temperature = 180°F and up slow
Drywell Pressure = 20 psig and down slow
Drywell Temperature = 300°F and down slow.

Which of the following is correct for these conditions?

A. Emergency Depressurize the reactor because the Drywell Temperature has exceeded the Drywell Design Temperature Limit.

B. Emergency Depressurize the reactor because the Heat Capacity Temperature Limit has been exceeded.

C. Spray the Wetwell because the Drywell Spray Initiation Limit has been exceeded.

D. Spray the Drywell because the Drywell Spray Initiation Limit has been exceeded.

ANSWER: B

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 295026 2.1.25 2.3/3.1 10CFR55.41, 43.5, &45 – Ability to obtain and interpret station reference materials such as graphs, monographs, and tables which contain performance data for High Suppression Pool Water Temperature.

REFERENCE: PPM 5.2.1 WW Temp leg and HCTL Curves

SOURCE: **NEW QUESTION –** SRO T1, GP1, #18 RO T1, GP2, #12

LO: 8303- Describe the reason for emergency depressurizing the RPV if wetwell termperature and reactor pressure connot be maintained below the HCTL.

RATING: H3

ATTACHMENT: **YES –** PPM 5.2.1 WW Temp Leg and HCTL curves

JUSTIFICATION: C and D are both incorrect because DSIL has not yet been exceeded. A is incorrect because Drywell temp is less than the design limit and going down. B is correct because the wetwell temperature has exceeded the HCTL.

COMMENTS:

QUESTION # 82 EXAM KEY
10/04/2002
ex02042

The plant was operating at 99% power when an accident occurred. The following conditions exist:

An electrical fault has caused a lockout on SM-8.
MS-LIS-38A (level 3 input to ADS) has failed upscale.
Reactor level has been at –135 inches for the last 2 minutes.
All other plant equipment has operated as designed.

Assuming no operator actions, which of the following is correct concerning these conditions?

A. ADS auto initiated from the Division 1 logic.

B. ADS auto initiated from the Division 2 logic.

C. ADS will **not** initiate automatically.

D. ADS will **not** initiate manually.

ANSWER: C

QUESTION TYPE: RO/SRO

KA # & KA VALUE: 218000K1.03 3.7/3.8 10CFR5541 & 45 - Knowledge of the physical connections and/or cause- effect relationships between AUTOMATIC DEPRESSURIZATION SYSTEM and the following: Nuclear Boiler Instrumentation.

REFERENCE: PPM 4.601.A3 rev. 13 page 4 LO000186 rev. 9, page 5 and fig. 7

SOURCE: **NEW QUESTION –** SRO T2, GP1, #12 RO T2, GP1, #17

LO: 5071 – State the condition that will automatically initiate ADS.
RATING: H3
ATTACHMENT: NONE
JUSTIFICATION: Under the conditions given, ADS will not auto initiate. There is not ECCS Div 2 systems and the Div 1 level 3 confirmatory y switch is out of service. However, because of the arrangement of the logic, ADS could be manually initiated by a manual arm and depress initiation. Therefore, C is the only correct answer.
COMMENTS:

The plant was operating at 100% power when a LOCA occurred. The following conditions exist:

RPV pressure = 403 psig
RHR-V-42B is closed with the amber “override” light illuminated
RHR B is in the drywell spray mode

Assuming no operator actions, which of the following is correct.

A. RHR-V-42B cannot be opened until one of the drywell spray valves is closed.

B. RHR-V-42B cannot be opened until both drywell spray valves are closed.

C. Drywell spray valves automatically close if RHR-V-42B is opened.

D. Drywell spray valves stay open if drywell pressure becomes negative.

ANSWER: D

QUESTION TYPE: RO/SRO

KA # & KA VALUE: 226001A2.20 3.7/4.1 10CFR55.41 & 45 - Ability to (a) predict the impacts of the following on the RHR/LPCI: CONTAINMENT SPRAY SYSTEM MODE; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: LOCA

REFERENCE: LO000198 rev. 10, pages 16, 19, & 20

SOURCE: **BANK QUESTION – DIRECT - #LO000835 –** SRO T2, GP1, #16 RO T2, GP2, #10

LO: 5781 – List the interlocks and trips for the RHR system.
RATING: H3
ATTACHMENT: NONE
JUSTIFICATION: A and B are both incorrect because the 42 valves (injection for LPCI) can be opened anytime the control switch is place in open and reactor pressure is less than 470 psig. C is incorrect because there are no auto close signals for the 16 or 17 valves (drywell spray valves). D is correct for the same reason given in C.
COMMENTS: Same question as LO00835

The plant is operating at 90% power. FWLC is operating in 3-Element control with RPV Level instrument “A” as the selected level input.

Which of the following describes the effect on ACTUAL RPV LEVEL if an open circuit occurs in the selected RPV level instrument output causing it to fail downscale?

RPV level …

A. goes up to the level 8 setpoint.

B. goes down to the level 3 setpoint.

C. goes down approximately 3 inches and returns to normal.

D. remains steady at the selected setpoint.

ANSWER: D

QUESTION TYPE: RO/SRO

KA # & KA VALUE: 259002A2.02 3.3/3.4 10CFR55.41 & 45 - Ability to (a) predict the impacts of the following on the REACTOR WATER LEVEL CONTROL SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of any number of Reactor Feedwater Inputs.

REFERENCE: LO000157 rev. 11 pages 7 & 8

SOURCE: **BANK QUESTION – DIRECT – LX00406 –** SRO T2, GP1, #18 RO T2, GP1, #23

LO: 5400 – Predict the expected response of the feedwater level control system in both Single and Three Element control to a failure of the selected level input.

RATING: L2
ATTACHMENT: NONE
JUSTIFICATION: Per the FWLC systems text, a failure of the selected FW level input to the FWLC system results in no change in the actual RPV level. FWLC auto shifts to the alternate level input. D is the correct answer.
COMMENTS:

Which of the following describes the basis for the requirement to match Recirculation flows at high power levels?

Matched flows ensure…

A. there are no axial flux anomalies at rated conditions.

B. there are no radial flux anomalies at rated conditions.

C. that during a LOCA due to a Recirc piping break, the assumptions of the LOCA analysis are satisfied.

D. that during a LOCA due to a Recirc piping break, the assumptions of the EOP Basis document are satisfied.

ANSWER: C

QUESTION TYPE: SRO

KA # & KA VALUE: 202001 2.2.25 2.5/3.7 10CFR55.43.2 – Knowledge of basis in Technical Specifications concerning the Recirc System.

REFERENCE: TS 3.4.1 and TS 3.4.1 Bases

SOURCE: **NEW QUESTION –** SRO T2, GP2, #5

LO: 6925 – Identify the basis for any LCO.

RATING: L3

ATTACHMENT: NONE

JUSTIFICATION: A and B are incorrect because as long as the reactor is not operating in the area of increased awareness, there is no concern with reactivity oscillations. D is incorrect because the TS basis for 3.4.1 states as long as the loop flows are matched, the assumptions of the LOCA analysis are satisfied. C is correct.

COMMENTS:

The plant was operating at 98% power when an ATWS occurred. The main turbine has tripped. There are 6 SRVs open.

Assuming no operator actions have been taken concerning pressure control, what is reactor pressure?

A. 1091

B. 1101

C. 1111

D. 1121

ANSWER: B

QUESTION TYPE: SRO

KA # & KA VALUE: 295037EA2.06 4.0/4.1 10CFR55.41, 43.5, & 45 - Ability to determine and/or interpret the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: Reactor pressure

REFERENCE: LO000128 rev. 8, page 4

SOURCE: **NEW QUESTION –** SRO T1, GP1, #4

LO: 5527 – List the opening setpoints for the SRVS

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: At 1101 psig in the reactor, there should be 6 SRVs open. 2 open at 1091 and 4 open at 1101. B is correct.

COMMENTS:

The plant was operating at 99% power when a Main Turbine Trip occurred but the reactor did not scram. Direction in the EOPs is given to manually open SRVs until pressure drops to 945 psig.

Which of the following describes the basis for this direction?

A. Assures that all possible energy is directed to the main condenser.

B. Maximizes the amount of steam condensed in the wetwell.

C. Maintains reactor water inventory in the containment.

D. Maintains pressure below the scram setpoint and allows resetting of the scram.

ANSWER: A

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 295037EK3.06 3.8/4.1 10CFR55.41 & 45 - Knowledge of the reasons for the following responses as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: Maintaining heat sinks external to the containment.

REFERENCE: PPM 5.0.10 rev 6, page 175

SOURCE: **BANK QUESTION – 2000 NRC EXAM – EX00044 –** SRO T1, GP1, #23 RO T1, GP1, # 11

LO: 8162 – Given a list identify the advantages of reducing reactor pressure to 945 psig if SRVS are cycling.

RATING: L3

ATTACHMENT: NONE

JUSTIFICATION: PPM 5.0.10 says reducing reactor pressure to 945 psig with SRVs minimizes the addition of energy to the containment. D is correct,

COMMENTS:

QUESTION # 90 EXAM KEY
10/04/2002
ex00019

The plant was operating at 99% power when a LOCA Signal was received. After verifying auto actions, the CRO notes neither LPCS-P-1 nor RHR-P-2A auto started nor do they have breaker indication on P601. Neither pump will start manually with their control switches on P601.

Which ONE of the following is the correct explanation for these conditions?

A loss of….

A. both B1-1 and C1-1 prior to the LOCA signal

B. both B1-1 and C1-1 after the LOCA signal

C. both B1-2 and C1-2 after the LOCA signal

D. both B1-2 and C1-2 prior to the LOCA signal

ANSWER: A

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 295004AA2.01 3.5/3.9 10CFR55.41, 43.5 - Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER: Cause of partial or complete loss of DC power.

REFERENCE: ABN-ELEC-125VDC rev. 1, page 11

SOURCE: **BANK QUESTION – 2000 NRC EXAM – EX00019 –** SRO T1, GP2, #1

LO: 5262 – Relationship of RHR to 125 VDC

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: A loss of Div 1 DC <u>prior</u> to a LOCA signal causes a loss of indication on P601 and a failure of the LPCS pump to start with either the control switch or from a LOCA signal. A is correct. D is incorrect because it references Div 2 DC power.

COMMENTS:

The plant is operating at 100% power when a leak in the CAS System causes pressure to go down. The following conditions exist:

CAS-C-1A, 1B, & 1C are running.
CAS-PCV-1, Dryer Bypass is open.
SA-PCV-2, Control/Service Air Crosstie is closed.
Annunciator CONTROL AIR HDR PRESS LOW is illuminated.
Annunciator SERVICE AIR HDR PRESS LOW is illuminated.
Annunciator SERVICE AIR HDR ISOLATED is illuminated.

Concerning these indications, which of the following is the correct Control Air Header pressure?

A. 72

B. 82

C. 92

D. 102

ANSWER: A

QUESTION TYPE: SRO

KA # & KA VALUE: 295019AA2.01 3.5/3.6 10CFR55.41, 43.5, & 45 - Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: Service air isolation valves

REFERENCE: LO000205 rev. 8, page23

SOURCE: **NEW QUESTION –** SRO T1, GP2, #7

LO: 5878 – List the expected automatic control air system response to a leak in the control air system.

RATING: H3

ATTACHMENT: NONE
JUSTIFICATION: CAS-PCV-1 opens at 75 psig, which is the lowest pressure of any on the given indications. A is correct.

COMMENTS:

ex02048

Which of the following is the reason an Emergency Depressurization is required if Secondary Containment Temperature exceeds its Max Safe Operating Value in more than one area?

Emergency Depressurization…

A. requires a manual scram, which may not have occurred up to this point.

B. reduces the total energy released to the primary and secondary containment.

C. promptly places the primary system in its lowest possible energy state.

D. rejects heat to the main condenser in preference to the reactor building.

ANSWER: C

QUESTION TYPE: SRO\RO

KA # & KA VALUE: 295032EK3.01 3.5/3.8 10CFR55.42 & 45 - Knowledge of the reasons for the following responses as they apply to HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS: Emergency Depressurization

REFERENCE: PPM 5.0.10 rev. 6, page 299

SOURCE: **NEW QUESITON –** SRO T1, GP2, #10 RO T1, GP3, #1

LO: 8459 – Given a list identify the statement that describes the two reasons for emergency depressurizing the RPV if on secondary containment parameter is above Maximum Safe Operation Levels in more than one area and a primary system is discharging reactor coolant into secondary containment .

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: A is incorrect because Emergency Depressurization does not require a manual scram. B is incorrect because the total energy released is the same, it is just directed to the wetwell. D is incorrect because the Emergency Depressurization is completed using SRVs and not BPVs. C is the reason stated in PPM 5.0.10 and is the correct answer.

COMMENTS:

QUESTION # 93 EXAM KEY
10/04/2002
ex02069

The plant is operating at 88% power when the following auto actions take place:

SGT starts
CSP/CEP isolates
CN makeups isolate
CR and TSC Emerg. Filtration starts and aligns to remote air intakes
RB Emerg. Room Coolers start
RB Lighting quenches
RB EDR and FDR discharge headers isolate

The plant remains operating at power following the initiations.

Which of the following is correct concerning these initiations?

These initiations were caused by a leak…….

A. in the Recirc Pump Suction line.

B. from a Main Steam Line flow element.

C. from RWCU in the Heat Exchanger Room.

D. of spent resin from a RWCU Demin during regeneration.

ANSWER:	C
QUESTION TYPE:	SRO
KA # & KA VALUE:	295034EA2.02 3.7/4.2 10CFR55.41, 43.5, & 45 - Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT VENTILATION HIGH RADIATION Cause of the high radiation areas
REFERENCE:	ABN-FAZ-QC pages 3 & 4
SOURCE:	**BANK QUESTION – MODIFIED – 2000 NRC EXAM EX00022 –** SRO T1, GP2, #13
LO:	6914 – Given plant conditions identify those annunciators and indications that would indicate an FAZ actuation and entry into ABN-FAZ.
RATING:	H4
ATTACHMENT:	NONE
JUSTIFICATION:	A and B are incorrect because they would have scrammed the reactor along with starting the listed equipment. D is incorrect because the RWCU demins are located in the Radwaste building and would not cause a Z signal. C is correct because a leak in the RWCU heat exchanger room would cause spread of contamination throughout the reactor building and result in a Z signal trip.
COMMENTS:	

QUESTION # 94 EXAM KEY
10/04/2002
ex00038

The plant is in MODE 1 at rated conditions with shift turnover in progress. The oncoming Shift Manager has been notified of the absence of the oncoming Mechanical Maintenance person and Plant Laborer due to an automobile accident.

Based on these conditions, which of the following is correct?

A. Both the Maintenance person and the Plant Laborer must be replaced within 2 hours of the start of the shift.

B. One equipment Operator can be designated as an emergency maintenance person, but the on duty Plant Laborer must remain on duty until relieved.

C. Neither the Maintenance person nor the Plant Laborer from the previous shift can leave the plant until relieved by a qualified employee.

D. One Health Physics person can be designated an emergency maintenance person and the Plant Laborer position can be left unmanned.

ANSWER: B

QUESTION TYPE: SRO

KA # & KA VALUE: 600000 2.1.5 2.3/3.4 10CFR55.41, 43.5 – Ability to locate and use procedures and directives related to shift staffing and activities: Fire related.

REFERENCE: PPM 1.3.1 rev 56, page 44 & 45

SOURCE: **BANK QUESTION – DIRECT – 2000 NRC EXAM –** SRO T1, GP21, #16

LO: 6071 – State the minimum staffing level and crew makeup required, both administrative and legal for any given situation.

RATING: H3

ATTACHMENT: **YES** - PPM 1.3.1 rev 56, page 44 & 45

JUSTIFICATION: Because the Laborer is a member of the Plant Fire brigade, he has to be relieved prior to leaving the plant, and must stay until relieved by a qualified Fire Brigade member. The Maintenance person is not a member of the Fire Brigade and can be replaced by one of the equipment operators on shift for emergency maintenance. B is correct.

COMMENTS:

QUESTION # 95 EXAM KEY
10/04/2002
ex02049

The plant was operating at 89% power when a transient occurred resulting in a scram. It is now several minutes following the scram and the following conditions exist:

Reactor level = 53 inches and rising.
Reactor pressure = 1057 and steady.
RCIC is on at 600 gpm in auto.
HPCS and Feedwater are both off.

Assuming no further operator actions, which of the following will occur given these conditions?

A. RCIC-V-13 and 45 close.
RCIC-V-13 and 45 must be manually reopened to restart RCIC.

B. RCIC-V-13 and 45 close.
Reactor pressure increases and causes a scram signal.
RCIC-V-13 and 45 open when level is reduced to the initiation setpoint.

C. RCIC trips, RCIC-V-1 closes.
RCIC-V-1must be manually reopen to restart RCIC.

D. RCIC trips, RCIC-V-1 closes.
Reactor pressure increases and causes a scram signal.
RCIC-V-1 opens when level is reduced to the initiation setpoint.

ANSWER: B

QUESTION TYPE:	SRO/RO
KA # & KA VALUE:	217000K3.02 3.6/3.6 10CFR55.41 & 45 - Knowledge of the effect that a loss or malfunction of the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) will have on following: Reactor vessel pressure
REFERENCE:	LO000180 rev., 11, pages10-12 & 20, 21 lo000161 rev. 11, page11
SOURCE:	**NEW QUESTION –** SRO T2, GP1, #10 RO T2, GP1, #15
LO:	5722 – list the RCIC system isolation signal and setpoints.
RATING:	H2
ATTACHMENT:	NONE
JUSTIFICATION:	RCIC-V-45 and 13 auto close at +54.5 inches. They auto open at –50 inches and initiate RCIC. Due to the design of RCIC, steam supply from the vessel and vessel return in the head spray line, it acts to suppress vessel pressure when in operation. When level reaches level 8 and the valves close, reactor pressure increases. B is correct.
COMMENTS:	

QUESTION # 96 EXAM KEY
10/04/2002
ex02050

Which of the following design features prevents vertical thermal stratification in the Suppression Pool?

A. SRV discharge quenchers are located in the upper elevations of the Suppression Pool.

B. Drywell floor downcomers are located in the upper elevations of the Suppression Pool.

C. RHR discharge lines are located below the elevation of the suctions lines.

D. RHR discharge lines are located above the elevation of the suctions lines.

ANSWER: D

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 295013AK1.01 2.5/2.6 10CFR55.41 - Knowledge of the operational implications of the following concepts as they apply to HIGH SUPPRESSION POOL TEMPERATURE: Pool Stratification.

REFERENCE: Columbia FSAR 6.2.2.3 page 6.2-33 & 3
4 and figure 6.2-32

SOURCE: **NEW QUESTION –** SRO T1, GP1, #9 RO T1, GP2, #7

LO: NO LO

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: C is incorrect because the FSAR states that the RHR discharge lines are located above the elevation of the suction lines for prevention of stratification. A and B are incorrect because they are both located at a lower elevation in the suppression pool. D is correct as stated in the Columbia Generating Station FSAR.

COMMENTS:

The plant is operating at 100% power when conditions inside the Reactor Building cause Reactor Building ventilation radiation level to increase to 18 mr/hr.

Which of the following is correct?

A. Manual action should have been taken to isolate Reactor Building Ventilation and start Standby Gas Treatment prior to exceeding the initiation setpoint to prevent a site boundary release.

B. Manual action should have been taken to isolate Reactor Building Ventilation and start Standby Gas Treatment prior to allow a ground level release from the reactor building and prevent a site boundary release.

C. SGT is verified to be in operation and Reactor Building Ventilation is verified to be isolated to allow a ground level release from the reactor building and prevent a site boundary release.

D. SGT is verified to be in operation and Reactor Building Ventilation is verified to be isolated to prevent a radioactive release to the environment from a system that should have isolated.

ANSWER: D

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 295038EA1.06 3.5/3.6 10CFR55.41 & 45 - Ability to operate and/or monitor the following as they apply to HIGH OFF-SITE RELEASE RATE: Plant Ventilation

REFERENCE: PPM 5.0.10 rev. 6, page 291

SOURCE: **NEW QUESTION –** SRO T1, GP1, #16 RO T1, GP2, #10

LO: 8460 – Given a list, identify the statement that describes the purpose to confirming RB HVAC isolation and SGT initiation when and FAZ exists.

