Task No.: 200226A0501

===

Task Title: AVERAGE DRYWELL TEMPERATURE CALCULATION

===

Trainee: ___________________________ Examiner: ______________________________

Pass:______ Fail:______ Examiner Signature: ____________________ Date: _________

Additional Program Information:

1. Appropriate Performance Locations: **SIMULATOR ONLY!!**
2. Appropriate Trainee level: RO/SRO
3. Evaluation Method: __ **Perform in simulator only**
4. Performance Time: 12 minutes
5. NRC K/A 216000 K5.07 (3.6/3.8), A1.01 (3.2/3.3)

Directions to Examiner:

NOTE: The standards for the values collected from the panels will be based on the evaluator's reading of the same parameter. The average value calculated standard will be based on the values collected.

1. This JPM evaluates the trainee's ability to perform an average drywell temperature calculation.
2. Only the cues preceded by "#" should be given.
3. All blanks must be filled out with either initials or an "NP" for "not performed"; an explanation may also be written in the space if desired by the examiner.
4. Give the trainee his copy of the Directions to the Trainee (Attachment 2) when ready to start the JPM.
5. Brief the trainee, place the simulator in run, and tell the trainee to begin.

Task No.: 200226A0501

==

Task Title: AVERAGE DRYWELL TEMPERATURE CALCULATION

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Directions to Trainee:

When I tell you to begin, you are to perform an average drywell temperature calculation. Before you start, I will state the general plant conditions, the Initiating Cues and answer any questions you may have.

During task performance, state the actions you are taking, e.g.: repositioning controls and observing instrumentation.

==

General Conditions:

1. PMIS is unavailable for accurate drywell temperature calculations.

General References:

1. Procedure 5.8.10, Average Drywell Temperature Calculation

General Tools and Equipment:

1. None

Special Conditions, References, Tools, Equipment:

1. Simulator Setup: See Attachment 1.
2. Critical checks denoted by "*".
3. Simulator cues denoted by "#".

Task Standards:

1. 100% of critical elements successfully completed without error.
2. 100% of safety and radiological work practices.

Initiating Cue(s):

The CRS has directed you to perform an average drywell temperature calculation per procedure 5.8.10. Inform the CRS of the average drywell temperature when complete.

Task No.: 200226A0501

Task Title: AVERAGE DRYWELL TEMPERATURE CALCULATION

Performance Checklist	Standards	Initials
1. Refer to Procedure 5.8.10.	Refers to Procedure 5.8.10, Section 3.0, and Attachment 1.	_________
2. Record time.	Record time on Attachment 1 (± 1 minute of simulator time on panel 9-5).	_________
3. Record PC-TI-508A, RX HEAD FLANGE TEMP (0˚).	Record PC-TI-508A, RX HEAD FLANGE TEMP (0˚) on Attachment 1 (± 5˚F of indicated value).	_________*
4. Record PC-TI-505E, ZONE 2B TEMP (288˚).	Record PC-TI-505E, ZONE 2B TEMP (288˚) on Attachment 1 (± 5˚F of indicated value).	_________*
5. Record PC-TI-510A, ZONE 2C TEMP (0˚).	Record PC-TI-510A, ZONE 2C TEMP (0˚) on Attachment 1 (± 5˚F of indicated value).	_________*

NOTE: Significant differences exist between the temperature indicated on the recorder scale and the temperature recorded on the recorder paper.

Performance Checklist	Standards	Initials
6. Record PC-TR-502, RECIRC PUMPS AREA TEMP (RED: PUMP A).	Record PC-TR-502, RECIRC PUMPS AREA TEMP (RED: PUMP A) on Attachment 1 (± 5˚F of indicated value). **#CUE:** If the candidate asks whether to use the temperature indicated on the scale or on the recorder paper instruct the candidate to use the value recorded on the paper.	_________*