RATING: H2

ATTACHMENT: NONE

JUSTIFICATION: A and B are both incorrect because auto actions should have occurred before the indicated rad level was this high (13 mr/hr). C is incorrect because operation of SGT and isolation of RB Ventilation do NOT cause a ground level release. D is correct as defined by PPM 5.0.10

QUESTION # 98 EXAM KEY
10/04/2002
COMMENTS:
ex02052

The plant had been operating at 100% power for several months. A normal controlled shutdown to cold conditions is in progress. Reactor power is approximately 20 on range 8 of all IRMs, with a control rod insertion in progress. An event occurs that causes the reactor to scram on high power.

Which of the following is the minimum power level attained that caused the scram?

A. 1%

B. 6%

C. 10%

D. 16%

ANSWER: B

QUESTION TYPE: SRO

KA # & KA VALUE: 295014AA2.01 10CFR55.41, 43.5, & 45 - Ability to determine and/or interpret the following as they apply to INADVERTENT REACTIVITY ADDITION: Reactor Power

REFERENCE: LO000138 rev. 7, page 8

SOURCE: **NEW QUESTION – COLUMBIA LER86-004-00 –** SRO T1, GP1, #17

LO: 5452 – Describe the correlation between IRM and APRM readings.

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: As stated in the LER, reactor pressure perturbations from bypass valve operation caused an increase in reactor power. The increase in power causes an increase in IRM indication. When IRM levels increase to 120/125s of scale, a scram is inserted. Power level of 16% would cause a scram from mode switch position, but the IRM scram happens much sooner. B is the only correct answer.

COMMENTS:

What is the basis for the 135° F Tech Spec LCO for Drywell Temperature?

Maintaining Drywell Temperature less than the LCO value assures that the…

A. external design pressure will not be exceed when starting one loop of RHR Drywell Spray.

B. drywell to wetwell interface will not fail during the blowdown portion of a DBA LOCA.

C. equipment inside the Drywell needed to mitigate the effects of a DBA LOCA will operate under conditions expected for the accident.

D. peak post LOCA drywell temperature does not exceed the design temperature of 290° F.

ANSWER: C

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 295028EK1.02 2.9/3.1 10CFR55.41 - Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL TEMPERATURE: Equipment environmental qualification.

REFERENCE: Tech Spec Bases B3.6.1.4 page B3.6.1.4-1

SOURCE: **NEW QUESTION –** SRO T1, GP2, #9 RO T1, GP2, #13

LO: 5635 – List the value for drywell design temperature.

RATING: L3

ATTACHMENT: NONE

JUSTIFICATION: The Basis for TS 3.6.1.4 states that as long as Drywell temperature is maintained less than the TS LCO of 135° F, that equipment inside the containment needed to mitigate the effects of a DBA LOCA will operate as expected. C is correct. A, B, and D are all misapplications or misstatements of other limits.

COMMENTS:

The plant is in MODE 5 with fuel movement underway. A fuel bundle has been dropped and released enough radioactivity into the reactor coolant and to the 606' elevation of the reactor building to result in Reactor Building Exhaust Plenum high radiation of 22 mr/hr.

Which of the following systems is designed to minimize the leakage to the outside atmosphere during these conditions?

A. Control Room Ventilation

B. Secondary Containment

C. Primary Containment

D. Reactor Building Ventilation.

ANSWER: B

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 295033EK1.03 3.9/4.2 10CFR55.41 - Knowledge of the operational implications of the following concepts as they apply to HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS: Radiation release.

REFERENCE: LO000139 rev. 9, page 3

SOURCE: **BANK QUESTION** – **DIRECT – 98 NRC EXAM –** SRO T1, GP2, #12 RO T1, G2, #116

LO: 6999 – State the purpose of Secondary Containment

RATING: L3 L2

ATTACHMENT: NONE

JUSTIFICATION: B is the correct answer. The purpose of the Sec Cont is to minimize the release of radioactivity under these conditions. The other systems either would not limit the spread of radioactivity to the environment or are isolated.

COMMENTS:

The plant is shutdown with RHR-P-2B running in the Shutdown Cooling Mode.

RRC-P-1A is off with RRC-V-67A (1A discharge) closed.
RRC-P-1B is off with RRC-V-67B (1B discharge) closed.
Reactor level is +60 inches

RRC-V-67A is then inadvertently opened.

Which of the following is correct concerning these conditions?

A. No effect on Shutdown Cooling, the suction for RHR-P-2B is from Recirculation Loop B.

B. No effect on Shutdown Cooling, the discharge from RHR-P-2B for Shutdown Cooling goes directly into the core.

C. Shutdown Cooling has bypassed the core and may cause undetected heating of the vessel if level falls below +50 inches.

D. Shutdown Cooling has bypassed the core and may cause undetected heating of the vessel if level falls below +60 inches.

ANSWER: D

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 205000K6.03 3.1/3.2 10CFR55.41 & 45 - Knowledge of the effect that a loss or malfunction of the following will have on the SHUTDOWN COOLING SYSTEM/MODE: Recirculation system.

REFERENCE: SOP-RHR-SDC rev. 0, page 16 and LO000198 rev 10, page 4

SOURCE: **NEW QUESTION –** SRO T2, GP2, #7 RO T2, GP2, #7
LO: 5774 – Describe the flow path for RHR Shutdown Cooling.
RATING: H3
ATTACHMENT: NONE
JUSTIFICATION: As stated in PPM 2.4.2, when the 67A valve is opened, a short circuit pathway is opened. If Reactor level is reduced to less than 60 inches, core circulation cannot be assured and undetected heating is the result. D is correct.
COMMENTS:

The plant is in MODE 4. The following conditions exist:

Reactor level = 72 inches
Coolant temperature = 160°F
RRC-P-1A = Off
RRC-P-1B = Off
RHR-P-2B = On in Shutdown Cooling.

What is the basis for the level limitation under these conditions?

Reactor level at 72 inches ensures water level is …

A. above the “turn around” point , there is flow through the core, and thermal stratification will be prevented

B. below the dryer skirt, there is flow from the steam dryer through the core, and thermal stratification is prevented.

C. below the feedwater spargers which promotes better mixing of incoming feedwater to prevent uneven heating of the reactor pressure vessel.

D. above the jet pumps which promotes mixing from the recirculation loops and prevents uneven heating in the reactor pressure vessel.

ANSWER: A

QUESTION TYPE: SRO
KA # & KA VALUE: 290002 2.4.21 (2) 3.7/4.3 10CFR55.43.5 & 45 – Knowledge of the parameters and logic used to assess the status safety functions including: 2 Core cooling and heat removal – Reactor Vessel internals.
REFERENCE: PPM 3.2.1 rev. 46, page 7 LO000198 rev. 10, page 38
SOURCE: **NEW QUESTION –** SROT2, GP3, #2
LO: NO LO
RATING: H3
ATTACHMENT: NONE
JUSTIFICATION: Reactor level at 72 inches ensures water is above the turnaround point and that thermal stratification will be avoided. A is correct. B is incorrect because reactor level is above the dryer skirt at 72 inches. C is incorrect because level is above the feedwater spargers. D is incorrect because mixing from the recirc loops does not promote core flow/circulation.
COMMENTS:

QUESTION # 105 EXAM KEY
10/04/2002
ex02056

As the Shift Manager, you have been notified that an employee with unescorted access has processed through the Primary Access Point and into the protected area without being observed by a Security Officer. There are no Security Officers in the Primary Access point.

Which of the following is the correct action for these conditions?

A. No action is necessary, the employee has unescorted access and passed through the explosive and metal detectors.

B. Notify the NRC within 14 days.

C. Immediately make a verbal notification to the Security Supervisor and notify the NRC within 1 hour.

D. Immediately make a verbal notification to the Security Supervisor and notify the NRC within 14 days.

ANSWER: C

QUESTION TYPE: SRO

KA # & KA VALUE: 2.1.13 2.0/2.9 10CFR55.41, 43.5, &45 – Knowledge of the facility requirements for controlling vital/controlled access.

REFERENCE: PPM 1.10.1 rev 22 pages 6 & 11

SOURCE: **BANK QUESTION – 98 NRC ADMIN EXAM –** SRO T3, #2
WNP02 LER 96-S01-00

LO: 6011 – Given procedures determine reportability for a specific event.

RATING: H3

ATTACHMENT: YES – PPM 1.10.1 pages 6 & 11

JUSTIFICATION: PPM 1.10.1 requires that any condition/event that threatens or lessens the physical security of the plant be reported to the Security Supervisor immediately. Furthermore, the actual event was reported under 10cfr73 app G, which requires a 1 hour notification to the NRC. C is the correct answer.

COMMENTS:

The plant is in MODE 5 with control rod removal underway. Control rod 30-31 has to be uncoupled from above the core.

Which of the following tools is used for this action?

A. Fuel support tool

B. Control rod grapple

C. Control rod guide tube grapple

D. Control rod latch tool

ANSWER: D

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 2.2.27 2.6/3.5 10CFR55.43 – Knowledge of the refueling process.

REFERENCE: LO000207 rev. 9, page 15

SOURCE: **BANK QUESTION #3982-** Slightly Modified – SRO T3, #6

LO: 7701 – State the purpose of the control rod latch tool.

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: The tool used to unlatch and remove a control rod from above the core is the control rod latch tool. D is correct.

COMMENTS: Changed question for clarification.

QUESTION # 108 EXAM KEY
10/04/2002
ex02059

The plant is operating at 99% power with the following equipment out of service:

Battery charger C2-1
CRD-P-1A
TO-P-BOP Main Bearing Oil Pump

Which of the following events will cause the 250 VDC battery discharge rate to go up (discharge faster)?

A. Inadvertent HPCS initiation.

B. Inadvertent Div 1 ECCS initiation.

C. Main Turbine Trip.

D. RFW-P-1A trip.

ANSWER: C

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 263000A1.01 2.5/2.8 10CFR55.41 & 45 – Ability to predict and/or monitor changes in parameters associated with operating the DC Electrical Distribution controls including: Battery charging discharging rate.

REFERENCE: LO000188 rev. 6, pages 28 & 29

SOURCE: **NEW QUESTION –** SRO T2, GP2, #11 RO T2, GP2, #14

LO: 5263 – Given a list of loads important to plant safety, identify its relation to 250 VDC bus S2-1.

RATING: H2

ATTACHMENT: NONE

JUSTIFICATION: A and B are both incorrect because they will cause no drain on the 250 VDC system directly, and no drain for any indirect reasons. The trip of the feedpump causes a runback of recirc, but no increased discharge either directly or indirectly on the 250 VDC system. C is correct because the Turbine Trip with the Bearing oil pump out of service causes the Emergency Bearing Oil Pump to start, which is powered from S2-1, 250 VDC.

COMMENTS:

The plant is in MODE 5 with a Full Core Verification in progress.

Which of the following is considered a core alteration?

A. Withdrawal of one SRM with the control switch from the control room.

B. Withdrawal of a control rod from a cell with no fuel.

C. Movement of an irradiated fuel bundle in the Fuel Pool.

D. Reseating of a fuel bundle in the core with the refuel mast.

ANSWER: D

QUESTION TYPE: SRO

KA # & KA VALUE: 2.2.34 2.3/3.3 10CFR55.43.6 – Knowledge of the process for determining the internal and external effects on core reactivity.

REFERENCE: PPM 6.3.5 rev. 10, page 3

SOURCE: **NEW QUESITON –** SRO T3, #7

LO: 7699 – For a given refueling operation, determine if the evolution is a Core Alteration.

RATING: L3

ATTACHMENT: NONE

JUSTIFICATION: A , B, and C are all incorrect because they do not meet the Tech Spec definition of a core alteration. D is correct because PPM 6.3.5 specifically states the reseating of a fuel bundle during core verification is a core alteration.

COMMENTS:

QUESTION # 110 EXAM KEY
10/04/2002
ex02060

Which of the following is correct when a High High Rad signal for Radwaste Effluent is received during a discharge to the river?

A. No auto actions, alarm only.

B. FDR-V-187 & 188 (RW Effluent) auto close.

C. FDR-P-45 (RW Eff. Sample Pump) trips, FDR-V-187 & 188 are closed manually.

D. CBD-LCV-1 auto closes to isolate the discharge to the river.

ANSWER: B

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 272000K1.05 2.8/3.1 10CFR55.41 & 45 - Knowledge of the physical connections and/or cause- effect relationships between RADIATION MONITORING SYSTEM and the following: Radwaste System.

REFERENCE: 82-RSY-0400-T6 rev. 9, page18

SOURCE: **NEW QUESTION –** SRO T2, GP2, #12 RO T2, GP2, #15

LO: 5647 – State the auto actions of Radwaste Effluent upon sensing high radiation.

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: A is incorrect because the Hi alarm has no actions, a Hi Hi causes the effluent isolation valves to close. C is incorrect because the sample pump does not trip and the valves auto close. D is incorrect because the action to isolate the discharge to the river occurs upstream of the CW System. B is correct.

COMMENTS:

A RWCU Demineralizer resin transfer is in progress when a leak in the system occurs. WEA-RIS-14 Radwaste Building Exhaust and WEA-RIS-14A Radwaste Building Exhaust Extended Range are both in alarm.

Which of the following is correct for these conditions?

A. No auto action occurs, these rad monitors are alarm only.

B. All resin transfer pumps trip immediately.

C. ROA-FN-1A/1B and WEA-FN-1A/1B trip.

D. ROA-FN-1A and WEA-FN-1A trip.

ANSWER: A

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 268000A1.01 2.7/3.1 10CFR55.41 & 45 - Ability to predict and/or monitor changes in parameters associated with operating the RADWASTE controls including: Radiation level

REFERENCE: 82-RSY-0400-T6 rev. 9, page 41

SOURCE: **NEW QUESTION –** SRO T2, GP3, #4 RO T2, GP3, #4

LO: 5647 – State the auto actions of Radwaste Effluent upon sensing high radiation.

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: Neither of the WEA radiation indicating switches have auto actions associated with Hi indications, only alarm. A is the only correct response.

COMMENTS:

The plant was operating at 100% power when a transient occurred. As the Shift Manager you have declared an Alert and have become the Emergency Director.

Which of the following can relieve the Shift Manager as the Emergency Director?

A. The OSC Manager or the EOF Manager

B. The TSC Manager or the JIC Manager.

C. The OSC Manager or the JIC Manager

D. The TSC Manager or the EOF Manager.

ANSWER: D

QUESTION TYPE: SRO

KA # & KA VALUE: 2.4.40 2.3/4.0 10CFR55.45 – Knowledge of the SROs responsibility in Emergency Plan Implementation.

REFERENCE: PPM 13.1.1 rev. 31, page 3

SOURCE: **NEW QUESTION –** SRO T3, #14

LO: 6132 – Identify the persons who may assume the position of Emergency Director.

RATING: L3

ATTACHMENT: NONE

JUSTIFICATION: As stated in PPM 13.1.1, only the TSC or the EOF Manager can assume the Emergency Director position. Neither the OSC nor the JIC Managers are permitted to assume this roll. D is correct.

COMMENTS:

The plant was operating at 99% power when a scram occurred. Four control rods stopped at position 02.

Which of the following is correct concerning these conditions?

Reactor shutdown is…

A. not assured; immediately depress the manual scram pushbuttons and initiate ARI.

B. not assured; immediately confirm reactor power is downscale on the APRMs.

C. assured; immediately depress the manual scram pushbuttons and initiate ARI.

D. assured; immediately confirm reactor power is downscale on the APRMs.

ANSWER: A

QUESTION TYPE: SRO

KA # & KA VALUE: 295006AA2.01 4.5/4.6 10CFR55.41, 43.5, & 45 - Ability to determine and/or interpret the following as they apply to SCRAM: Reactor power.

REFERENCE: PPM 3.3.1 REV. 39, page 6 PPM 5.0.10 rev. 6, page 186

SOURCE: **NEW QUESTION –** SRO T1, GP1, #20

LO: 8182 – Given a list, identify the criteria that must be met to ensure that the reactor is shutdown with no boron injected.

RATING: L3

ATTACHMENT: NONE

JUSTIFICATION: C and D are incorrect because reactor shutdown cannot be assured with 4 control rods at position 02. B is incorrect because confirmation of reactor power on the APRMs is not an immediate scram action. A is correct because shutdown cannot be assured and the scram procedure requires that the manual scram pushbuttons be depressed and ARI be initiated immediately if all rods do not insert.

COMMENTS:

The plant was operating at 96% power when a transient occurred resulting in high hydrogen concentration in the containment. You have been directed to perform PPM 5.5.16 Emergency Drywell and Wetwell Purging.

Which of the following parameters must be monitored during the performance of this procedure?

A. Suppression Chamber Temperature

B. Suppression Pool Level

C. Drywell Temperature

D. Drywell Pressure

ANSWER: D

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 500000EA1.07 3.4/3.3 10CFR55.41 & 45 - Ability to determine and/or interpret the following as they apply to SCRAM: Nitrogen purge system.

REFERENCE: PPM 5.5.16 rev. 6, page 3 and PPM 4.603.A7.6-4 rev. 28 page 65

SOURCE: **NEW QUESTION –** SRO T1, GP1, #26 RO T1, GP1, #12

LO: NO LO

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: During a nitrogen purge, containment pressure is the parameter of concern. Without a concurrent containment vent in progress, containment pressure increases. D is correct.

COMMENTS:

The plant is operating at 98% power. Maintenance is performing repair work in the area of the Main Turbine Front Standard. Each person is allowed a maximum dose of 300 mrem for the job.

Which of the following will increase the allowable individual stay time for this job?

A. Failure of the Main Oil Pump on the HP end of the Main Turbine.

B. Condenser air in-leakage greater than the capacity of Offgas.

C. Assignment of three additional Maintenance Personnel.

D. Rotate personnel out of the rad area for short periods of time.

ANSWER: B

QUESTION TYPE: SRO

KA # & KA VALUE: 295005 2.3.2 2.5/2.9 10CFR55.41, 43.4, & 49 – Knowledge of the ALARA program – Turbine Trip.

REFERENCE: LO000129 rev. 9, page 32 LO000161 rev. 11, page 11

SOURCE: **NEW QUESTION –** SRO T1, GP2, #4

LO: 5568 – List parameters and setpoints that will cause a turbine trip.

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: Because a maximum allowable dose is given, individual stay time can be increased by reduction of the rad field, increasing distance from the source, or adding shielding. Since this is a job on a particular piece of equipment, the distance from the source is fixed. Additional shielding was not given as a possible alternative, therefore only a reduction in the strength of the rad field is possible. C and D are both incorrect because they do not increase the stay time given the maximum dose. A is incorrect because it does not result in a reduction of the rad field; the main turbine does not trip. B is correct because the leak results in a trip of the main turbine and a reactor scram resulting in a reduction of the rad field.

COMMENTS:

ex02065

The plant was operating at 99% power when a transient occurred. The following conditions now exist:

Drywell Temperature = 274°F
Drywell Pressure = 59 psig
Suppression Chamber Pressure = 54 psig
Suppression Pool Temperature = 72°F
Suppression Pool Level = 31 feet

Which of the following is the cause of these conditions?

A. A LOCA with an SRV tail pipe broken above the suppression pool.

B. A LOCA with a broken Drywell Floor.

C. An ATWS with all Bypass Valves failed closed.

D. An ATWS with a broken Drywell Floor Downcomer.

ANSWER: B

QUESTION TYPE:	SRO
KA # & KA VALUE:	295028EA2.05 3.6/3.8 10CFR55.41, 43.5, & 44.5 - Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE: Torus/Suppression chamber pressure.
REFERENCE:	PPM 5.0.10 rev. 6 pages 91 & 92
SOURCE:	**NEW QUESITON –** SRO T1, GP1, #8
LO:	8339 – Given a list recognize the primary containment functions that the Pressure Suppression Curve is designed to protect.
RATING:	H4
ATTACHMENT:	**YES –** PPM 5.0.10 page 91 (without text)
JUSTIFICATION:	The only way to achieve the conditions given is to have a steam discharge into the containment with a bypass of the suppression pool: a loss of the pressure suppression function. The only response that results in this condition is B. B is correct. The other responses do not result in the loss of the pressure suppression function.
COMMENTS:	

An accident occurred causing a radioactive release. An evacuation has been advised for Sector 1. Meteorological conditions have changed with the wind now blowing from 135°.

Which of the following is correct for these conditions?

Evacuate…

A. Section 2.

B. Section 3a.

C. Section 3b.

D. Section 4

ANSWER: D

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 295038EK1.03 2.8/3.8 10CFRE55.41 - Knowledge of the operational implications of the following concepts as they apply to HIGH OFF-SITE RELEASE RATE: Meteorological effects on Rad release

REFERENCE: Columbia Classification Notification Form (CNF) #968-24075 r16

SOURCE: **NEW QUESTION –** SRO T1, GP1, #24 RO T1, GP2, #17

LO: 8893 – Identify required PARs fro each Emergency Classification.

RATING: H3

ATTACHMENT: **YES –** CNF Form 968-24075 rev 16

JUSTIFICATION: Wind direction has changed 90° from its original direction. It is now blowing from the 135° to 315° , requiring evacuation of Sector 4. D is correct.

COMMENTS:

The plant was operating at 97% power when a complete loss of offsite power occurred. All plant systems responded as designed. Reactor level is –10 inches and up fast. There have been no operator actions since the LOOP. HPCS-P-2 (HPCS Service Water) then trips.

Which of the following is correct for these conditions?

A. DG-3 is allowed to run indefinitely without HPCS-P-2 in operation.

B. DG-3 is allowed to run until the HPCS pump is no longer needed for level control.

C. Immediately trip DG-3 at the local panel with the Emergency Stop pushbutton.

D. Immediately trip DG-3 from P601 with the Emergency Stop pushbutton.

ANSWER: C

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 209002K6.03 2.5/2.6 10CFR55.41 & 45 - Knowledge of the effect that a loss or malfunction of the following will have on the HIGH PRESSURE CORE SPRAY SYSTEM (HPCS): Component Cooling Water system.

REFERENCE: ABN-SW rev. 2, page 2

SOURCE: **NEW QUESITON –** SRO T2, GP1, #4 RO T2, GP1, #7

LO: 6760 – State the immediate actions associated with ABN-SW.

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: With a loss of service water to the HPCS DG, ABN-SW requires that DG3 be tripped with the emergency stop pushbutton at the local panel. C is the only correct action.

COMMENTS:

Which of the following activities require the initiation of an Action Request?

A. Relocation of the control switches for SGT-B-1A1 & 1B1.

B. Replacement of the motor operator (same design) for SGT-V-2A.

C. Performance of an Electrical Surveillance on SGT-V-3A1.

D. Performance of a valve lineup following a refueling outage.

ANSWER: A

QUESTION TYPE: SRO

KA # & KA VALUE: 261000 2.2.14 2.1/3.0 10CFR55.43.3, 45 – Knowledge of the process for making configuration changes - SGT

REFERENCE: SWP-AIT-01 rev. 2, page6 SWP-DES-01 rev. 1, page 7

SOURCE: **NEW QUESTION –** SRO T2, GP1, #15

LO: NO LO

RATING: H3

ATTACHMENT: **YES -** SWP-DES-01 rev. 1, page 7

JUSTIFICATION: A is correct because it is a modification to a system in the power block as defined in SWP-DES-1. B is incorrect because the replacement of a motor operator with the same design motor is a maintenance procedure and not a system modification. C and D are both incorrect because they are covered by surveillance procedures.

COMMENTS:

The plant is operating at 99% when reactor level decreases several inches and stabilizes at a new level lower than the setpoint. The Channel A level instrument is selected.

Which of the following is the cause of these indications?

A. MS-RV-4D on Main Steam Line D fails open.

B. MS-V-160A (#1 BPV) fails open.

C. Steam flow transmitter RFW-DPT-803A fails open instantly.

D. Reactor level transmitter RFW-DPT-4A fails downscale instantly.

ANSWER: A

QUESTION TYPE: RO

KA # & KA VALUE: 259002A3.03 3.2/3.2 10CFR55.41 & 45 - Ability to monitor automatic operations of the REACTOR WATER LEVEL CONTROL SYSTEM including: Changes in Main Steam Flow

REFERENCE: LO000157 rev. 11, pages 19 & 20

SOURCE: **NEW QUESTION –** RO T2, GP1, #24

LO: 5400 – Predict the expected response of the Feedwater Level Control System with an SRV failed open.

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: A failure of instrumentation to FWLC causes an instantaneous shift to single element control. The result of this failure is no indicated change in level. C and D are incorrect. B is incorrect because the BPV is located downstream of the flow restrictors and will have no effect on indicated steam flow. A is correct because the open RV causes indicated steam flow to decrease and a corresponding decrease in feed flow until the level signal increase feed flow and stabilizes level at a new lower level.

COMMENTS:

QUESTION # 121 EXAM KEY
10/04/2002
ex02084

Several minutes ago several alarms were received in the control room. Upon investigation, the control room operator finds Standby Gas Treatment in operation controlling reactor building pressure. Reactor Building HVAC has tripped. The reactor continues to operate at 99% power.

Which of the following caused these indications?

A. 1.72 psig in the drywell.

B. +3 inches reactor level.

C. Leaking fuel in the spent fuel pool.

D. High reactor building dp.

ANSWER: C

QUESTION TYPE: RO

KA # & KA VALUE: 261000A2.12 3.2/3.4 10CFR55.41 & 45 - Ability to (a) predict the impacts of the following on the STANDBY GAS TREATMENT SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: High fuel pool ventilation radiation.

REFERENCE: 82-RSY-1000-T6 rev. 8, pages1 & 10 LO000144 rev. 11, page14

SOURCE: **NEW QUESTION –** RO T2, GP1, #26

LO: 5679 – Describe the RB HVAC system response to the LOCA isolation signals - FAZ

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: A and B are both incorrect because they would result in a reactor scram. D is incorrect because it does not result in a start of SGT. C is correct because the hi rad/airborne contamination would enter the RB HVAC system from the suction over the fuel pool and cause a Z signal when the ventilation exhaust exceeded 13 mrem/hr.

COMMENTS:

QUESTION # 123 EXAM KEY
10/04/2002
ex01103

The plant is operating at 89% power. DG-1 has been started and loaded per the monthly operability surveillance. During the operability run, Drywell pressure increases to 2.02 psig. Five minutes later, a loss of all offsite power occurs.

Which of the following is correct for these conditions?

A. DG-1 continues to run until the trip of DG1-7 on the loss of offsite power.

B. DG-1 trips and has to be manually restarted. DG1-7 is manually synced to SM-7.

C. DG1-7 trips, DG-1 continues to run, DG1-7 auto closes when the loss of all offsite power occurs.

D. DG1-7 trips, DG-1 trips, DG-1 restarts and DG1-7 auto closes when the loss of all offsite power occurs.

ANSWER: C

QUESTION TYPE: RO

KA # & KA VALUE: 264000A2.09 3.7/4.1 10CFR55.41 & 45 - Ability to (a) predict the impacts of the following on the EMERGENCY GENERATORS (DIESEL/JET); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of AC power.

REFERENCE: LO000200 rev. 8, pages 20 & 21

SOURCE: **BANK QUESTION 2001 NRC exam–** RO T2, GP1, #28

LO: 5317 – List the four trip signals for DG1-7.

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: During the operability run, DG-1 is paralleled to the Backup Transformer. When the high drywell pressure occurs, DG1-7 trips and DG-1 continues to run. When the loss of offsite power occurs, DG1-7 recloses automatically. C is correct.

COMMENTS:

At which of the following power levels will the change in neutron flux cause the most damage during a control rod drop accident?

A. 5

B. 20

C. 35

D. 50

ANSWER: A

QUESTION TYPE: RO

KA # & KA VALUE: 201003K3.02 2.8/3.1 10CFR55.41 & 45 - Knowledge of the effect that a loss or malfunction of the CONTROL ROD AND DRIVE MECHANISM will have on following: Flux shaping

REFERENCE: LO000137 rev. 11, page3 LO000154 rev.10, pages 2 & 3

SOURCE: **NEW QUESTION –** RO T2, GP2, #1

LO: 7283 – State the purpose of flux shaping and rod sequencing.

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: The damage caused during a control rod drop accident is caused by the change in flux or reactivity. When power exceeds 20% power, there is no operator error that can cause fuel enthalpy to exceed 280 cal./gram. A is correct.

COMMENTS:

The plant is operating at rated power when a fault causes an automatic power reduction to approximately 60% of rated.

Assuming no operator actions were taken, which of the following would result in these conditions?

A. SH-6 trip.

B. SM-1 trip.

C. SM-3 trip

D. SM-8 trip

ANSWER: A

QUESTION TYPE: RO

KA # & KA VALUE: 202001A4.01 3.7/3.7 10CFR55.41 & 45 - Ability to manually operate and/or monitor in the control room: Recirculation pumps

REFERENCE: LO000196 rev. 12, page 32

SOURCE: **BANK QUESTION – 2000 NRC EXAM** - RO T2, G2, #5

LO: 5030 – State the power supplies for the Recirc pumps.

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: A is correct because the loss of power to RRC-P-1B would result in a power reduction to approximately 60% power. B and C would cause a full scram due to a loss of suction pressure trip on the feedpumps (with no operator action). D would cause a 1/2 scram but no reduction in power.

COMMENTS:

QUESTION # 127 EXAM KEY
10/04/2002
ex02072

The plant was operating at 100% power when a scram occurred due to an MSIV isolation.

Which of the following caused this isolation?

A loss of …

A. RPS-A

B. RPS-B

C. IN-2

D. IN-3

ANSWER: C

QUESTION TYPE: RO

KA # & KA VALUE: 239001K2.01 3.2/3.3 10CFR55.41 - Knowledge of electrical power supplies to the following: MSIV solenoids

REFERENCE: LO000173 rev. 9, page 13

SOURCE: **NEW QUESTION –** SRO T2, GP2, #12

LO: 7783 – Predict the effect a failure of IN2 will have on MSIVs.

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: A, B, and C, are both incorrect because they only de-energize ½ of the logic design needed to cause a full isolation. IN-2 both sub channels of the inboard MSIVs causing all 4 to close and a full scram. C is correct.

COMMENTS:

The plant is operating at 99% power with LPCS-P-1 running in full flow test for a surveillance. A small leak in the drywell causes drywell pressure to increase to the LPCS initiation setpoint.

Which of the following valves responds first?

A. LPCS-V-1 Suction

B. LPCS-V-5 Injection

C. LPCS-FCV-11 Minimum Flow

D. LPCS-V-12 Test Return

ANSWER: D

QUESTION TYPE: RO

KA # & KA VALUE: 295024EA1.03 4.0/3.9 10CFR55.41 & 45 - Ability to operate and/or monitor the following as they apply to HIGH DRYWELL PRESSURE: LPCS

REFERENCE: LO000192 rev. 9 pages 4 and 5

SOURCE: **BANK QUESTION – LR00688 –** RO T1, GP1, #7

LO: 5482 – List the automatic system response when LPCS initiates.

RATING: H2

ATTACHMENT: NONE

JUSTIFICATION: A is incorrect because there is no auto signal for V-1. B is incorrect because the injection valve does not open until reactor pressure decreases to less than 470 psig. C is incorrect because the min flow valve does not open until flow decreases. D is correct: as soon as the initiation occurs, the Test Return valve closes.

COMMENTS:

QUESTION # 129 EXAM KEY
10/04/2002
ex02074

Which of the following is prevented by the performance of the PC Gas leg of PPM 5.2.1 Primary Containment Control?

A. Damage to Standby Gas Treatment from excessive hydrogen concentration.

B. An uncontrolled release of radioactivity to the environment.

C. Damage to drywell equipment from drywell sprays.

D. A failure of the drywell downcomers.

ANSWER: B

QUESTION TYPE: RO

KA # & KA VALUE: 500000 2.3.11 2.7/3.2 10CFR55.45 – Ability to control radiation release during high containment hydrogen.

REFERENCE: PPM 5.0.10 rev. 6 page 267

SOURCE: **NEW QUESTION –** RO T1, GP1, #13

LO: 8425 – Identify the possible consequence of a deflagration in containment.

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: PPM 5.0.10 states the reason/basis for the PC Gas control leg of PPM 5.2.1 is to prevent the uncontrolled release of radioactivity to the environment. B is correct.

COMMENTS:

The plant is operating at 99% power. The following conditions exist:

Five cooling towers are in operation.
Six Circ Water fans are running on each tower.
Two Circ Water pumps are running.
Condenser vacuum is going down.

Which of the following is true concerning these conditions?

A. Starting all available cooling tower fans causes main generator output to go down because of an increase in plant auxiliary power.

B. Starting all available cooling tower fans causes main generator output to go up because of an increase in Circ Water flow.

C. Starting the third Circ Water Pump causes main condenser vacuum to go down because of an increase in Circ Water flow.

D. Starting the third Circ Water Pump causes main condenser vacuum to go up because of an increase in condensate sub cooling.

ANSWER: D

QUESTION TYPE:	RO
KA # & KA VALUE:	295002AA1.07 3.1/2.9 10CFR55.41 & 45 - Ability to operate and/or monitor the following as they apply to LOSS OF MAIN CONDENSER VACUUM: Condenser Circ Water System
REFERENCE:	BWR Gen. Fun. Thermodynamics, Chap. 4, rev. 3, pages35 & 36
SOURCE:	**NEW QUESTION –** RO T1, GP2, #2
LO:	7393 – Explain vacuum formation in the condenser process.
RATING:	L2
ATTACHMENT:	NONE
JUSTIFICATION:	Vacuum can be increased (back pressure decreased) by increasing the condensate sub cooling. Sub cooling can be increased by increasing CW flow or decreasing CW temperature. D is correct because vacuum increases by increasing CW flow (starting the 3rd pump). C is incorrect because vacuum increases. B is incorrect because starting all available fans does not increase CW flow. A is incorrect because main generator output goes up not down.
COMMENTS:	

The plant was operating at 25% power when an ATWS occurred. You have been directed to insert control rods per PPM 5.5.11 Alternate Control Rod Insertions by the CRS.

Which of the following is correct concerning these conditions?

A. The continuous in pushbutton can be used for control rod insertion.

B. Single notch insertion must be used for control rod insertion.

C. Control rods must be inserted according to the fast shutdown sequence.

D. All control rods must be inserted to position 12 before insertion to position 00.

ANSWER: A

QUESTION TYPE: RO

KA # & KA VALUE: 201002 2.4.6 3.1/4.0 10CFR55.41, 43.5, & 45 – Knowledge of the symptom based EOP mitigation strategies – Reactor Manual control system.

REFERENCE: PPM 5.0.10 rev. 6, page 200

SOURCE: **NEW QESTION –** RO T2, GP1, #1

LO: NO LO

RATING: L3

ATTACHMENT: NONE

JUSTIFICATION: Per the EOP IMPLEMENTATION POLICY given in PPM 5.0.10, the CONTINUOUS IN pushbutton can be used for control rod insertion directed by PPM 5.5.11. A is correct.

COMMENTS:

With reactor level GT TAF, operation of the LPCS Pump is **NOT** allowed with suppression pool level LT the Vortex Limit of the pump.

Which of the following describes the reason for this limit?

A. Loss of NPSH results in pump run out and motor overheating.

B. Loss of NPSH results in a pump trip from low suction pressure.

C. Air entrainment can cause pitting and failure in the spray ring nozzles.

D. Air entrainment could occur and cause system damage during subsequent restarts.

ANSWER: D

QUESTION TYPE: RO

KA # & KA VALUE: 209001K6.03 3.3/3.4 10CFR55.41 & 45 - Knowledge of the effect that a loss or malfunction of the following will have on the LOW PRESSURE CORE SPRAY SYSTEM: Suppression Pool water level

REFERENCE: PPM 5.0.10 rev. 6 pages 262 & 263

SOURCE: **BANK QUESTION – 2001 NRC EXAM –** RO T2, GP1, #5

LO: 8388 - Given a list, identify the statement that describes a centrifugal pump's response to operation below its Vortex Limit.

RATING: H2

ATTACHMENT: NONE

JUSTIFICATION: A is incorrect because a loss of NPSH does not cause pump run out and overheating. B is incorrect because there is no low suction pressure trip on LPCS. C is incorrect because the spray ring is in a high pressure area. D is correct as stated in PPM 5.0.10.

COMMENTS: This question was used for the 2001 exam but with HPCS as the system in question. The only changes to the question were based on LPCS as the system in question versus HPCS. This is considered a direct bank question.

A plant startup is in progress with the following conditions:

Reactor power is approximately 3%
IRM A = 75/125 R8
IRM B = 39/40 R7
IRM C = 65/125 R8
IRM D = 25/40 R7
IRM E = 59/125 R8
IRM F = 60/125 R8
IRM G = 47/125 R8
IRM H = 35/125 R8

The High Voltage supply for IRM E then fails.

Which of the following is correct concerning these conditions?

A. There is no effect on plant operation.

B. Full scram.

C. Rod out Block.

D. ½ scram on RPS A.

ANSWER: B

QUESTION TYPE: RO

KA # & KA VALUE: 215003K3.04 3.6/3.6 10CFR55.41, & 45 - Knowledge of the effect that a loss or malfunction of the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM will have on following: Reactor Protection System.

REFERENCE: LO000138 rev. 7, page 13
SOURCE: **NEW QUESTION –** RO T2, GP1, #11
LO:
RATING: 5459 – List the IRM scrams and rod blocks
ATTACHMENT: H3
JUSTIFICATION: NONE
The loss of the HV power supply for IRM E causes an INOP scram signal on RPS A. IRM B already has generated a ½ scram on RPS B, so the combination results in a full scram. B is correct.

The plant was operating at 89% power when a transient occurred. The CRS has directed the CRO to open the seven ADS SRVs by arming and depressing the A and C Logic Channel pushbuttons. When the CRO pushes the pushbuttons, the seven ADS SRVs open immediately. All seven ADS SRVs close immediately upon release of the pushbuttons by the CRO.

Which one of the following is correct concerning these conditions?

A. RHR-P-2A is not running.

B. RHR-P-2C is not running.

C. The Division 2 Inhibit switch is in the INHIBIT position.

D. The Division 1 Inhibit switch is in the INHIBIT position.

ANSWER: D

QUESTION TYPE: RO

KA # & KA VALUE: 218000A4.02 4.2/4.2 10CFR55.41 & 45 - Ability to manually operate and/or monitor in the control room: ADS initiation logic.

REFERENCE: LO000186 rev. 9, page 4

SOURCE: **BANK QUESTON – 2001 NRC EXAM –** RO T2, GP1, #18

LO: 5073 – State how ADS is manually initiated including permissive.

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: With all ADS logic made up (all auto contacts made up and the 105 second timer timed out) and the INHIBIT Switches in inhibit, there is no auto initiation. If all ADS logic is made up and the Arm and Depress logic pushbuttons are pushed with the INHIBIT Switches in inhibit, the valves open. When the pushbutton is released, the valves close. D is correct.

COMMENTS:

The plant is operating at 100% power. Multiple alarms illuminate on FCP-1, 2, and 3. Immediately following these alarms a plant laborer calls and informs you there is smoke billowing out of the RFW-P-1B room.

Which of the following is correct for these conditions?

Immediately…

A. Evacuate all non-emergency personnel from the TG Building.

B. Call OPS 3 to verify the fire.

C. Notify the Hanford FD by use of the pushbutton on FCP-1

D. Notify SCC of the location of the fire.

ANSWER: A

QUESTION TYPE: RO

KA # & KA VALUE: 259001 2.4.25 2.9/3.4 10CFR55.41 & 45 – Knowledge of Fire Protection procedures – Reactor Feedwater Systems

REFERENCE: ABN-FIRE rev. 3, page 2

SOURCE: **NEW QUESTION –** RO T2, GP1, #22

LO: 6902 – Describe the immediate actions for a fire.

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: B is incorrect because someone has already verified the existence of a fire. C is incorrect because SCC notifies the Hanford FD. D is incorrect because notification of SCC is an immediate action if the fire is outside of the power block. The immediate actions for ABN-FIRE require that all non-emergency personnel be evacuated from the affected building immediately. A is correct.

COMMENTS:

The plant is operating at 100% power. A loss of power occurs on SM-8 due to a lockout.

Which of the following is correct for this condition?