Task No.: 200226A0501

Task Title: AVERAGE DRYWELL TEMPERATURE CALCULATION

Performance Checklist	Standards	Initials
7. Record PC-TR-502, RECIRC PUMPS AREA TEMP (BLUE: PUMP B).	Record PC-TR-502, RECIRC PUMPS AREA TEMP (BLUE: PUMP B) on Attachment 1 (± 5˚F of indicated value). **#CUE:** If the candidate asks whether to use the temperature indicated on the scale or on the recorder paper instruct the candidate to use the value recorded on the paper.	_________*
8. Multiply by weight factor.	Multiply PC-TI-508A reading x 0.1 and record on Attachment 1 (Actual number obtained by calculation).	_________*
9. Multiply by weight factor.	Multiply PC-TI-505E reading x 0.2 and record on Attachment 1 (Actual number obtained by calculation).	_________*
10. Multiply by weight factor.	Multiply PC-TI-510A reading x 0.1 and record on Attachment 1 (Actual number obtained by calculation).	_________*
11. Multiply by weight factor.	Multiply PC-TR-502: RED reading x 0.3 and record on Attachment 1 (Actual number obtained by calculation).	_________*
12. Multiply by weight factor.	Multiply PC-TR-502: BLUE reading x 0.3 and record on Attachment 1 (Actual number obtained by calculation).	_________*
13. Total the products.	Adds the products from the 5 calculations to determine AVG DW TEMP (Actual sum of the 5 values).	_________*

Task No.: 200226A0501

Task Title: AVERAGE DRYWELL TEMPERATURE CALCULATION

Performance Checklist	Standards	Initials
14. Informs the CRS.	Informs the CRS of the AVG DW TEMP. #CUE: As the CRS, acknowledge the report.	_________

Task No.: 200226A0501

==

Task Title: AVERAGE DRYWELL TEMPERATURE CALCULATION

==

ATTACHMENT 1

SIMULATOR SET-UP

A. Materials Required

None

B. Initialize the Simulator in any power IC.

Batch File Name - None

C. Change the simulator conditions as follows:

1. Triggers

None

2. Malfunctions (suggested. Any may be used that results in high drywell pressure)

Number	Title	Trigger	TD	Severity	Ramp	Initial
RR20A	Reactor Coolant Leakage Inside Containment.	A	0	10	0	N/A

3. Remotes

None

4. Overrides

Instrument	Tag	Trigger	TD	Value	Ramp
None					

Task No.: 200226A0501

==

Task Title: AVERAGE DRYWELL TEMPERATURE CALCULATION

==

5. Panel Setup

   a. Insert both listed malfunctions.

   b. Allow drywell conditions to stabilize.

   c. Ensure all SPDS screens are blank.

   d. Place the Simulator in FREEZE.

Note: If this JPM is to be performed more than once, snap the simulator into an IC after the panel setup is complete.

ATTACHMENT 2

Directions to Trainee:

When I tell you to begin, you are to perform an average drywell temperature calculation per procedure 5.8.10. Before you start, I will state the general plant conditions, the Initiating Cues and answer any questions you may have.

During task performance, state the actions you are taking, e.g.: repositioning controls and observing instrumentation.

General Conditions:

1. PMIS is unavailable for accurate drywell temperature calculations.

Initiating Cues:

The CRS has directed you to perform an average drywell temperature calculation per procedure 5.8.10. Inform the CRS of the average drywell temperature when complete.

Cooper Nuclear Station Category "A" - Examination Outline Cross Reference	
Operating Test Number	Cat "A" Test: 1
Examination Level	RO
Administrative Topic	A.1
Subject Description:	Control Rod Coupling Check
Question Number:	1

THIS IS A OPEN REFERENCE QUESTION!

Question:

When is a coupling check of control rods required?

Answer:

Control rod coupling shall be verified each time a control rod is withdrawn to the full out position (also accept each time the control rod is withdrawn to position 48) (**Minimum required for passing**)

AND

Prior to declaring control rod OPERABLE after work on control rod or CRD System that could affect coupling. **(Required for full credit**)

Technical Reference(s):
T.S. SR 3.1.3.5

K/A #:	**Importance:**
2.2.1	3.7/3.6

Comments:

THIS IS A OPEN REFERENCE QUESTION!

Question:

When is a coupling check of control rods required?

Cooper Nuclear Station **Category “A” - Examination Outline Cross Reference**	
Operating Test Number	Cat “A” Test: 1
Examination Level	RO
Administrative Topic	A.1
Subject Description:	Second Check of Control Rod Movement
Question Number:	2

THIS IS A OPEN REFERENCE QUESTION!

Question:

What are the requirements for the second checking of control rod movement during a start up?

Answer:

All control rod moves performed when REACTOR MODE switch is in START & HOT STBY or RUN shall be checked by a second Licensed Operator or an individual certified as an STE.

Technical Reference(s):

Procedure 2.0.3, Conduct of Operations
Procedure 2.1.1, Startup Procedure
Procedure 10.13, Control Rod Sequence And Movement Control

K/A #:	Importance:
2.1.2	3.0/4.0

Comments:

Student Handout

THIS IS A OPEN REFERENCE QUESTION!

Question:

What requirements must be met for the second checking of control rod movement during a start up?