APRM power is indicated on…

A. all normal power indicators on P603.

B. Division 1 power indications on P603.

C. Division 2 power indications on P603.

D. Graphic Display System on P602.

ANSWER: D

QUESTION TYPE: RO

KA # & KA VALUE: 215005A4.02 2.8/2.8 10CFR55.41 & 45 - Ability to manually operate and/or monitor in the control room: CRT display indicators

REFERENCE: ABN-ELEC-SM3/SM8 rev. 0, page 19 GDS Design Spec. SGPS9401 rev/ 2, APP A table 1.1, page 1

SOURCE: **NEW QUESTION –** RO T2, GP1, #14

LO: NO LO

RATING: L3

ATTACHMENT: NONE

JUSTIFICATION: The loss of SM-8 causes a loss of all reactor power indication on P603 and a loss of Div 2 APRMs. The GDS system selects the highest power input it receives from the operable APRMs (Div 1) and displays that value. D is correct.

COMMENTS:

During the last refuel outage, two holes (5 inches by 6 inches) were opened in the floor under H13-P682 in the main control room. The plant was started up and is now operating at 97% power.

Which of the following describes the operational implication of operating the plant in this configuration?

A. The normal Control Room Ventilation system flow balance will be changed and will not be able to cool all control room equipment.

B. The Control Room Deluge Systems will not be effective because of the change in the spray patterns caused by the change in ventilation.

C. The Control Room Emergency Filtration System will not be able to pressurize the control room when required during emergency conditions.

D. The Control Room Emergency Filtration System will not be able to filter the incoming atmosphere due to the bypass flow path created.

ANSWER: C

QUESTION TYPE: RO

KA # & KA VALUE: 290003K5.02 2.8/2.8 10CFR55.41 & 45 - Knowledge of the operational implications of the following concepts as they apply to CONTROL ROOM HVAC: Differential pressure control.

REFERENCE: LO000201 rev. 9, pages 2 & 3 **Columbia LER 94-021**

SOURCE: **NEW QUESTON -** RO T2, GP2, #19

LO: 5220 – State the purpose of the Control Room HVAC system.

RATING: H4

ATTACHMENT: NONE

JUSTIFICATION: A is incorrect because there would be no change in the flow balance or cooling for control room equipment. B is incorrect because there are not deluge systems in the control room. D is incorrect because the holes in the floor are downstream from the filters and would have no effect on filtration. C is correct as stated in the LER.

COMMENTS:

The following conditions exist:

Isolated ATWS
Reactor level +19 inches and down slow
Reactor pressure 999 psig and up slow
Suppression pool level 31 feet and up slow
Suppression pool temperature 220°F and going up

Which of the following is correct for the above conditions?

A. Start RCIC for reactor level control.

B. Reduce reactor pressure to 600 psig to allow injection with Condensate Booster Pumps.

C. Immediately Emergency Depressurize the reactor.

D. Start HPCS and reduce suppression pool level.

ANSWER: C

QUESTION TYPE: RO

KA # & KA VALUE: 295025EA2.03 3.9/4.1 10CFR55.41, 43.5, &45 - Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE: Suppression Pool Temperature.

REFERENCE: PPM 5.2.1, rev 12 - HCTL

SOURCE: **BANK QUESTION – 98 NRC EXAM -** RO T1, G1, #8

LO: 8302 – Given plant conditions, determine current operating point on the HCTL curve.

RATING: H3

ATTACHMENT: YES - PPM 5.2.1
JUSTIFICATION: A and B are both incorrect because level control is secondary to protecting the containment at this time. D is incorrect because suppression pool level is normal and a level reduction would only hurt at this time.
COMMENTS:

QUESTION # 142 EXAM KEY
10/04/2002
ex01113

Which of the following describes how low pressure LPCI Injection piping is protected from full reactor pressure?

RHR-V-42A (42B and 42C) are interlocked closed until reactor pressure is less than…

A. 160 psig.

B. 220 psig

C. 320 psig.

D. 470 psig.

ANSWER: D

QUESTION TYPE: RO

KA # & KA VALUE: 203000K4.02 3.3/3.4 10CFR55.41.7 - Knowledge of RHR/LPCI: INJECTION MODE design feature(s) and/or interlocks which provide for the following: Prevention of piping overpressurization.

REFERENCE: LO000198 rev. 10, page 3

SOURCE: **BANK QUESTION – 2001 NRC EXAM –** T2, GP1, #4

LO: 7728 – Describe the physical connections and/or cause and effect relationships between RHR and the RPV.

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: The LPCI Injection valves are interlocked closed to prevent overpressurization until reactor pressure is reduced to less than 470 psig. D is correct.

COMMENTS:

The plant was operating at 97% power when a LOCA occurred. The LOCA signal is sealed in and has not been reset. All plant equipment functioned as designed. RHR-P-2A is in operation in Upper Drywell Spray. RHR-P-2B is in operation in Wetwell Spray. A lockout on Bkr 7-1 then causes the Startup Transformer to trip.

Which of the following is correct for these conditions?

A. RHR-P-2A is in operation with power from the Backup transformer. RHR-P-2B is in operation with power from the Backup Transformer

B. RHR-P-2A is in operation with power from DG-1. RHR-P-2B is in operation with power from DG-2.

C. RHR-P-2A is off. RHR-P-2B is in operation with power from the Backup Transformer

D. RHR-P-2A is off. RHR-P-2B is in operation with power from DG-2.

ANSWER: C

QUESTION TYPE: RO

KA # & KA VALUE: 226001K2.02 2.9/2.9 10CFR55.41.7 - Knowledge of electrical power supplies to the following: Pumps

REFERENCE: LO000182 rev. 12. page 30 LO000198 rev. 10, page 43

SOURCE: **BANK QUESTION – 2001 NRC EXAM –** RO T2, GP2, #11

LO: 5058 – Identify the loads on SM-7.

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: RHR-P-2A is powered from SM-7. With the 86 on Bkr 7-1, neither the DG nor the Backup transformer close onto the bus. RHR-P-2A is not in operation. The loss of TR-S causes an undervoltage on SM-8. Bkr B-8 closes and supplies the bus. RHR-P-2B is in operation. C is correct.

COMMENTS:

QUESTION # 145 EXAM KEY
10/04/2002
ex99097

The plant is in MODE 5 with a full core off load 98% completed. RHR-P-2B is in operation in Fuel Pool Cooling Assist mode. Breaker 8-3 trips due to overcurrent.

Which of the following is correct based on these conditions?

A. DG-2 starts and supplies bus SM-8.

B. Breaker B-8 closes and supplies bus SM-8.

C. The spent fuel pool temperature begins to increase.

D. RHR-P-2C will be started in Fuel Pool Cooling Assist Mode.

ANSWER: C

QUESTION TYPE: RO

KA # & KA VALUE: 233000K2.02 2.8/2.9 10CFR55.41.7 - Knowledge of electrical power supplies to the following: RHR pumps

REFERENCE: LO000202 page 14

SOURCE: **BANK QUESTION- 99 NRC EXAM** RO T2, G3, #3

LO: 5371- Predict how FPCC responds to a loss of RHR.

RATING: H3

ATTACHMENT: N/A

COMMENTS: A trip of breaker 8-3 on overcurrent causes a lockout and none of the power supplies close onto the bus. The result is a loss of RHR-P-2B and a heatup of the spent fuel pool. RHR-P-2C cannot be lined up for FPC Assist. C is correct.

JUSTIFICATION:

ex98086

The reactor is in MODE 5 with fuel movement underway. After moving a bundle through the "cattle chute" and into the vessel cavity, it is observed that the "ROD BLOCK INTERLOCK #1" light does not illuminate. The "HOIST LOADED" indicator is illuminated. The control room reports no rod block indication.

Which of the following actions is correct for these conditions?

A. Immediately stop the refuel bridge until the inoperable rod block is corrected.

B. Immediately initiate action to insert all insertable control rods in core cells containing one or more fuel assemblies.

C. The fuel bundle may be moved back to the spent fuel pool, then immediately suspend in-vessel fuel movement.

D. Fuel movement may continue as long as ROD BLOCK INTERLOCK #2 is operable.

ANSWER: C

QUESTION TYPE: RO

KA # & KA VALUE: 2.1.11 3.0/3.8 10CFR55.43.2 &45 – Knowledge of less than 1 hour Tech Spec Action statements for systems.

REFERENCE: TS 3.9.1 and TS Bases 3.9.1

SOURCE: **BANK QUESTION – 98 NRC EXAM –** RO T3, #4

LO: 6926 – State from memory, Tech Spec actions required to be taken in less than 15 minutes.

RATING: H3 H4
ATTACHMENT: NONE
JUSTIFICATION: C is correct because TS Bases specifically allow the movement of the component to a safe condition even though the action is for immediate suspension of in vessel fuel movement. Tech Specs does not allow movement to continue until a second fault occurs.
COMMENTS:

The plant is operating at 98% power. At 1500 Wednesday, RHR-P-2C is declared inoperable due to a motor failure. At 1900 Wednesday, DG-1 and all systems supported by the diesel are declared inop.

Which of the following is correct concerning these conditions?

A. Restore RHR-P-2C to operable status in 7 days from 1500 Wednesday.

B. Restore DG-1 to operable status by 1900 Thursday.

C. Perform SR 3.8.1.1 for OPERABLE offsite circuits by 2000 Wednesday, and restore DG-1 to OPERABLE status by 0700 Saturday.

D. Take action within 1 hour (from 1900 Wednesday) to place the unit in MODE 2 within 7 hours, MODE 3 within 13 hours and MODE 4 within 37 hours.

ANSWER: D

QUESTION TYPE: RO

KA # & KA VALUE: 2.2.23 2.6/3.8 10CFR55.43.2 & 45 – Ability to track limiting conditions for operations.

REFERENCE: TS. 3.0.3, 3.5.1

SOURCE: **BANK QUESTION - 2000 NRC EXAM –** RO T3, #7

LO: 9540 – Interpret required Tech Specs from plant conditions.

RATING: H4

ATTACHMENT: YES - TS. 3.0.3, 3.5.1

JUSTIFICATION: When DG-1 and all of its supported systems are declared inop, 3 ECCS Systems are out of service and require entry into TS 3.0.3. D is correct.

COMMENTS: This question replaces EX00098 on the RO Exam as number 67.

ex98022

The reactor was operating at 100% power when a loss of feedwater caused a reactor scram approximately 5 minutes ago. HPCS and RCIC automatically started and are restoring reactor level. Reactor level is +10 inches and up slow.

Which of the following level instruments is most "accurate" in this situation?

A. Narrow Range

B. Wide Range

C. Upset Range

D. Shutdown Range

ANSWER: B

QUESTION TYPE: RO

KA # & KA VALUE: 295007AA2.03 3.7/3.7 10CFR55.41, 43.5, & 45 - Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE: Reactor water level

REFERENCE: LO000126 rev. 8, page 4

SOURCE: **BANK QUESTION – 98 NRC EXAM -** RO T1, G1, #3

LO: 5582 – List the calibration conditions for the wide range.

RATING: H2

ATTACHMENT: NONE

JUSTIFICATION: A and C are incorrect because the narrow range and upset ranges are calibrated with core flow. D is incorrect because the shutdown range is calibrated at 0 psig and 212°F coolant temperature.

COMMENTS:

QUESTION # 149 EXAM KEY
10/04/2002
ex02085

The pressure control leg of PPM 5.1.1 RPV Control directs the operator to open SRVs and reduce reactor pressure if any SRV is cycling.

Which of the following is the basis for this direction?

Manual opening of SRVs…

A. minimizes significant dynamic loads on the RPV and containment structures.

B. prevents reactor water level from going above the feedpump trip setpoint.

C. prevents reactor water level from going below the scram setpoint.

D. reduces stress induced on the Main Condenser.

ANSWER: A

QUESTION TYPE: RO

KA # & KA VALUE: 295007AK3.04 4.0/4.1 10CFR55.41 & 45 - Knowledge of the reasons for the following responses as they apply to HIGH REACTOR PRESSURE: Safety relied valve operation.

REFERENCE: PPM 5.0.10 rev. 6, page 128

SOURCE: **NEW QUESTION –** RO T1, GP1, #2

LO: 8053 – List advantages of reducing RPV pressure when an SRV is cycling.

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: B and C are both incorrect because manual opening of the SRVs may cause both of these setpoints to be exceeded. D is incorrect because the pressure is not supposed to be reduced below the DEH setpoint. This results in no change in the total steam flow to the condenser. A is correct as stated in PPM 5.0.10

COMMENTS:

Which of the following neutron flux trip signals models the fuel thermal transient characteristics?

A. SRM Flux Hi

B. IRM Flux Hi

C. APRM Flow Biased

D. APRM Neutron

ANSWER: C

QUESTION TYPE: RO

KA # & KA VALUE: 212000K5.01 2.7/2.9 10CFR55.41 & 45 - Knowledge of REACTOR PROTECTION SYSTEM design feature(s) and/or interlocks which provide for the following: Bypassing of selected scram signals.

REFERENCE: LO000161 rev. 11, page 11

SOURCE: **NEW QUESTION –** RO T2, GP1, #9

LO: 7682 – Describe the physical connection and/or cause and effect relationship between RPS and Neutron Monitoring.

RATING: L3

ATTACHMENT: NONE

JUSTIFICATION: A time constant of 6 seconds = or – 1 second is applied to the APRM signal to more closely approximate the thermal characteristics of the fuel. This is the APRM Flow Biased trip. C is correct. A, B, and C are all incorrect because the signals are not corrected to the fuel thermal characteristics.

COMMENTS:

The plant is in MODE 5 with the refuel bridge loaded and over the core. A failure in the Electronic Load Cell causes the sensed load to decrease and the HOIST LOADED Light to go out.

Which of the following is correct for these conditions?

A. A rod block is generated in the control room.

B. The rod withdraw block indicated in the control room clears.

C. The refuel bridge stops at the location where the failure occurred and can only be moved back into the Spent Fuel Pool.

D. The refuel bridge stops at the location where the failure occurred and cannot be moved until the fault is cleared.

ANSWER: B

QUESTION TYPE: RO

KA # & KA VALUE: 234000K3.01 2.9/3.9 10CFR55.41 & 45 - Knowledge of the effect that a loss or malfunction of the FUEL HANDLING EQUIPMENT will have on following: RMCS

REFERENCE: LO00207 rev. 9, page 9, 26, 31, and 32 LO000148 rev. 10, page 13

SOURCE: **NEW QUESTION –** RO T2, GP3, #2

LO: 5359 – Refuel interlocks in effect when a control rod is withdrawn and Mode Switch is in REFUEL.

RATING: L3

ATTACHMENT: NONE

JUSTIFICATION: C and D are both incorrect because the failure does not cause the bridge to stop. The failure causes the existing rod block to clear. B is correct.

COMMENTS:

Which of the following is a control room operator responsibility?

A. Classify the event.

B. Notify offsite agencies of the event.

C. Function as the Emergency Director in the absence of the Shift Manager.

D. Inform the Shift Manager if any parameter exceeds the emergency action levels.

ANSWER: D

QUESTION TYPE: RO

KA # & KA VALUE: 2.4.39 3.3/3.1 10CFR55.45 – Knowledge of RO responsibilities in emergency plan implementation.

REFERENCE: PPM 13.1.1 rev. 31, pages 3 & 4

SOURCE: **NEW QUESTION –** RO T3, #10

LO: 6130 – Describe the duties of the CRO during emergencies.

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: 13.1.1 states the CRO has the responsibility to notify the Shift Manager of any condition/parameter that exceeds an EAL. D is correct. A, B, and C are incorrect because they are duties/responsibilities of other control room personnel.

COMMENTS:

2002 EXAM MATRIX

SRO #	QUESTION	RO#
1	Ex02006	1
2	Ex98030	2
3	Ex02001	3
4	Ex98103	4
5	Ex02002	
6	Ex00012	5
7	Ex02003	6
8	Ex00073	7
9	Ex02007	8
10	Ex02004	9
11	Ex00091	10
12	Ex02008	11
13	Ex99099	12
14	Ex02009	13
15	Ex00132	14
16	Ex02010	15
17	Ex02005	16
18	Ex02011	17
19	Ex00042	18
20	Ex02012	19
21	Ex00102	20
22	Ex02013	21
23	Ex01126	22
24	Ex02014	
25	Ex02015	
26	Ex02016	23
27	Ex01062	
28	Ex02017	24
29	Ex00121	25
30	Ex02018	26
31	Ex02090	
32	Ex02019	27
33	Ex02020	
34	Ex02021	28
35	Ex02022	29
36	Ex02023	30
37	Ex02024	31
38	Ex01087	32
39	Ex02025	
40	Ex01039	33
41	Ex02026	34
42	Ex98040	35

43	Ex02027	36
44	Ex00004	37
45	Ex02028	38
46	Ex02029	39
47	Ex02032	40
48	Ex02030	41
49	Ex02031	42
50	Ex02082	
51	Ex02033	43
52	Ex00041	44
53	Ex02034	45
54	Ex00099	46
55	Ex98001	47
56	Ex02035	48
57	Ex02036	49
58	Ex02037	
59	Ex00100	
60	Ex02089	
61	Ex00011	
62	Ex00062	50
63	Ex00115	51
64	Ex02038	52
65	Ex02039	53
66	Ex02040	
67	Ex02041	54
68	Ex02042	55
69	Ex02043	56
70	Ex02044	57
71	Ex02045	
72	Ex02046	
73	Ex00044	58
74	Ex00019	59
75	Ex02047	
76	Ex02048	60
77	Ex02069	
78	Ex00038	
79	Ex02049	61
80	Ex02050	62
81	Ex02051	63
82	Ex02052	
83	Ex02053	64
84	Ex98116	65
85	Ex02054	66
86	Ex02055	
87	Ex02056	

88	Ex00039	67
89	Ex02059	68
90	Ex02058	
91	Ex02060	69
92	Ex02061	70
93	Ex02062	
94	Ex02070	
95	Ex02063	71
96	Ex02064	
97	Ex02065	
98	Ex02066	72
99	Ex02067	73
100	Ex02068	
	Ex02069	74
	Ex02084	75
	Ex01103	76
	Ex02071	77
	Ex00003	78
	Ex02072	79
	Ex02073	80
	Ex02074	81
	Ex02075	82
	Ex02077	83
	Ex00053	84
	Ex02078	85
	Ex01042	86
	Ex02079	87
	Ex02080	88
	Ex02081	89
	Ex98023	90
	Ex01113	91
	Ex01128	92
	Ex99097	93
	Ex98096	94
	Ex00135	95
	Ex98022	96
	Ex02085	97
	Ex02086	98
	Ex02087	99
	Ex02088	100

Facility: Columbia Generating Station	Task No: RO-287
Task Title: Start SGT B for Containment Venting	Job Performance Measure No: JPMB.1.Ar0
K/A Reference: 261000A4.07 3.1/3.2	
Examinee:	NRC Examiner:
Facility Evaluator:	Date:

Method of testing:

Simulator - Perform

JPM SETUP INFORMATION

Initial Conditions:	A drywell atmosphere sample has been analyzed by chemistry within the last hour. Sample results permit containment venting, Both SGT trains are operable Reactor Building HVAC is operational. Drywell Noble Gas monitor indication is well below the High-High setpoint and has not changed in the past hour. Steps in PPM 2.3.1 have been complete through step 5.1.5.
Task Standard:	Preparations for venting the containment are performed in accordance with PPM 2.3.1.
Required Materials:	N/A
General References:	PPM 2.3.1 rev. 39, section 5.1
Initiating Cue:	You have been directed by the CRS to start SGT “B” in preparation for venting the drywell in accordance with PPM 2.3.1, starting at step 5.1.6. Inform the CRS when you are ready to open CEP-V-1B.
Time Critical Task:	NO
Validation Time:	8 minutes
Simulator ICs:	198
Malfunctions/Remote Triggers:	Loaded in IC-198
Overrides:	Loaded in IC-198
Special Setup Instructions:	Loaded in IC-198

PERFORMANCE INFORMATION

START TIME:

Critical Step: **YES**	
Performance Step: 1	5.1.6: At H13-P813 (Bd H), ensure CEP-V-11 is closed.
CUE:	**If informed as CRS that CEP-V-11 is open, tell examinee to close CEP-V-11.**
Standard:	CEP-V-11 is closed.
Comment: SAT / UNSAT	

Critical Step: NO	
Performance Step: 2	5.1.7: Ensure SGT-V-2B is open.
CUE:	
Standard:	Verifies SGT-V-2B is open.
Comment: SAT / UNSAT	

Critical Step: NO	
Performance Step: 3	5.1.8: If SGT is required to be operable, enter SGT B an inoperable in the Plant Logging System.
CUE:	**SGT is logged as inoperable in the Plant Logging System.**
Standard:	CRS is notified.
Comment: SAT / UNSAT	

Critical Step: **YES**	
Performance Step: 4	5.1.10: Place SGT-DPIC-1B2 in the MANUAL mode and adjust the controller output to 90% o 95%.
CUE:	
Standard:	SGT-DPIC-1B2 in the MANUAL mode between 90% and 95%.
Note;	100% output is closed, 0% output is open.
Comment: SAT / UNSAT	

Critical Step: **YES**	
Performance Step: 5	5.1.11: Open SGT-V-5B2.
CUE:	
Standard:	OpenSSGT-V-5B2.
Comment: SAT / UNSAT	

Critical Step: **YES***	
Performance Step: 6	5.1.12: Momentarily turn SGT-EHC-1B2 control switch to on*and observe SGT-FN-1B2 automatically starts 10 seconds after heaters are energized.
CUE:	
Standard:	Places control switch to on and observes fan start.
Comment: SAT / UNSAT	

Critical Step: **YES**	
Performance Step: 7	5.1.13: Adjust SGT-DPIC-1B2 to obtain a SGT flowrate of approximately 5000 CFM.
CUE:	
Standard:	SGT flowrate established at approximately 5000 CFM
Comment: SAT / UNSAT	

Critical Step: NO	
Performance Step: 8	4.1.14: Place SGT-DPIC-1B2 in AUTO with the controller set at –1.7 inches.
CUE:	
Standard:	SGT-DPIC-1B2 set at –1.7 inches.
Comment: SAT / UNSAT	

Critical Step: NO	
Performance Step: 9	5.1.15; If SGT is required to be operable, enter SGT B as operable in the Plant Logging System.
CUE:	**SGT is logged in the Plant Logging System.**
Standard:	CRS is notified to log SGT in Plant Logging System.
Comment: SAT / UNSAT	

THE EXAMINEE SHOULD ANNOUNCE THE TERMINATION POINT OF THE JPM AT THIS POINT.

JPM TERMINATION TIME:	
JPM START TIME:	
-	______________________________
JPM COMPLETION TIME:	

VERIFICATION OF COMPLETION

JPM Number:	JPMB.1.Ar0
Examinee's Name:	
Examiner's Name:	
Date Performed:	
Facility Evaluator:	
Number of Attempts:	
Time to Complete:	

JPM INFORMATION CARD

HAND THE STUDENT INFORMATION CARD TO THE EXAMINEE

READ TO THE EXAMINEE:
I will explain the initial conditions, which steps to simulate or discuss, and provide initiation cues. When you complete the task successfully, the objective for this job performance measure will be satisfied.

Task Standard:	Preparations for venting the containment are performed in accordance with PPM 2.3.1.
Required Materials:	N/A
Safety Equipment:	N/A
General References:	PPM 2.3.1 rev. 39, section 5.1
Time Critical Task:	NO
Initial Conditions:	A drywell atmosphere sample has been analyzed by chemistry within the last hour. Sample results permit containment venting, Both SGT trains are operable Reactor Building HVAC is operational. Drywell Noble Gas monitor indication is well below the High-High setpoint and has not changed in the past hour. Steps in PPM 2.3.1 have been complete through step 5.1.5.

INITIATING CUE

You have been directed by the CRS to start SGT “B” in preparation for venting the drywell in accordance with PPM 2.3.1, starting at step 5.1.6. Inform the CRS when you are ready to open CEP-V-1B.

INFORMATION BELOW THIS LINE NOT SHARED WITH EXAMINEE

Task Number: RO-287	Validation Time: 8 minutes
NUREG 1123 Reference: 261000A4.07 3.1/3.2	Time Critical: NO
Location: SIMULATOR	Performance Method: PERFORM
Prepared/Revised by: S Hutchison	Revision Date: 7/28/02

STUDENT INFORMATION

Initial Conditions: A drywell atmosphere sample has been analyzed by chemistry within the last hour. Sample results permit containment venting, Both SGT trains are operable Reactor Building HVAC is operational. Drywell Noble Gas monitor indication is well below the High-High setpoint and has not changed in the past hour. Steps in PPM 2.3.1 have been complete through step 5.1.5.

INITIATING CUE

You have been directed by the CRS to start SGT “B” in preparation for venting the drywell in accordance with PPM 2.3.1, starting at step 5.1.6. Inform the CRS when you are ready to open CEP-V-1B.

Facility: Columbia Generating Station	Task No: RO-0327-N-TG
Task Title: Generator Capability Curve Interpretation – Faulted JPM – Respond to loss of H_2 in Main Generator	Job Performance Measure No: JPMB.1.Br1
K/A Reference: 245000K1.01 (3.1/3.3) 245000A4.05 (2.7/2.7)	
Examinee:	NRC Examiner:
Facility Evaluator:	Date:

Method of testing:

Simulator - Perform

JPM SETUP INFORMATION

Initial Conditions:	Reactor power is 97%. The Plant is operating normally.
Task Standard:	Respond to the loss of H_2 by reducing Main Generator output to less than the capability curve.
Required Materials:	PPM 4.820.B3 drop 2-3, PPM 2.5.7 att. 6.6
General References:	PPM 4.820.B3 drop 2-3, PPM 2.5.7 att. 6.6
Initiating Cue:	The CRS has directed you to increase reactor power to 100% with Recirculation Flow at the rate of 5 MWe/min. Notify the CRS when Reactor Power is 100%.
Time Critical Task:	NO
Validation Time:	10 minutes
Simulator ICs:	199
Malfunctions /Remote Triggers:	Loaded in IC-199
Overrides:	Loaded in IC-199
Special Setup Instructions:	Loaded in IC-199

PERFORMANCE INFORMATION

START TIME:

Critical Step: NO	
Performance Step: 1	Increase reactor power as directed to 100% power.
Standard:	Increase reactor power with recirc flow as directed
NOTE:	When the CRO approches the board, insert the H_2 leak causing H_2 pressure to decrease. Stop the leak when H_2 pressure is 70 psig on the computer panel. It takes about 2 ½ minutes for the leak to cause the **GEN H2 PRESS LOW** annunciator.
CUE:	**The candidate will have to be cued that this alarm is his.**
Comment: SAT / UNSAT	

Critical Step: YES	
Performance Step: 2	At H13-P620 (BD B) check hydrogen pressure on H_2-PI-1.
Standard:	CRO checks pressure as directed and verifies pressure less than 72 psig.
Comment: SAT / UNSAT	

Critical Step: NO	
Performance Step: 3	CRO dispatches an operator to check hydrogen pressure at the Generator H_2 Control Station on H2-PI-3.
Standard:	CRO either dispatches or asks CRS to dispatch operator as directed by the procedure.
CUE:	**If asked by the CRO to dispatch an operator, acknowledge the operator has been dispatched to chech the H_2 and seal oil systems.**
Comment: SAT / UNSAT	

NOTE: It is possible to reduce reactive load by the use of the voltage regulator to maintain the generator within the capability curve. If needed use the following cue to direct reduction of generator load by the reduction of Recirculation Flow.

CUE: Reduce generator output with Recirc Flow to maintain operation of the generator within the capability curve.

NOTE: STOP THE H2 LEAK AT 70# INDICATED ON THE COMPUTER PANEL.

Critical Step: YES	
Performance Step: 4	Maintain the Main Generator within the limits of the Generator Capability Curve in PPM 2.5.4 H_2 /CO_2 System.
Standard:	Reduce Main Generator load by recirculation flow to less than the value in the table in att. 6.6 of 2.5.7. 1162 MW for 68 psig hydrogen pressure.
CUE:	**If directed by the CRO to add H_2 to the generator, cue that there is no hydrogen available. A truck is on the way but will not be on site for at least 4 hours.**
Comment: SAT / UNSAT	

TERMINATION CUE: When generator load has been reduced to a value at least as low as the value in the table, announce to the CRO, "THE TERMINATION POINT OF THIS JPM HAS BEEN REACHED."

JPM TERMINATION TIME: JPM START TIME: - JPM COMPLETION TIME:	____________________

VERIFICATION OF COMPLETION

JPM Number:	JPMB.1.Br1
Examinee's Name:	
Examiner's Name:	
Date Performed:	
Facility Evaluator:	
Number of Attempts:	
Time to Complete:	

JPM INFORMATION CARD

HAND THE STUDENT INFORMATION CARD TO THE EXAMINEE

READ TO THE EXAMINEE:
I will explain the initial conditions, which steps to simulate or discuss, and provide initiation cues. When you complete the task successfully, the objective for this job performance measure will be satisfied.

Task Standard:	Respond to the loss of H_2 by reducing Main Generator output to less than the capability curve.
Required Materials:	PPM 4.820.B3 drop 2-3, PPM 2.5.7 att. 6.6
Safety Equipment:	NA
General References:	PPM 4.820.B3 drop 2-3, PPM 2.5.7 att. 6.6
Time Critical Task:	NO
Initial Conditions:	Reactor power is 97%. The Plant is operating normally.

INITIATING CUE

The CRS has directed you to increase reactor power to 100% with Recirculation Flow at the rate of 5 MWe/min. Notify the CRS when Reactor Power is 100%.

INFORMATION BELOW THIS LINE <u>NOT SHARED WITH EXAMINEE</u>

Task Number: RO-0327-N-TG	Validation Time: 10 minutes
NUREG 1123 Reference: 245000K1.01 (3.1/3.3) 245000A4.05 (2.7/2.7)	Time Critical: NO
Location: Simulator	Performance Method: Perform
Prepared/Revised by: S Hutchison	Revision Date: 7/23/02

STUDENT INFORMATION

Initial Conditions: Reactor power is 97%. The Plant is operating normally.

INITIATING CUE

The CRS has directed you to increase reactor power to 100% with Recirculation Flow at the rate of 5 MWe/min. Notify the CRS when Reactor Power is 100%.

Facility: Columbia Generating Station	Task No: RO-0048-A-RCC
Task Title: Change RCC Pumps – Alternate Path JPM – Respond to loss of RCC Pump	Job Performance Measure No: JPMB.1.Cr6
K/A Reference: 400000K1.02 (3.2/3.4)	
Examinee:	NRC Examiner:
Facility Evaluator:	Date:

Method of testing:

Simulator - Perform

JPM SETUP INFORMATION

Initial Conditions:	The reactor is shutdown following a scram. All equipment is normal. Maintenance needs to tag out RCC-P-1A for breaker maintenance.
Task Standard:	Respond the loss of an RCC pump and the subsequent closure of RCC-V-6 in accordance with PPM 4.8.3.2.
Required Materials:	NA
General References:	PPM 2.8.3, rev. 19 page 13, ABN-RCC rev 2, pages3-6, and PPM 4.820.B1 rev. 15, drop 4-1
Initiating Cue:	The CRS has directed you change RCC Pumps per PPM 2.8.3, section 5.4 Reactor and Radwaste Building Close Cooling Water System. After RCC-P-1A has been stopped, Place the control switch in Pull to Lock in preparation for hanging the tag for maintenance. Notify the CRS when RCC-P-1A is in PTL and RCC-P-1C is in operation with all system parameters normal.
Time Critical Task:	NO
Validation Time:	10 minutes
Simulator ICs:	198
Malfunctions/Remote Triggers:	Loaded in IC-198
Overrides:	Loaded in IC-198
Special Setup Instructions:	Loaded in IC-198

PERFORMANCE INFORMATION

START TIME:

Critical Step: NO	
Performance Step: 1	Ensure suction valve is open for RCC-P-1C.
CUE:	**Suction valve for RCC-P-1C is open.**
Standard:	Verifies suction valve is open for RCC-P-1C (RCC-V-1C).
Comment: SAT / UNSAT	

Critical Step: YES	
Performance Step: 2	Ensure discharge valve is open for RCC-P-1C.
CUE:	**Discharge valve for RCC-P-1C is open.**
Standard:	Verifies discharge valve is open for RCC-P-1C (RCC-V-2A).
Comment: SAT / UNSAT	

Critical Step: YES	
Performance Step: 3	Start RCC-P-1C.
Standard:	Place the control switch for RCC-P-1C in start and releases when the pump starts.
NOTE: May announce the start of RCC-P-1C.	
Comment: SAT / UNSAT	

Critical Step: YES	
Performance Step: 4	Stop RCC-P-1A and place the control switch in PTL.
CUE:	**IF CALLED AS OPS2, VERIFY THE DISCHARGE CHECK VALVE IS CLOSED.**
Standard:	Stops RCC-P-1A and places the control switch in PTL.
NOTE:	When the control switch for RCC-P-1A is placed in the PTL position, RCC-P-1C trips.
Comment: SAT / UNSAT	

Critical Step: NO	
Performance Step: 5	Refers to PPM 4.820.B1 drop 4-1, RCC PUMP C MOTOR OL TRIP. May attempt to restart RCC-P-1A. Refers to ABN-RCC Loss of RCC
CUE:	**If needed cue operator to respond to the Board N annunciator indication on Board S.**
Standard:	As stated above.
Comment: SAT / UNSAT	

Critical Step: NO	
Performance Step: 6	Verifies RCC-V-6 has closed.
Standard:	At BD N, verifies RCC-V-6 has closed.
Comment: SAT / UNSAT	

Critical Step: **YES***	
Performance Step: 7	1. Trip RWCU-P-1A (1B) 2. **Close RWCU-V-4*** 3. Throttle open RWCU-V-104
Standard:	1. Trips RWCU-P-1A (1B) 2. Closes RWCU-V-4 3. Throttles open RWCU-V-4
Comment: SAT / UNSAT	

TERMINATION CUE: THE TERMINATION POINT OF THIS JPM HAS BEEN REACHED.

JPM TERMINATION TIME:	
JPM START TIME:	
-	______________________________
JPM COMPLETION TIME:	

VERIFICATION OF COMPLETION

JPM Number:	JPMB.1.Cr6
Examinee's Name:	
Examiner's Name:	
Date Performed:	
Facility Evaluator:	
Number of Attempts:	
Time to Complete:	

JPM INFORMATION CARD

HAND THE STUDENT INFORMATION CARD TO THE EXAMINEE

READ TO THE EXAMINEE:
I will explain the initial conditions, which steps to simulate or discuss, and provide initiation cues. When you complete the task successfully, the objective for this job performance measure will be satisfied.

Task Standard:	Respond the loss of an RCC pump and the subsequent closure of RCC-V-6 in accordance with PPM 4.8.3.2.
Required Materials:	NA
Safety Equipment:	NA
General References:	PPM 2.8.3, rev. 19 page 13, ABN-RCC rev 2, pages3-6, and PPM 4.820.B1 rev. 15, drop 4-1
Time Critical Task:	NO
Initial Conditions:	The reactor is shutdown following a scram. All equipment is normal. Maintenance needs to tag out RCC-P-1A for breaker maintenance.

INITIATING CUE

The CRS has directed you change RCC Pumps per PPM 2.8.3, section 5.4 Reactor and Radwaste Building Close Cooling Water System. After RCC-P-1A has been stopped, Place the control switch in Pull to Lock in preparation for hanging the tag for maintenance.
Notify the CRS when RCC-P-1A is in PTL and RCC-P-1C is in operation with all system parameters normal.

INFORMATION BELOW THIS LINE NOT SHARED WITH EXAMINEE

Task Number: RO-0048-A-RCC	Validation Time: 10 minutes
NUREG 1123 Reference: : 400000K1.02 (3.2/3.4)	Time Critical: NO
Location: Simulator	Performance Method: Perform

STUDENT INFORMATION

Initial Conditions: The reactor is shutdown following a scram. All equipment is normal. Maintenance needs to tag out RCC-P-1A for breaker maintenance.

INITIATING CUE

The CRS has directed you change RCC Pumps per PPM 2.8.3, section 5.4 Reactor and Radwaste Building Close Cooling Water System. After RCC-P-1A has been stopped, Place the control switch in Pull to Lock in preparation for hanging the tag for maintenance.

Notify the CRS when RCC-P-1A is in PTL and RCC-P-1C is in operation with all system parameters normal.

Facility: Columbia Generating Station	Task No: RO-0269
Task Title: Start RCIC with the Arm and Depress Pushbutton – faulted – Recover from RCIC Electrical Overspeed Trip.	Job Performance Measure No: JPMB.1.Dr0
K/A Reference: 217000A4.02 3.9/3.9	
Examinee:	NRC Examiner:
Facility Evaluator:	Date:

Method of testing:

Simulator - Perform

JPM SETUP INFORMATION

Initial Conditions:	The plant was operating at 100% power when a transient occurred causing a loss of feedwater.
Task Standard:	Recovery of the RCIC System is completed per PPM 2.4.6 following RCIC Turbine Trip.
Required Materials:	N/A
General References:	PPM 2.4.6 rev.34, sections 5.2 and 5.10 PPM 4.601.A4 rev 20, drop 1-5
Initiating Cue:	You have been directed by the CRS to start RCIC for level control using the Arm and Depress Pushbutton. Notify the CRS when RCIC is injecting and level is stable in the band of +13 inches to +54 inches.
Time Critical Task:	NO
Validation Time:	7 min
Simulator ICs:	198
Malfunctions/Remote Triggers:	All loaded in IC-198
Overrides:	All loaded in IC-198
Special Setup Instructions:	All loaded in IC-198

PERFORMANCE INFORMATION

START TIME:

Critical Step: **YES**	
Performance Step: 1	5.2.2: ARM and DEPRESS RCIC-RMS-S36 (RCIC Manual Initiation) pushbutton.
CUE:	
Standard:	Arm and Depress pushbutton is rotated and depressed.
Note:	As the RCIC Turbine accelerates to speed, it will trip on an electrical Overspeed. The examinee will note the failure and respond as follows.
Comment: SAT / UNSAT	

Critical Step: NO	
Performance Step: 2	4.601.A4: 1. Confirm RCIC-V-1 is closed. 2. Ensure RCIC-V-46 is closed. 3. Refer to PPM 2.4.6 for recovery.
CUE:	
Standard:	
Comment: SAT / UNSAT	

Critical Step: NO	
Performance Step: 3	5.10.1, 2, and 3: For these steps cue the following if asked
CUE:	**5.10.1 RCIC is operable.** **5.10.2 RCIC will not be inop for GT 10 minutes.** **5.10.3 The cause of the trip has been corrected and the system is to be restored to service.**
Standard:	

Comment:
SAT / UNSAT

Critical Step: **YES**	
Performance Step: 4	5.10.4: N/A 5.10.5: Close or check close RCIC-V-45 isolation valve.
CUE:	
Standard:	RCIC-V-45 is closed or verified to be close.
Comment: SAT / UNSAT	

Critical Step: **YES**	
Performance Step: 5	5.10.6: Hold the control switch for RCIC-V-1 in the CLOSE position until both the valve stem and valve actuator indicate fully close.
CUE:	
Standard:	The control switch must be held in the CLOSE position until both the stem and the actuator indicate full closed.
Comment: SAT / UNSAT	

Critical Step: **YES**	
Performance Step: 6	5.10.7: Ensure both RCIC-V-63 and RCIC-V-8 (inboard and outboard isolation valves) are open.
CUE:	
Standard:	Both RCIC-V-63 and RCIC-V-8 must be verified open.
Comment: SAT / UNSAT	

Critical Step: NO	
Performance Step: 7 5.10.8: N/A – No initiation signal present.	
CUE:	
Standard:	
Comment: SAT / UNSAT	

Critical Step: **YES**	
Performance Step: 8 5.10.9: a. Open RCIC-V-54. b. Slowly jog open RCIC-V-1. c. Ensure the valve stem and actuator indicate open.	
CUE:	
Standard:	Valve manipulations performed per PPM 2.4.6
Comment: SAT / UNSAT	

ANNOUNCE TO THE EXAMINEE THAT THE TERMINATION POINT OF THE JPM HAS BEEN REACHED.