Task No.: 202012O0201

==

Task Title: PERFORM JET PUMP OPERABILITY CHECK

==

Trainee: ____________________________ Examiner: ___________________________

Pass:____ Fail:___ Examiner Signature: ______________________ Date: ________

Additional Program Information:

1. Appropriate Performance Locations: **Simulator Only**
2. Appropriate Trainee Level: RO / SRO
3. Evaluation Method: _____**Perform in Simulator Only.**
4. Performance Time: 18 minutes
5. NRC K/A 202001 K1.06 3.6/3.6

Directions to Examiner:

1. This JPM evaluates the trainee's ability to perform the daily Jet Pump and Recirc Pump Flow Check of the Daily Tech Specs Surveillance Log.
2. Only the cues preceded by "#" should be given.
3. All blanks must be filled out with either initials or an "NP" for "not performed"; an explanation may also be written in the space, if desired, by the examiner.
4. Give the trainee his copy of the Directions to the Trainee (Attachment 2) when ready to start the JPM.
5. Brief the trainee, place the Simulator in RUN, and tell the trainee to begin.

Directions to Trainee:

When I tell you to begin, you are to perform the daily Jet Pump **AND** Recirc Pump Flow Check. Before you start, I will state the general plant conditions, the initiating cues and answer any questions you may have.

During task performance, state the actions you are taking, e.g.: repositioning controls and observing instrumentation.

===

General Conditions:

1. The plant is operating at rated power with DEH in Mode 4.

Task No.: 202012O0201

==

Task Title: PERFORM JET PUMP OPERABILITY CHECK

==

2. Both Reactor Recirculation pumps are operating with pump flows balanced.

General References:

1. Procedure 6.LOG.601

General Tools and Equipment:

1. Calculator.
2. Jet pump operability curves.

Special Conditions, References, Tools, Equipment:

1. Simulator Setup: See Attachment 1.
2. Critical checks denoted by "*".
3. Simulator cues denoted by "#".

Task Standards:

1. 100% of critical elements successfully completed without error.
2. 100% of safety and radiological work practices.

Initiating Cue(s):

The Control Room Supervisor directs you to perform the daily Jet Pump (6.LOG.601 Attachment 12) and Recirc Pump Flow Check (6.LOG.601 Attachment 13) as part of the routine shift activities. Notify the CRS when the task is complete.

Task No.: 202012O0201

Task Title: PERFORM JET PUMP OPERABILITY CHECK

Performance Checklist	Standards	Initials
NOTE: Steps 1 - 5 may be in any order.		
1. Record indicated core flow (10^6 lb/hr)	Record Core flow from Recorder NBI-DPR/FR-95 (± 2). **CUE**: Core Flow = 67	________*
2. Record RR pump flow (10^3 gpm)	Record RR pump flow from RR-FR-163 for Pumps A & B (± 2). **CUE**: Pump A = 44.8; Pump B =45.8	________*
3. Record RRMG Set speed	Record RRMG Set speed from the following (± 2): a. RRFC-SIC-16A for RRMG A b. RRFC-SIC-16B for RRMG B **CUE**: RRMG A = 93; RRMG B = 93.	________*
4. Record Jet Pump Flow	Record Jet Pump Flow from the following (± 2): a. NBI-FI-92A for LOOP A b. NBI-FI-92B for LOOP B **CUE**: LOOP A = 33.5; LOOP B = 33.5	________*

Task No.: 202012O0201

==

Task Title: PERFORM JET PUMP OPERABILITY CHECK

==

Performance Checklist	Standards	Initials
5. Record Jet Pump Differential Pressure	Record differential pressures from individual jet pump instruments NBI-FI-78A through NBI-FI-78Z on Panel 9-38 in control room (± 2). **CUE**: 1 = 38 8 = 3815 = 40 2 = 39 9 = 3816 = 41 3 = 39 10 = 4117 = 36 4 = 36 11 = 3718 = 35 5 = 43 12 = 3619 = 38 6 = 41 13 = 3620 = 37 7 = 38 14 = 36	________*
6. Record B and A Average	Add JP #1 through 10 and divide by 10 for LOOP B, then add JP #11 through 20 and divide by 10 for LOOP A (± 2). **CUE**: Average 39.1 for LOOP B and 37.2 for LOOP A.	________*

NOTE: **Curves are contained in the binder labeled "Cooper Nuclear Station Jet Pump Operability Graphs and Instability Noise Level Data".**

7. Verify RR pump flow and RRMG set speed within limits.	Determine that the values recorded in Items B and C are within the limits of the curve for **Check 1 (SAT checked).**	________*