JPM TERMINATION TIME:	
JPM START TIME:	
-	______________________________
JPM COMPLETION TIME:	

VERIFICATION OF COMPLETION

JPM Number:	JPMB.1.Dr0
Examinee's Name:	
Examiner's Name:	
Date Performed:	
Facility Evaluator:	
Number of Attempts:	
Time to Complete:	

JPM INFORMATION CARD

HAND THE STUDENT INFORMATION CARD TO THE EXAMINEE

READ TO THE EXAMINEE:
I will explain the initial conditions, which steps to simulate or discuss, and provide initiation cues. When you complete the task successfully, the objective for this job performance measure will be satisfied.

Task Standard:	Recovery of the RCIC System is completed per PPM 2.4.6 following RCIC Turbine Trip.
Required Materials:	N/A
Safety Equipment:	N/A
General References:	PPM 2.4.6 rev.34, sections 5.2 and 5.10 PPM 4.601.A4 rev 20, drop 1-5
Time Critical Task:	NO
Initial Conditions:	The plant was operating at 100% power when a transient occurred causing a loss of feedwater.

INITIATING CUE

You have been directed by the CRS to start RCIC for level control using the Arm and Depress Pushbutton.
Notify the CRS when RCIC is injecting and level is stable in the band of +13 inches to +54 inches.

INFORMATION BELOW THIS LINE NOT SHARED WITH EXAMINEE

Task Number: RO-0269	Validation Time: 7 min
NUREG 1123 Reference: 217000A4.02 3.9/3.9	Time Critical: NO
Location: Simulator	Performance Method: Perform
Prepared/Revised by: S Hutchison	Revision Date: 7/30/02

STUDENT INFORMATION

Initial Conditions: The plant was operating at 100% power when a transient occurred causing a loss of feedwater.

INITIATING CUE

You have been directed by the CRS to start RCIC for level control using the Arm and Depress Pushbutton.
Notify the CRS when RCIC is injecting and level is stable in the band of +13 inches to +54 inches.

Facility: WNP-2	Task No: RO-0390-N-AC
Task Title: Transfer 480V Bus Power Supply From Normal to Alternate – Alternate Path.	Job Performance Measure No: JPMB.1.e
K/A Reference: 262001A4.04 3.6/3.7	
Examinee:	NRC Examiner:
Facility Evaluator:	Date:

Method of testing:

Simulator – Perform

JPM SETUP INFORMATION

Initial Conditions:	Bus SL-11 is currently powered from the normal power supply through circuit SM-1.
Task Standard:	Bus SL-11 is transferred from the normal power supply to the alternate power supply, in accordance with plant procedures.
Required Materials:	N/A
General References:	PPM 2.7.1B rev 14, section 5.0
Initiating Cue:	The CRS has directed you to transfer the SL-11 power source from the normal source, CB 11-1, to the alternate source, CB 21-11, per PPM 2.7.1B. Inform the CRS when the transfer of SL-11 to SL-21 is completed.
Time Critical Task:	NO
Validation Time:	5 minutes
Simulator ICs:	199
Malfunctions/ Remote Triggers:	Loaded in IC-199
Overrides:	Loaded in IC-199
Special Setup Instructions:	Loaded in IC-199

PERFORMANCE INFORMATION

START TIME:

Critical Step: **YES**	
Performance Step: 1	5.1.1: Ensure the CB-21/11 green tripped light is illuminated and the green position flag is being displayed in the CB-21/11 control switch window.
Standard:	Verifies green tripped light is illuminated and the green position flag is displayed in the window.
Comment: SAT / UNSAT	

Critical Step: **YES**	
Performance Step: 2	5.1.2: Place the BUS 11, 21, and 31 Trip Permissive selector switch in the TRIP CB-11/1 position.
Standard:	Trip switch place in the TRIP CB-11/1 position.
Comment: SAT / UNSAT	

Critical Step: **YES**	
Performance Step: 3	5.1.3: Place the CB-21/11control switch to the CLOSE position.
Standard:	CB-21/11 control switch in CLOSE.
Comment: SAT / UNSAT	

Critical Step: NO	
Performance Step: 4	5.1.4: Ensure the CB-21/11 green tripped light extinguishes and red closed light illuminates.
Standard:	Place the CS for CB-21/11 in the close position.
Comment: SAT / UNSAT	

Critical Step: **YES***	
Performance Step: 5	5.1.5/5.1.6: a) Ensure CB11/1 auto trips and the green tripped light illuminates at the time of breaker CB-21/11 closure. **b) *Manually trip CB-11/1** c) Verify CB-11/1 is tripped by the green tripped light and the green flag is displayed in the control switch window.
Standard:	Verifies indications and *trips CB-11/1.
NOTE: Candidate may announce the action to the CRS	
Comment: SAT / UNSAT	

Critical Step: NO	
Performance Step: 5	5.1.7: Place the BUS11, 21, and 31 Trip Permissive Selector switch in an off position.
Standard:	Place the switch in an off position.
Comment: SAT / UNSAT	

THE EXAMINEE SHOULD ANNOUNCE THE TERMINATION POINT OF THE JPM AT THIS POINT.

JPM TERMINATION TIME:	
JPM START TIME:	
-	______________________
JPM COMPLETION TIME:	

VERIFICATION OF COMPLETION

JPM Number:	JPMB.1.ER2
Examinee's Name:	
Examiner's Name:	
Date Performed:	
Facility Evaluator:	
Number of Attempts:	
Time to Complete:	

JPM INFORMATION CARD

HAND THE STUDENT INFORMATION CARD TO THE EXAMINEE

READ TO THE EXAMINEE:
I will explain the initial conditions, which steps to simulate or discuss, and provide initiation cues. When you complete the task successfully, the objective for this job performance measure will be satisfied.

Task Standard:	Bus SL-11 is transferred from the normal power supply to the alternate power supply, in accordance with plant procedures.
Required Materials:	N/A
Safety Equipment:	N/A
General References:	PPM 2.7.1B rev 12, section 5.1
Time Critical Task:	NO
Initial Conditions:	Bus SL-11 is currently powered from the normal power supply through circuit SM-1.

INITIATING CUE

The CRS has directed you to transfer the SL-11 power source from the normal source, CB 11-1, to the alternate source, CB 21-11, per PPM 2.7.1B.
Inform the CRS when the transfer of SL-11 to SL-21 is completed.

INFORMATION BELOW THIS LINE <u>NOT SHARED WITH EXAMINEE</u>

Task Number: RO-0390-N-AC	Validation Time: 5 minutes
NUREG 1123 Reference: 262001A4.04 3.6/3.7	Time Critical: NO
Location: Simulator	Performance Method: Perform
Prepared/Revised by: S Hutchison	Revision Date: 7/21/02

STUDENT INFORMATION

Initial Conditions: Bus SL-11 is currently powered from the normal power supply through circuit SM-1.

INITIATING CUE

The CRS has directed you to transfer the SL-11 power source from the normal source, CB 11-1, to the alternate source, CB 21-11, per PPM 2.7.1B.
Inform the CRS when the transfer of SL-11 to SL-21 is completed.

Facility: Columbia Generating Station	Task No: RO-659
Task Title: Shift Control Rod Drive Pumps	Job Performance Measure No: JPMB.1.F
K/A Reference: 201001A4.01 3.1/3.1	
Examinee:	NRC Examiner:
Facility Evaluator:	Date:

Method of testing:

Simulator - Perform

JPM SETUP INFORMATION

Initial Conditions:	The plant is at shutdown following a scram. All systems are normal for this condition.
Task Standard:	The operating CRD pump will be changed per PPM 2.1.1.
Required Materials:	N/A
General References:	PPM 2.1.1 rev. 30 Control Rod Drive System
Initiating Cue:	You have been directed to Change the operating CRD pump per PPM 2.1.1 section 5.6. Notify the CRS when the task is completed and the CRD System is stable.
Time Critical Task:	NO
Validation Time:	4 minutes
Simulator ICs:	199
Malfunctions/Remote Triggers:	Loaded in 199
Overrides:	Loaded in 199
Special Setup Instructions:	Loaded in 199

PERFORMANCE INFORMATION

START TIME:

Critical Step: NO	
Performance Step: 1	5.6.1: Request an Equipment Operator to check the CRD pumps locally for a start.
CUE:	**Report as OPS-2 that the pump is ready for a start.**
Standard:	
Comment: SAT / UNSAT	

Critical Step: NO	
Performance Step: 2	5.6.2: Place CRD-FC-600 in manual.
CUE:	
Standard:	CRD-FC-600 is placed in manual.
Note:	Step 5.6.3 is N/A.
Comment: SAT / UNSAT	

Critical Step: **YES**	
Performance Step: 3	5.6.4: Start CRD-P-1B
CUE:	
Standard:	CRD-P-1B is started.
Comment: SAT / UNSAT	

Critical Step: **YES**	
Performance Step: 4	5.6.5: Stop CRD-P-1A
CUE:	
Standard:	CRD-P-1A is stopped.
Comment: SAT / UNSAT	

Critical Step: **YES**	
Performance Step: 5	5.6.6: Null controller CRD-FC-600 and transfer to auto.
CUE:	
Standard:	Controller is nulled prior to returning to automatic.
Comment: SAT / UNSAT	

Critical Step: NO	
Performance Step: 6	5.6.7: Slowly adjust CRD-V-3 to establish a drive water differential pressure of 260 psid on CRD-DPI-602.
CUE:	
Standard:	
Comment: SAT / UNSAT	

THE EXAMINEE SHOULD ANNOUNCE THE TERMINATION POINT OF THE JPM AT THIS POINT.

JPM TERMINATION TIME:	
JPM START TIME:	
-	______________________________
JPM COMPLETION TIME:	

VERIFICATION OF COMPLETION

JPM Number:	JPMB.1.Fr0
Examinee's Name:	
Examiner's Name:	
Date Performed:	
Facility Evaluator:	
Number of Attempts:	
Time to Complete:	

JPM INFORMATION CARD

HAND THE STUDENT INFORMATION CARD TO THE EXAMINEE

READ TO THE EXAMINEE:
I will explain the initial conditions, which steps to simulate or discuss, and provide initiation cues. When you complete the task successfully, the objective for this job performance measure will be satisfied.

Task Standard:	The operating CRD pump will be changed per PPM 2.1.1.
Required Materials:	N/A
Safety Equipment:	N/A
General References:	PPM 2.1.1 rev. 30 Control Rod Drive System
Time Critical Task:	NO
Initial Conditions:	The plant is at shutdown following a scram. All systems are normal for this condition.

INITIATING CUE

You have been directed to Change the operating CRD pump per PPM 2.1.1 section 5.6. Notify the CRS when the task is completed and the CRD System is stable.

INFORMATION BELOW THIS LINE NOT SHARED WITH EXAMINEE

Task Number: RO-659	Validation Time: 4 minutes
NUREG 1123 Reference: 201001A4.01 3.1/3.1	Time Critical: NO
Location: Simulator	Performance Method: Perform
Prepared/Revised by: S Hutchison	Revision Date: 7/24/02

STUDENT INFORMATION

Initial Conditions: The plant is at shutdown following a scram. All systems are normal for this condition.

INITIATING CUE

You have been directed to Change the operating CRD pump per PPM 2.1.1 section 5.6.
Notify the CRS when the task is completed and the CRD System is stable.

Facility: Columbia Generating Station	Task No: RO-0298-N-PC
Task Title: Purge the Drywell	Job Performance Measure No: JPMB.1.G
K/A Reference: 223001A4.07 4.2/4.1	
Examinee:	NRC Examiner:
Facility Evaluator:	Date:

Method of testing:

Simulator - Perform

JPM SETUP INFORMATION

Initial Conditions: A plant shutdown is in progress in MODE 4 with the following conditions:

1. A wetwell purge has been completed.
2. "B" SGT is in operation for support of the purge.
3. A Fire Protection Impairment has been opened for the Primary Containment due to the loss of Nitrogen inerting.
4. Chemistry sampled Containment Atmosphere 2 hours ago and approve the purge.
5. Sections 5.1 and 5.2 of PPM 2.3.1 Primary Containment Venting, Purging, and Inerting are complete.

Task Standard: Simulate the lineup and purging of the Drywell per PPM 5.3.1 section 5.3.

Required Materials: N/A

General References: PPM 5.3.1 rev. 39, section 5.3

Initiating Cue: The CRS has directed you to purge the Drywell per PPM 5.3.1 starting at step 5.3.8. Inform the CRS when the drywell purge is in progress and the pressure control has been established.
NO CONTROL MANIPULATIONS ARE TO BE PERFORMED. THIS IS A SIMULATE ONLY JPM.

Time Critical Task: NO
Validation Time: 15 minutes
Simulator ICs: N/A
Malfunctions/Remote Triggers: N/A
Overrides: N/A
Special Setup Instructions: N/A

PERFORMANCE INFORMATION

START TIME:

NO CONTROL MANIPULATIONS ARE TO BE PERFORMED. THIS IS A SIMULATE ONLY JPM.

Critical Step: NO	
Performance Step: 1	5.3.9: At H13-P813 (Bd H) ensure ROA-FIC-A in MANUAL and adjust the controller output to zero.
CUE:	
Standard:	Simulate actions as directed by procedure.
Comment: SAT / UNSAT	

Critical Step: NO	
Performance Step: 2	5.3.10 and 11 1. Ensure SGT is operating per section 5.1. 2. The plant is in MODE 4 so the SGT train does not have to be logged as inoperable.
CUE:	**Cue as required for above.**
Standard:	
Comment: SAT / UNSAT	

Critical Step: **YES**	
Performance Step: 3	5.3.12: At H13-P813 (Bd H), open CEP-V-1A and CEP-V-2A.
CUE:	
Standard:	Correctly indicate CEP-V-1A and 2A OPEN. NOTE: Valve opening does not have to be logged in MODE 4.
Comment: SAT / UNSAT	

Critical Step: **YES**	
Performance Step: 4	5.3.13: Open SGT-V-1B
CUE:	
Standard:	Correctly indicates SGT-V-1B open.
Comment: SAT / UNSAT	

Critical Step: **YES**	
Performance Step: 5	5.3.14: At H13-P813 (Bd H) open CSP-V-1 and CSP-V-2.
CUE:	
Standard:	Correctly indicates CSP-V-1 and CSP-V-2 are open.
Comment: SAT / UNSAT	

Critical Step: **YES**	
Performance Step: 6	5.3.15: At H13-P813 (Bd H) open CSP-V-11.
CUE:	
Standard:	Correctly indicates CSP-V-11 is open.
Comment: SAT / UNSAT	

Critical Step: NO	
Performance Step: 7	5.3.16: Record the purge start time in OSP-INST-H101.
CUE:	**Start time has been recorded.**
Standard:	
Comment: SAT / UNSAT	

Critical Step: **YES**	
Performance Step: 8	5.3.17: At H13-P813 (Bd H) slowly increase ROA-FIC-1 controller output. Monitor drywell pressure on CMS-PR-1 (2) at H13-P601.
CUE:	
Standard:	Simulates controller output increase on ROA-FIC-1.
Comment: SAT / UNSAT	

Critical Step: **YES**	
Performance Step: 9	5.3.18: Close SGT-V-2B.
CUE:	
Standard:	Simulate SGT-V-2B closed.
Comment: SAT / UNSAT	

Critical Step: **YES**	
Performance Step: 10	5.3.19: Place SGT-DPIC-1B2 in MANUAL and adjust ROA purge supply and SGT flow rates, indicated on ROA-FIC-1 and SGT-DPIC-1B2 respectively as required to control drywell pressure LE 1.0 psig.
CUE:	**Following demonstration of controllers, CUE: Drywell pressure is 0.5 psig and stable.**
Standard:	Simulate adjusting controllers as directed.
Comment: SAT / UNSAT	

THE EXAMINEE SHOULD ANNOUNCE THE TERMINATION POINT OF THE JPM AT THIS POINT.

JPM TERMINATION TIME: JPM START TIME: -

JPM COMPLETION TIME:	

VERIFICATION OF COMPLETION

JPM Number:	JPMB.1.Gr1
Examinee's Name:	
Examiner's Name:	
Date Performed:	
Facility Evaluator:	
Number of Attempts:	
Time to Complete:	

JPM INFORMATION CARD

HAND THE STUDENT INFORMATION CARD TO THE EXAMINEE

READ TO THE EXAMINEE:
I will explain the initial conditions, which steps to simulate or discuss, and provide initiation cues. When you complete the task successfully, the objective for this job performance measure will be satisfied.

Task Standard: Simulate the lineup and purging of the Drywell per PPM 5.3.1 section 5.3.

Required Materials: N/A

Safety Equipment: N/A

General References: PPM 5.3.1 rev. 39, section 5.3.

Time Critical Task: NO

Initial Conditions: A plant shutdown is in progress in MODE 4 with the following conditions:

1. A wetwell purge has been completed.
2. “B” SGT is in operation for support of the purge.
3. A Fire Protection Impairment has been opened for the Primary Containment due to the loss of Nitrogen inerting.
4. Chemistry sampled Containment Atmosphere 2 hours ago and approve the purge.
5. Sections 5.1 and 5.2 of PPM 2.3.1 Primary Containment Venting, Purging, and Inerting are complete.

INITIATING CUE

The CRS has directed you to purge the Drywell per PPM 5.3.1 starting at step 5.3.8. Inform the CRS when the drywell purge is in progress and the pressure control has been established.
NO CONTROL MANIPULATIONS ARE TO BE PERFORMED. THIS IS A

SIMULATE ONLY JPM.

INFORMATION BELOW THIS LINE NOT SHARED WITH EXAMINEE

Task Number: RO-0298-N-PC
NUREG 1123 Reference: 223001A4.07 4.2/4.1
Location: Main Control Room
Prepared/Revised by: S Hutchison

Validation Time: 15 minutes
Time Critical: NO
Performance Method: Simulate
Revision Date: 7/21/02

STUDENT INFORMATION

Initial Conditions: A plant shutdown is in progress in MODE 4 with the following conditions:

1. A wetwell purge has been completed.
2. “B” SGT is in operation for support of the purge.
3. A Fire Protection Impairment has been opened for the Primary Containment due to the loss of Nitrogen inerting.
4. Chemistry sampled Containment Atmosphere 2 hours ago and approve the purge.
5. Sections 5.1 and 5.2 of PPM 2.3.1 Primary Containment Venting, Purging, and Inerting are complete.

INITIATING CUE

The CRS has directed you to purge the Drywell per PPM 5.3.1 starting at step 5.3.8. Inform the CRS when the drywell purge is in progress and the pressure control has been established.

NO CONTROL MANIPULATIONS ARE TO BE PERFORMED. THIS IS A SIMULATE ONLY JPM.

Facility: Columbia Generating Station	Task No: RO-0117 (SRO Upgrade Task) SRO-0251-A-RSP
Task Title: Control Room Evacuation – Preparations for ED	Job Performance Measure No: JPMB.2.Ar9
K/A Reference: 295016AA1.08 (4.04.0)	
Examinee:	NRC Examiner:
Facility Evaluator:	Date:

Method of testing:

Plant - Simulate

JPM SETUP INFORMATION

Initial Conditions:	The Control Room has been evacuated at the Shift Managers direction due to a fire. The immediate and subsequent operator actions for the control room have been successfully completed. Operators have been dispatched to perform all actions outside the Remote and Alternate Remote Shutdown Panels and have started on ATT. 7.2, 7.3, and 7.4 of ABN-CR-EVAC. SM-8 is powered from the Backup Transformer.
Task Standard:	Preparations for opening SRVs from the Remote Shutdown Panel are complete in accordance with ABN-CR-EVAC section 4.2.
Required Materials:	N/A
General References:	ABN-CR-EVAC rev. 2, section 4.2
Initiating Cue:	The CRS has directed you to initiate Section 4.2, of ABN-CR-EVAC to emergency depressurize the reactor. Notify the CRS when MS-RV-4C, 4B and 4A are energized and ready for operation. **THE PERFORMANCE OF THIS JPM WILL BE SIMULATED. NO CONTROL MANIPULATIONS WILL BE PERFORMED.**
Time Critical Task:	NO
Validation Time:	10 Minutes
Simulator ICs:	N/A
Malfunctions/Remote Triggers:	N/A
Overrides:	N/A
Special Setup Instructions:	N/A

PERFORMANCE INFORMATION

START TIME:

THE PERFORMANCE OF THIS JPM WILL BE SIMULATED. NO CONTROL MANIPULATIONS WILL BE PERFORMED.

Critical Step: **YES**	
Performance Step: 1	4.2.1: Place the following switches in EMERG: RHR-RPV INSTRUMENTATION POWER TRANSFER RCIC FLOW CONTROL RCIC-FIC-1R POWER TRANSFER
CUE:	
Standard:	Control manipulations simulated correctly.
Comment: SAT / UNSAT	

Critical Step: NO	
Performance Step: 2	4.2.2: Monitor the following: RCIC flow rate: RCIC-FI-1R/1 = 600 gpm RPV pressure: MS-PI-2 = 992 psig RPV water level: MS-LI-10 = -141 inches and down slow RHR-P-2B flow rate: RHR-FI-5 = min flow SW-P-1B discharge pressure: SW-PI-32BR = 198 psig
CUE:	**Cue as required.**
Standard:	
Comment: SAT / UNSAT	

Critical Step: **NO**	
Performance Step: 3	4.2.3, 4, 5, and 6:
CUE:	**These steps are in progress.**
Standard:	
Comment: SAT / UNSAT	

Critical Step: **YES**	
Performance Step: 4	4.2.7: At FRTP-1, place the following switches in the EMERG position: - FRTS-1 - FRTS-2 - FRTS-5 - FRTS-6 - E-RMS-FRTS7
CUE:	**Cue as required.**
Standard:	Control manipulations simulated correctly.
Comment: SAT / UNSAT	

Critical Step: NO	
Performance Step: 5	4.2.8: Establish communications between DG-2, and SM-8 Operators.
CUE:	**Communications are established.**
Standard:	
Comment: SAT / UNSAT	

Critical Step: NO	
Performance Step: 6	4.2.9: If evacuating the control room because of fire, enter and perform ABN-FIRE concurrently.
CUE:	**ABN-FIRE HAS BEEN ENTERED**
Standard:	
Comment: SAT / UNSAT	

Critical Step: NO	
Performance Step: 7	4.2.10: If not already performed, classify the event.
CUE:	**The event has been classified.**
Standard:	
Comment: SAT / UNSAT	

Critical Step: NO	
Performance Step: 8	4.2.11: N/A this step.
CUE:	**SM-8 is powered from the Backup Transformer.**
Standard:	
Comment: SAT / UNSAT	

Critical Step: NO	
Performance Step: 9	4.2.12: Place RHR-P-2B and SW-P-1B in service.
CUE:	**RHR-P-2B and SW-P-1B are in operation.**
Standard:	
Comment: SAT / UNSAT	

Critical Step: **YES**	
Performance Step: 10	4.2.13: Ensure the following relief valve control switches are in the CLOSE position at the Remote Shutdown Panel: MS-RV-4C MS-RV-4B MS-RV-4A
CUE:	**Cue as needed.**
Standard:	The relief valves are indicated correctly.
Comment: SAT / UNSAT	

Critical Step: **YES**	
Performance Step: 11	4.2.14: Place the MS-RV-4A, 4B, & 4C POWER TRANSFER switch to the EMERG position.
CUE:	**Switch is in EMERG.**
Standard:	Correct Power Transfer switch is indicated.
Comment: SAT / UNSAT	

THE EXAMINEE SHOULD ANNOUNCE THE TERMINATION POINT OF THE JPM AT THIS POINT.

JPM TERMINATION TIME: JPM START TIME: - JPM COMPLETION TIME:	______________________

VERIFICATION OF COMPLETION

JPM Number:	JPMB.2.Ar9
Examinee's Name:	
Examiner's Name:	
Date Performed:	
Facility Evaluator:	
Number of Attempts:	
Time to Complete:	

JPM INFORMATION CARD

HAND THE STUDENT INFORMATION CARD TO THE EXAMINEE

READ TO THE EXAMINEE:
I will explain the initial conditions, which steps to simulate or discuss, and provide initiation cues. When you complete the task successfully, the objective for this job performance measure will be satisfied.

Task Standard:	Preparations for opening SRVs from the Remote Shutdown Panel are complete in accordance with ABN-CR-EVAC section 4.2.
Required Materials:	N/A
Safety Equipment:	N/A
General References:	ABN-CR-EVAC rev. 2, section 4.2
Time Critical Task:	NO
Initial Conditions:	The Control Room has been evacuated at the Shift Managers direction due to a fire. The immediate and subsequent operator actions for the control room have been successfully completed. Operators have been dispatched to perform all actions outside the Remote and Alternate Remote Shutdown Panels and have started on ATT. 7.2, 7.3, and 7.4 of ABN-CR-EVAC. SM-8 is powered from the Backup Transformer.

INITIATING CUE

The CRS has directed you to initiate Section 4.2, of ABN-CR-EVAC to emergency depressurize the reactor. Notify the CRS when MS-RV-4C, 4B and 4A are energized and ready for operation.

THE PERFORMANCE OF THIS JPM WILL BE SIMULATED. NO CONTROL MANIPULATIONS WILL BE PERFORMED.

INFORMATION BELOW THIS LINE NOT SHARED WITH EXAMINEE

Task Number: RO-0117 (SRO Upgrade Task) SRO-0251-A-RSP	Validation Time: 10 min.
NUREG 1123 Reference: 295016AA1.08 (4.04.0)	Time Critical: NO
Location: Plant	Performance Method: Simulate7/30/02
Prepared/Revised by: S Hutchison	Revision Date:

STUDENT INFORMATION

Initial Conditions:

The Control Room has been evacuated at the Shift Managers direction due to a fire.

The immediate and subsequent operator actions for the control room have been successfully completed.

Operators have been dispatched to perform all actions outside the Remote and Alternate Remote Shutdown Panels and have started on ATT. 7.2, 7.3, and 7.4 of ABN-CR-EVAC.

SM-8 is powered from the Backup Transformer.

INITIATING CUE

The CRS has directed you to initiate Section 4.2, of ABN-CR-EVAC to emergency depressurize the reactor. Notify the CRS when MS-RV-4C, 4B and 4A are energized and ready for operation.

THE PERFORMANCE OF THIS JPM WILL BE SIMULATED. NO CONTROL MANIPULATIONS WILL BE PERFORMED.

Facility: Columbia Generating Station	Task No: RO-0247
Task Title: Close RPS EPA Breakers	Job Performance Measure No: JPMB.2.Br4
K/A Reference: 212000A.02 3.7/3.9	
Examinee:	NRC Examiner:
Facility Evaluator:	Date: 02/22/01

Method of testing:

Plant - Simulate

JPM SETUP INFORMATION

Initial Conditions:	RPS Division A has been de-energized due to operator error. PPM 2.7.6 section 5.1 has been completed.
Task Standard:	Actions taken to close RPS-EPA-3A are in accordance with PPM 2.7.6.
Required Materials:	N/A
General References:	PPM 2.7.6 rev. 16, section 5.3
Initiating Cue:	The CRS has directed you to close EPA breaker RPS-EPA-3A. Inform the CRS when RPS-EPA-3A is closed. **THE PERFORMANCE OF THIS JPM WILL BE SIMULATED. CONTROL MANIPULATIONS WILL NOT BE PERFORMED.**
Time Critical Task:	NO
Validation Time:	8 minutes
Simulator ICs:	N/A
Malfunctions/Remote Triggers:	N/A
Overrides:	N/A
Special Setup Instructions:	N/A

PERFORMANCE INFORMATION

START TIME:

Critical Step: NO	
Performance Step: 1	5.3.1: Ensure Section 5.1 has been completed.
CUE:	**This was given in the initial conditions and should not have to be cued.**
Standard:	
Comment: SAT / UNSAT	

Critical Step: **YES**	
Performance Step: 2	5.3.2: Obtain keys 166 and 168 from the control room.
CUE:	**When the candidate identifies the need to go to the control room for the keys, cue – "You have the necessary keys".**
Standard:	
Comment: SAT / UNSAT	

Critical Step: **YES***	
Performance Step: 3	5.3.3:In RPS-MG2 Room, close EPA breaker RPS-EPA-3A as follows: a. Ensure keylock switch S1 is in the NORMAL position. b. Ensure keylock switch S2 in the OPER position. c. Ensure the POWER IN indicator is illuminated. d. Ensure the following indicators are extinguished; 1. OVERVOLTAGE 2. UNDERVOLTAGE 3. UNDERFREQUENCY 4. POWER OUT
CUE:	**The UNDERVOLTAGE and UNDERFREQUENCY lights are illuminated.** e. Rotate keylock switch S2 to the RESET position and return to OPERATE.* f. Reset EPA breaker RPS-EPA-3A by opening it fully.* g. Close RPS-EPA-3A. * h. Ensure POWER OUT light is illuminated.
Standard:	Steps are completed in accordance with PPM 2.7.6.
Comment: SAT / UNSAT	

THE EXAMINEE SHOULD ANNOUNCE THE TERMINATION POINT OF THE JPM AT THIS POINT.

JPM TERMINATION TIME: JPM START TIME: - JPM COMPLETION TIME:	______________________

VERIFICATION OF COMPLETION

JPM Number:	JPMB.2.Br4
Examinee's Name:	
Examiner's Name:	
Date Performed:	
Facility Evaluator:	
Number of Attempts:	
Time to Complete:	

JPM INFORMATION CARD

HAND THE STUDENT INFORMATION CARD TO THE EXAMINEE

READ TO THE EXAMINEE:
I will explain the initial conditions, which steps to simulate or discuss, and provide initiation cues. When you complete the task successfully, the objective for this job performance measure will be satisfied.

Task Standard:	Actions taken to close RPS-EPA-3A are in accordance with PPM 2.7.6.
Required Materials:	N/A
Safety Equipment:	N/A
General References:	PPM 2.7.6 rev. 16, section 5.3
Time Critical Task:	NO
Initial Conditions:	RPS Division A has been de-energized due to operator error. PPM 2.7.6 section 5.1 has been completed.

INITIATING CUE

The CRS has directed you to close EPA breaker RPS-EPA-3A. Inform the CRS when RPS-EPA-3A is closed.

THE PERFORMANCE OF THIS JPM WILL BE SIMULATED. CONTROL MANIPULATIONS WILL NOT BE PERFORMED.

INFORMATION BELOW THIS LINE NOT SHARED WITH EXAMINEE

Task Number: RO-0247	Validation Time: 8 minutes
NUREG 1123 Reference: 212000A.02 3.7/3.9	Time Critical: NO
Location: Plant	Performance Method: Simulate
Prepared/Revised by: S Hutchison	Revision Date: 7/30/02

STUDENT INFORMATION

Initial Conditions: RPS Division A has been de-energized due to operator error. PPM 2.7.6 section 5.1 has been completed.

INITIATING CUE

The CRS has directed you to close EPA breaker RPS-EPA-3A. Inform the CRS when RPS-EPA-3A is closed.

THE PERFORMANCE OF THIS JPM WILL BE SIMULATED. CONTROL MANIPULATIONS WILL NOT BE PERFORMED.

Facility: Columbia Generating Station	Task No: RO-0706-N-DGHP
Task Title: Perform Manual Start of the HPCS DG from Local Panel	Job Performance Measure No: JPMB.2.Cr3
K/A Reference: 264000A4.04 3.7/3.7	
Examinee:	NRC Examiner:
Facility Evaluator:	Date:

Method of testing:

PLANT - Simulate

JPM SETUP INFORMATION

Initial Conditions:	A manual start of DG-3 is in progress at step 5.1.3 of SOP-DG3-START. The Unit Mode Select Switch is in the MAINT position. DG-3 has been entered in the Plant Logging System as inoperable. The plant is in a non-emergency condition, SM-2 is energized from TR-S. DG-3 governor droop switch is in DROOP. HPCS-P-2 HPCS Service Water is in service. Generator space heater breaker is open. All HPCS diesel generator annunciators are clear except for H13-P601.A1.6-8, HPCS System Out Of Service.
Task Standard:	The task will be completed when the candidate has simulated the start of DG-3 per SOP-SG3-START section 5.1.
Required Materials:	N/A
General References:	SOP-SG3-START rev. 0
Initiating Cue:	The CRS has directed you to continue the local, fast start of DG-3 at step 5.1.5, SOP-DG3-START. Notify the CRS when DG-3 is operating at 900 rpm. **CONTROL MANIPULATIONS WILL <u>NOT</u> BE PERFORMED. ALL ACTIONS AND STEPS WILL BE SIMULATED.**
Time Critical Task:	NO
Validation Time:	15 min.
Overrides:	N/A
Special Setup	N/A

Instructions:

PERFORMANCE INFORMATION

CONTROL MANIPULATIONS WILL NOT BE PERFORMED. ALL ACTIONS AND STEPS WILL BE SIMULATED.

START TIME:

Critical Step: NO	
Performance Step: 1	5.1.5: Place the Unit Mode Select Switch to the AUTO position. a. Ensure the red light illuminates (E-CP-DG/EP3). b. Enter in the Plant Tracking Log as Operable.
CUE:	**Red light on.** **DG-3 is logged as operable.**
Standard:	
Comment: SAT / UNSAT	

Critical Step: NO	
Performance Step: 2	5.1.6: Verify HPCS-P-2 is running.
CUE:	**AS REQUIRED**
Standard:	
Comment: SAT / UNSAT	

Critical Step: NO	
Performance Step: 3	5.1.7: Verify SW flow to the Diesel Engine GT 780 GPM (SW-FIS-9 local).
CUE:	**Flow GT 780 gpm.**
Standard:	Verifies flow on SW-FIS-9.
Comment: SAT / UNSAT	

Critical Step: NO	
Performance Step: 4	5.1.8: Verify the annunciator alarms are clear on E-CP-DG-RP3, except drop 1.1 and 3.1.
CUE:	**AS REQUIRED**
Standard:	Alarms indicated on E-CP-DG-RP3.
Comment: SAT / UNSAT	

Critical Step: NO	
Performance Step: 5	5.1.9: Verify HPCS Diesel Generator annunciators are clear (except drop 6.8 at H13-P601).
CUE:	**AS REQUIRED**
Standard:	Calls control room for verification.
Comment: SAT / UNSAT	

Critical Step: NO	
Performance Step: 6	5.1.10: At E-CP-DG/CP3, verifies the Generator Space Heater, breaker 1, is open
CUE:	**AS REQUIRED**
Standard:	Indicates breaker at E-CP-DG-CP3
Comment: SAT / UNSAT	

Critical Step: NO	
Performance Step: 6	5.1.11: N/A step 5.1.11
CUE:	**AS REQUIRED**
Standard:	
Comment: SAT / UNSAT	

Critical Step: YES	
Performance Step: 7	5.1.12: Place the Diesel Generator Mode Selector switch to the LOCAL position. (Located in the control room) Depress the UNIT START pushbutton at E-CP-DG/EP3.
CUE:	**DG Mode Selector Switch is in the Local position.** **NOTE: IF THE OPERATOR PRESSES THE *<u>ENGINE START</u>* PB (INSTEAD OF THE *<u>UNIT START</u>* PB), THE DIESEL FAILS TO START.**
Standard:	At E-CP-DG/EP3, depress the UNIT START pushbutton *
Comment: SAT / UNSAT	

Critical Step: **YES**	
Performance Step: 8	5.1.13: Verify the air start motors disengaged after engine start.
CUE:	**CUE AS REQUIRED**
Standard:	Indicate location of at least 1 air start motor.
Comment: SAT / UNSAT	

Critical Step: NO	
Performance Step: 9	5.1.14: Verify adequate oil in the starting a in-line lubricators within approximately 2 minutes of the diesel start.
CUE:	**CUE AS REQUIRED**
Standard:	Indicate the location of at least 1 in-line lubricator.
Comment: SAT / UNSAT	

Critical Step: NO	
Performance Step: 10	Classify and log the start in the Plant Logging System.
CUE:	**CUE AS REQUIRED**
Standard:	
Comment: SAT / UNSAT	

Critical Step: NO	
Performance Step: 11	5.1.16: N/A this step.
CUE:	**CUE AS REQUIRED**
Standard:	
Comment: SAT / UNSAT	

Critical Step: NO	
Performance Step: 12	5.1.17: Verify engine speed is approximately 900 RPM.
CUE:	**CUE AS REQUIRED**
Standard:	Indicate speed on HPCS-SI-DG3.
Comment: SAT / UNSAT	

THE EXAMINEE SHOULD ANNOUNCE THE TERMINATION POINT OF THE JPM AT THIS POINT.

JPM TERMINATION TIME: JPM START TIME: - JPM COMPLETION TIME:	____________________

VERIFICATION OF COMPLETION

JPM Number:	JPMB.2.Cr3
Examinee's Name:	
Examiner's Name:	
Date Performed:	
Facility Evaluator:	
Number of Attempts:	
Time to Complete:	

JPM INFORMATION CARD

HAND THE STUDENT INFORMATION CARD TO THE EXAMINEE

READ TO THE EXAMINEE:
I will explain the initial conditions, which steps to simulate or discuss, and provide initiation cues. When you complete the task successfully, the objective for this job performance measure will be satisfied.

Task Standard:	The task will be completed when the candidate has simulated the start of DG-3 per SOP-SG3-START section 5.1.
Required Materials:	N/A
Safety Equipment:	N/A
General References:	SOP-SG3-START rev. 0
Time Critical Task:	NO
Initial Conditions:	A manual start of DG-3 is in progress at step 5.1.3 of SOP-DG3-START. The Unit Mode Select Switch is in the MAINT position. DG-3 has been entered in the Plant Logging System as inoperable. The plant is in a non-emergency condition, SM-2 is energized from TR-S. DG-3 governor droop switch is in DROOP. HPCS-P-2 HPCS Service Water is in service. Generator space heater breaker is open. All HPCS diesel generator annunciators are clear except for H13-P601.A1.6-8, HPCS System Out Of Service.

INITIATING CUE

The CRS has directed you to continue the local, fast start of DG-3 at step 5.1.5, SOP-DG3-START. Notify the CRS when DG-3 is operating at 900 rpm.
CONTROL MANIPULATIONS WILL NOT BE PERFORMED. ALL ACTIONS

AND STEPS WILL BE SIMULATED.

INFORMATION BELOW THIS LINE <u>NOT SHARED WITH EXAMINEE</u>

Task Number: RO-0706-N-DGHP	Validation Time: 10 min.
NUREG 1123 Reference: 264000A4.04 3.7/3.7	Time Critical: NO
Location: PLANT	Performance Method: SIMULATE
Prepared/Revised by: S Hutchison	Revision Date: 7/29/02

STUDENT INFORMATION

Initial Conditions: A manual start of DG-3 is in progress at step 5.1.3 of SOP-DG3-START.
The Unit Mode Select Switch is in the MAINT position.
DG-3 has been entered in the Plant Logging System as inoperable.
The plant is in a non-emergency condition, SM-2 is energized from TR-S.
DG-3 governor droop switch is in DROOP.
HPCS-P-2 HPCS Service Water is in service.
Generator space heater breaker is open.
All HPCS diesel generator annunciators are clear except for H13-P601.A1.6-8, HPCS System Out Of Service.

INITIATING CUE

The CRS has directed you to continue the local, fast start of DG-3 at step 5.1.5, SOP-DG3-START. Notify the CRS when DG-3 is operating at 900 rpm.

CONTROL MANIPULATIONS WILL <u>NOT</u> BE PERFORMED. ALL ACTIONS AND STEPS WILL BE SIMULATED.

Facility: Columbia Generating Station	Task No: N/A
Task Title: Determination of Adequate Feedwater Temperature Prior to AIA	Job Performance Measure No: RA.1-1JPMr0
K/A Reference: 2.1.7 3.7/4.4	
Examinee:	NRC Examiner:
Facility Evaluator:	Date:

Method of testing:

Admin – Perform (Any location with reference material.)

JPM SETUP INFORMATION

Initial Conditions:

Task Standard:	Correctly determines action regarding entering AIA per PPM 3.1.2.
Required Materials:	N/A
General References:	PPM 3.1.2 rev. 57
Initiating Cue:	Using the above information, PPM 3.1.2 Reactor Plant Startup, step 5.8.16, and the Single Loop Power to flow map, determine if it is allowable to enter the Area of Increased Awareness. Justify your answer. Inform the CRS when you have completed this determination.
Time Critical Task:	N/A
Validation Time:	8 minutes
Simulator ICs:	N/A
Malfunctions/Remote Triggers:	N/A
Overrides:	N/A
Special Setup Instructions:	N/A

PERFORMANCE INFORMATION

START TIME:

Critical Step: **YES**	
Performance Step: 1	Determines reactor power from Single Loop Power to Flow Map.
CUE:	
Standard:	Using the 22500 gpm loop flow rate and 54% control rod line given in the initial conditions, determines that power is 30% (+ or – 2%).
NOTE:	
Comment: SAT / UNSAT	

Critical Step: **YES**	
Performance Step: 2	Determines feedwater temperature is satisfactory for entering AIA
CUE:	
Standard:	Using the 30% power from the above step and attachment 7.3 of PPM 3.1.2, determines that 296° F feedwater temperature is to the left of the line and is satisfactory.
NOTE:	
Comment: SAT / UNSAT	

THE EXAMINEE SHOULD INFORM THE CRS OF HIS COMPLETED DETERMINATION.

JPM TERMINATION TIME:	
JPM START TIME:	____________________

- JPM COMPLETION TIME:	___________________________

VERIFICATION OF COMPLETION

JPM Number:	RA.1-1JPMr0
Examinee's Name:	
Examiner's Name:	
Date Performed:	
Facility Evaluator:	
Number of Attempts:	
Time to Complete:	

JPM INFORMATION CARD

HAND THE STUDENT INFORMATION CARD TO THE EXAMINEE

READ TO THE EXAMINEE:
I will explain the initial conditions, which steps to simulate or discuss, and provide initiation cues. When you complete the task successfully, the objective for this job performance measure will be satisfied.

Task Standard:	Correctly determines action regarding entering AIA per PPM 3.1.2.
Required Materials:	N/A
Safety Equipment:	N/A
General References:	PPM 3.1.2 rev. 57
Time Critical Task:	NO
Initial Conditions:	A plant startup is in progress with RRC-P-1A tagged out for repair. The following conditions exist: Feedwater temperature = 296° F RRC Loop B flow = 22500 gpm Rod Line = 54% It is necessary to enter the Area of Increased Awareness

INITIATING CUE

Using the above information, PPM 3.1.2 Reactor Plant Startup, step 5.8.16, and the Single Loop Power to flow map, determine if it is allowable to enter the Area of Increased Awareness. Justify your answer.

Inform the CRS when you have completed this determination.

INFORMATION BELOW THIS LINE NOT SHARED WITH EXAMINEE

Task Number: N/A	Validation Time: 8 minutes
NUREG 1123 Reference: 2.1.7 3.7/4.	Time Critical: NO
Location: Any location with references.	Performance Method: Perform
Prepared/Revised by: S Hutchison	Revision Date: 6/26/02

STUDENT INFORMATION

Initial Conditions: A plant startup is in progress with RRC-P-1A tagged out for repair. The following conditions exist:

Feedwater temperature = 296° F

RRC Loop B flow = 22500 gpm

Rod Line = 54%

It is necessary to enter the Area of Increased Awareness

INITIATING CUE

Using the above information, PPM 3.1.2 Reactor Plant Startup, step 5.8.16, and the Single Loop Power to flow map, determine if it is allowable to enter the Area of Increased Awareness. Justify your answer.

Inform the CRS when you have completed this determination.

<table>
<tr><td>Question No.
RA.3-1:</td><td colspan="2">A co-worker has been injured and is unconscious in a High High Radiation area. You have decided to enter the area and carry the worker to safety.

What is the Columbia Generating Station administrative maximum allowable dose in a life saving situation?

CLOSED REFERENCE

ANSWER: 25 rem TEDE</td></tr>
<tr><td colspan="3">Response:</td></tr>
<tr><td colspan="3">SAT / UNSAT</td></tr>
<tr><td>2.3.1 (2.6/3.0)</td><td>6016</td><td>GEN-RPP-07 rev 3, page 8</td></tr>
</table>

Question No. RA.3-2

You have been directed to enter a radiation area with a general area dose rate of 245 mrem/hr.

What are the dosimetry requirements for entering this area?

CLOSED REFERENCE

ANSWER: Any ONE of the following is correct;

- A radiation monitoring device that continuously indicates the radiation dose rate in the area (e.g., survey instrument)

- A radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received (e.g., an alarming electronic dosimeter), with an appropriate alarm setpoint.

- A radiation monitoring device that continuously transmits dose rate and cumulative dose to a remote receiver monitored by Health Physics personnel responsible for controlling personnel radiation exposure within the area (e.g., an electronic dosimeter used in conjunction with a remote monitoring system)

- A self-reading dosimeter and,
  - Be under the surveillance (as specified in the RWP), while in the area of and individual at the work site who is qualified in radiation protection procedures and who is equipped with a radiation dose rate monitoring and indicating device and is responsible for controlling personnel radiation exposure within the area
  - Be under the surveillance (as specified in the RWP), by means of close circuit television, of an individual qualified in radiation protection procedures who is responsible for controlling personnel radiation exposure in the area.

Response:

SAT / UNSAT

2.3.1 (2.6/3.0)	6036	PPM 11.2.7.3 rev 18, pages 4 and 5

Question No. RA.3-1:	A co-worker has been injured and is unconscious in a High High Radiation area. You have decided to enter the area and carry the worker to safety. What is the Columbia Generating Station administrative maximum allowable dose in a life saving situation? **CLOSED REFERENCE**

Question No. RA.3-2	You have been directed to enter a radiation area with a general area dose rate of 245 mrem/hr. What are the dosimetry requirements for entering this area? **CLOSED REFERENCE**

Question No. RA.4-1:	At which Emergency Action Level are the Columbia Administrative Exposure Hold points automatically waived? **CLOSED REFERENCE** ANSWER: Alert	
Response:		
SAT / UNSAT		
2.4.29 (2.6/4.0)	6019	PPM 13.2.1 rev. 15, page 2

Question No. RA.4-2	The plant was operating at 100% power on Sunday morning when a transient occurred which required an emergency to be declared. The Shift manager is in the plant but cannot be contacted and his exact whereabouts are not known to the control room staff. Who is the Emergency Director? **CLOSED REFERENCE** ANSWER: The Control Room Supervisor
Response:	
SAT / UNSAT	

2.4.29 (2.6/4.0)	6132	PPM 13.1.1 rev. 31 page 7

Question No. RA.4-1:	At which Emergency Action Level are the Columbia Administrative Exposure Hold points automatically waived? **CLOSED REFERENCE**

Question No. RA.4-2	The plant was operating at 100% power on Sunday morning when a transient occurred which required an emergency to be declared. The Shift manager is in the plant but cannot be contacted and his exact whereabouts are not known to the control room staff. Who is the Emergency Director? **CLOSED REFERENCE**

Facility: Columbia Generating Station	Task No:
Task Title: DETERMINATION OF HEAVY LOAD OVER THE SPENT FUEL POOL	Job Performance Measure No: SA1.1-1JPMr0
K/A Reference: 2.1.20 4.3/4.2	
Examinee:	NRC Examiner:
Facility Evaluator:	Date:

Method of testing:

Admin – Perform (Any location with reference material.)

JPM SETUP INFORMATION

Initial Conditions:	The plant has been operating at 99% power since the last refueling outage. Preparations are in progress of the next biannual refuel outage, which is to start in 6 weeks. All plant systems are normal for 99% power. A load of 1490 pounds has to be lifted high enough to clear the handrails around the spent fuel pool and transported from the south side of the spent fuel pool to the north side of the pool at that height.
Task Standard:	The proposed lift will not be allowed for the given conditions per PPM 1.3.40 and LCS 1.9.2.
Required Materials:	N/A
General References:	PPM 1.3.40 and LCS 1.9.2
Initiating Cue:	Using PPM 1.3.40. determine if this lift is allowed and explain your answer.
Time Critical Task:	NO
Validation Time:	10 minutes
Simulator ICs:	N/A
Malfunctions/Remote Triggers:	N/A
Overrides:	N/A
Special Setup Instructions:	N/A

PERFORMANCE INFORMATION

START TIME:

Critical Step: NO	
Performance Step: 1	1.3.40 tep 5.2.45: Refer to Attachment 7.5 to determine: Irradiated fuel has been subcritical for at least 90 days. One train of control room air conditioning is in operation. Monitors are available to monitor an offsite release.
CUE:	
Standard:	Determine equipment requirements for the desired lift over the spent fuel pool satifactory.
NOTE:	Hand examinee LCS 1.9.2 when required.
Comment: SAT / UNSAT	

Critical Step: **YES**	
Performance Step:	Using LCS 1.9.2, determine the following: 1490 pounds can only be lifted and transported 3 inches over the spent fuel pool. The desired lift cannot be made because of the height limitation over the spent fuel pool.
CUE:	
Standard:	Dtermination is made that the lift cannot be made with the given conditions.
Comment: SAT / UNSAT	

THE EXAMINEE SHOULD HAND THE JPM TO THE EXAMINER AT THIS POINT.

JPM TERMINATION TIME:	
JPM START TIME:	
-	______________________
JPM COMPLETION TIME:	

VERIFICATION OF COMPLETION

JPM Number:	SA.1-1JPMr0
Examinee's Name:	
Examiner's Name:	
Date Performed:	
Facility Evaluator:	
Number of Attempts:	
Time to Complete:	

JPM INFORMATION CARD

HAND THE STUDENT INFORMATION CARD TO THE EXAMINEE

READ TO THE EXAMINEE:
I will explain the initial conditions, which steps to simulate or discuss, and provide initiation cues. When you complete the task successfully, the objective for this job performance measure will be satisfied.

Task Standard:	The proposed lift will not be allowed for the given conditions per PPM 1.3.40 and LCS 1.9.2.
Required Materials:	N/A
Safety Equipment:	N/A
General References:	PPM 1.3.40 and LCS 1.9.2
Time Critical Task:	NO
Initial Conditions:	The plant has been operating at 99% power since the last refueling outage. Preparations are in progress of the next biannual refuel outage, which is to start in 6 weeks. All plant systems are normal for 99% power. A load of 1490 pounds has to be lifted high enough to clear the handrails around the spent fuel pool and transported from the south side of the spent fuel pool to the north side of the pool at that height.

INITIATING CUE

Using PPM 1.3.40. determine if this lift is allowed and explain your answer.

INFORMATION BELOW THIS LINE <u>NOT SHARED WITH EXAMINEE</u>

Task Number:	Validation Time: 10 minutes
NUREG 1123 Reference: 2.1.20 4.3/4.2	Time Critical: NO
Location: Any with reference material	Performance Method: Perform
Prepared/Revised by: S Hutchison	Revision Date: 6/26/02

STUDENT INFORMATION

Initial Conditions: The plant has been operating at 99% power since the last refueling outage. Preparations are in progress of the next biannual refuel outage, which is to start in 6 weeks. All plant systems are normal for 99% power. A load of 1490 pounds has to be lifted high enough to clear the handrails around the spent fuel pool and transported from the south side of the spent fuel pool to the north side of the pool at that height.

INITIATING CUE

Using PPM 1.3.40. determine if this lift is allowed and explain your answer.

Facility: Columbia Generating Station	Task No: SRO-0026
Task Title: COMPLETE PLANNED SPECIAL EXPOSURE	Job Performance Measure No: SA.3JPM
K/A Reference: 2.3.4 2.5/3.1	
Examinee:	NRC Examiner:
Facility Evaluator:	Date:

Method of testing:

Admin – Perform (Any location with reference material.)

JPM SETUP INFORMATION

Initial Conditions: The plant is in a refueling outage with the drywell open. A valve manipulation is required to flush a hot spot from a trap in the drywell. The operator selected is expected to receive a total of 350 mrem to perform the task. An Increased Exposure Request is required for the task.

Task Standard: Increased Exposure Request form is completed in accordance with GEN-RPP-07.

Required Materials: N/A

General References: GEN-RPP-07 rev 3, page 13

Initiating Cue: The plant is in a refueling outage with the drywell open. A valve manipulation is required to flush a hot spot from a trap in the drywell. The operator selected is 30 years old and has been employed by Energy Northwest for 5 years. He is expected to receive a total of 350 mrem to perform the task. The following information applies to the operator.

1. John Q Operator
2. SS # - 555-55-5555
3. DOB – 01/01/71
4. 1.8 rem TEDE year to date.
5. 6.875 rem TEDE lifetime.
6. All dose is from Energy Northwest exposure.

As the operator's supervisor, complete the Increased Exposure Request per GEN-RPP-07.

Time Critical Task: N/A

Validation Time: 10 minutes

Simulator ICs: N/A

Malfunctions/Remote N/A

Triggers:

Overrides: N/A

Special Setup Instructions: N/A

PERFORMANCE INFORMATION

START TIME:

Critical Step: **YES**	
Performance Step: 1	Complete the first 3 sections of the Increased Exposure Request as the Supervisor of the designated operator.
CUE:	
Standard:	Increased Exposure Request form is completed in accordance with GEN-RPP-07. Grading Standard – compare candidates completed form with the attached form. Passing Criteria = each required section must match the attached form. The reason and justification must match the intent.
NOTE:	
Comment: SAT / UNSAT	

THE EXAMINEE SHOULD HAND THE CNF FORM TO THE EXAMINER AT THIS POINT.

JPM TERMINATION TIME:	
JPM START TIME:	
-	______________________________
JPM COMPLETION TIME:	

VERIFICATION OF COMPLETION

JPM Number:	SA.3JPMr0
Examinee's Name:	
Examiner's Name:	
Date Performed:	
Facility Evaluator:	
Number of Attempts:	
Time to Complete:	

JPM INFORMATION CARD

HAND THE STUDENT INFORMATION CARD TO THE EXAMINEE

READ TO THE EXAMINEE:
I will explain the initial conditions, which steps to simulate or discuss, and provide initiation cues. When you complete the task successfully, the objective for this job performance measure will be satisfied.

Task Standard:	Increased Exposure Request form is completed in accordance with GEN-RPP-07.
Required Materials:	N/A
Safety Equipment:	N/A
General References:	GEN-RPP-07 rev 3, page 13
Time Critical Task:	NO
Initial Conditions:	The plant is in a refueling outage with the drywell open. A valve manipulation is required to flush a hot spot from a trap in the drywell. The operator selected is expected to receive a total of 350 mrem to perform the task. An Increased Exposure Request is required for the task.

INITIATING CUE

The plant is in a refueling outage with the drywell open. A valve manipulation is required to flush a hot spot from a trap in the drywell. The operator selected is 30 years old and has been employed by Energy Northwest for 5 years. He is expected to receive a total of 350 mrem to perform the task. The following information applies to the operator.

7. John Q Operator
8. SS # - 555-55-5555
9. DOB – 01/01/71
10. 1.8 rem TEDE year to date.
11. 6.875 rem TEDE lifetime.
12. All dose is from Energy Northwest exposure.

As the operator's supervisor, complete the Increased Exposure Request per

GEN-RPP-07.

INFORMATION BELOW THIS LINE <u>NOT SHARED WITH EXAMINEE</u>

Task Number: SRO-0026	Validation Time:
NUREG 1123 Reference: 2.3.4 2.5/3.1	Time Critical:
Location: Any location with references.	Performance Method: Perform
Prepared/Revised by: S Hutchison	Revision Date: 6/27/02

STUDENT INFORMATION

Initial Conditions: The plant is in a refueling outage with the drywell open. A valve manipulation is required to flush a hot spot from a trap in the drywell. The operator selected is expected to receive a total of 350 mrem to perform the task. An Increased Exposure Request is required for the task.

INITIATING CUE

The plant is in a refueling outage with the drywell open. A valve manipulation is required to flush a hot spot from a trap in the drywell. The operator selected is 30 years old and has been employed by Energy Northwest for 5 years. He is expected to receive a total of 350 mrem to perform the task. The following information applies to the operator.

13. John Q Operator
14. SS # - 555-55-5555
15. DOB – 01/01/71
16. 1.8 rem TEDE year to date.
17. 6.875 rem TEDE lifetime.
18. All dose is from Energy Northwest exposure.

As the operator’s supervisor, complete the Increased Exposure Request per GEN-RPP-07.

Facility: Columbia Generating Station	Task No: SRO-0529
Task Title: CLASSIFY A SECURITY EVENT	Job Performance Measure No: SA.4JPMr0
K/A Reference: 2.4.41 2.3/4.1	
Examinee:	NRC Examiner:
Facility Evaluator:	Date:

Method of testing:

Admin – Perform (Any location with reference material.)

JPM SETUP INFORMATION

Initial Conditions:	The plant is operating at 100% power. One (1) hour ago, the FBI notified security and plant management that a mid-eastern terrorist group has threatened to interfere with the operation of Columbia Generating Station. The FBI considers this a credible threat. An Unusual Event was declared 50 minutes ago.
Task Standard:	The security event is classified correctly per PPM 13.1.1.
Required Materials:	N/A
General References:	PPM 13.1.1 rev. 31, page 27
Initiating Cue:	Five (5) minutes ago, security called and notified you that an explosive device has been discovered in Service Water Pump house A. All personnel have been evacuated from the immediate area. Meteorological data: Stability class E Wind direction 245 Wind speed 4 mph No precipitation Complete a CNF form for this condition
Time Critical Task:	**YES – 15 MINUTES**
Validation Time:	10 minutes
Simulator ICs:	N/A
Malfunctions/Remote Triggers:	N/A
Overrides:	N/A
Special Setup Instructions:	N/A

PERFORMANCE INFORMATION

START TIME:

Critical Step: NO	
Performance Step: 1	Complete the CNF Form for this condition.
CUE:	
Standard:	CNF Form is completed as attached. Grading standard: The Form must be completed as a reclassification and as a Site area Emergency. Description of the event (box 7) must meet the intent of the reference CNF Form.
NOTE:	
Comment: SAT / UNSAT	

THE EXAMINEE SHOULD HAND THE CNF FORM TO THE EXAMINER AT THIS POINT.

JPM TERMINATION TIME:	
JPM START TIME:	
-	______________________
JPM COMPLETION TIME:	

VERIFICATION OF COMPLETION

JPM Number:	SA.4JPMr0
Examinee's Name:	
Examiner's Name:	
Date Performed:	
Facility Evaluator:	
Number of Attempts:	
Time to Complete:	

JPM INFORMATION CARD

HAND THE STUDENT INFORMATION CARD TO THE EXAMINEE

READ TO THE EXAMINEE:
I will explain the initial conditions, which steps to simulate or discuss, and provide initiation cues. When you complete the task successfully, the objective for this job performance measure will be satisfied.

Task Standard:	The security event is classified correctly per PPM 13.1.1.
Required Materials:	N/A
Safety Equipment:	N/A
General References:	PPM 13.1.1 rev. 31, page 27
Time Critical Task:	NO
Initial Conditions:	The plant is operating at 100% power. One (1) hour ago, the FBI notified security and plant management that a mid-eastern terrorist group has threatened to interfere with the operation of Columbia Generating Station. The FBI considers this a credible threat. An Unusual Event was declared 50 minutes ago.

INITIATING CUE

Five (5) minutes ago, security called and notified you that an explosive device has been discovered in Service Water Pump house A. All personnel have been evacuated from the immediate area.

Meteorological data:
Stability class E
Wind direction 245
Wind speed 4 mph
No precipitation

Complete a CNF form for this condition

INFORMATION BELOW THIS LINE NOT SHARED WITH EXAMINEE

Task Number: SRO-0529	Validation Time: 10 minutes
NUREG 1123 Reference: 2.4.41 2.3/4.1	Time Critical: **YES 15 MINUTES**
Location: Any with reference material	Performance Method: Perform
Prepared/Revised by: S Hutchison	Revision Date: 6/26/02

STUDENT INFORMATION

Initial Conditions: The plant is operating at 100% power. One (1) hour ago, the FBI notified security and plant management that a mid-eastern terrorist group has threatened to interfere with the operation of Columbia Generating Station. The FBI considers this a credible threat. An Unusual Event was declared 50 minutes ago.

INITIATING CUE

Five (5) minutes ago, security called and notified you that an explosive device has been discovered in Service Water Pump house A. All personnel have been evacuated from the immediate area.

Meteorological data:
Stability class E
Wind direction 245
Wind speed 4 mph
No precipitation

Complete a CNF form for this condition