NOTE: **The values obtained in the simulator for JP flow and RRMG set speed are very close to the limits of the Jet Pump Operability Graphs. The candidate may determine that it falls outside the limit.**

Task No.: 202012O0201

Task Title: PERFORM JET PUMP OPERABILITY CHECK

Performance Checklist	Standards	Initials
8. Verify JP flow and RRMG set speed within limits.	Determine that the values recorded in Items C and D are within or outside the limits of the curve for **Check 2.**	________*
9. Jet Pump △p differs by ≤20% from established patterns.	Determine that Jet Pump △p differs by ≤20% from established patterns for **Check 3 (SAT checked).**	________*
10. Verify check 1 and 2 SAT or check 3 SAT.	Verify check 1 and 2 SAT or check 3 are SAT.	________*
11. Verify Item A value is not in Stability Exclusion Region of Power to Flow Map (2.1.10 Power to Flow Map).	Determines operation is outside Stability Exclusion Region of Power to Flow Map	________*
12. Verify Item D values for Loop A and Loop B flow mismatch is ≤ 7.35×10^6 lbs/hr at < 51.45×10^6 lbs/hr Rated Core Flow.	Determine Step is N/A.	________
13. Verify Item D values for Loop A and Loop B flow mismatch is ≤ 3.67×10^6 lbs/hr at ≥ 51.45×10^6 lbs/hr Rated Core Flow.	Determine Item D values for Loop A and Loop B flow mismatch is ≤ 3.67×10^6 lbs/hr (SAT entered into block).	________*

Task No.: 202012O0201

===

Task Title: PERFORM JET PUMP OPERABILITY CHECK

===

Performance Checklist	Standards	Initials
14. Verify Recirc Pump operating or RHR Pump operating in SDC.	Determine Step is N/A.	________
15. Inform the CRS that the task is complete.	Inform Control Room Supervisor that the daily Jet Pump and Recirc Pump Flow Check is Complete. #**CUE**:The CRS Acknowledges the report.	________

Task No.: 202012O0201

==

Task Title: PERFORM JET PUMP OPERABILITY CHECK

==

ATTACHMENT 1

SIMULATOR SET-UP

A. Materials Required

None

B. Initialize the Simulator in IC-18.

Batch File Name - none.

C. Change the simulator conditions as follows:

1. Triggers

None

2. Malfunctions

None

3. Remotes

None

4. Overrides

Instrument	Tag	Trigger	TD	Value	Ramp
None					

5. Panel Setup

None

D. Place the Simulator in RUN to allow conditions to stabilize.

ATTACHMENT 2

Directions to Trainee:

When I tell you to begin, you are to perform the daily Jet Pump and Recirc Pump Flow Check. Before you start, I will state the general plant conditions, the initiating cues and answer any questions you may have.

During task performance, state the actions you are taking, e.g.: repositioning controls and observing instrumentation.

General Conditions:

1. The plant is operating at rated power with DEH in Mode 4.
2. Both Reactor Recirculation pumps are operating with pump flows balanced.

Initiating Cues:

The Control Room Supervisor directs you to perform the daily Jet Pump (6.LOG.601 Attachment 12) and Recirc Pump Flow Check (6.LOG.601 Attachment 13) as part of the routine shift activities. Notify the CRS when the task is complete.

Task No.:299007O0304

==

Task Title: Determine the Radiological Protection Requirements

==

Trainee: ______________________________ Examiner: _____________________________

Score: _____ Pass _____ Fail Examiner Signature: __________________ Date: ________

Additional Program Information:

1. Appropriate Performance Locations: Any
2. Appropriate Trainee Levels: RO / SRO / STE
3. Evaluation Method: Simulate
4. Performance Time: 10 minutes
5. NRC K/A 2.3.10 (2.9/3.3)

Directions to Examiner:

1. This JPM evaluates the trainee's ability to determine the radiation protection requirements required to perform CRD system venting.
2. All blanks must be filled out with either initials or an "NP" for "not performed, " and an explanation may also be written in the space if desired by the examiner.
3. Give the trainee his copy of the Directions to the Trainee (Attachment 1) when ready to start the JPM.
4. Brief the trainee and tell the trainee to begin.

Directions to Trainee:

When I tell you to begin, you are to determine the radiation protection requirements required to perform CRD Drive System Venting above the HCUs in the Reactor Building 903 South. This includes removal of any tools from the area following task completion.

During task performance, state the actions you are taking, e.g.: repositioning controls and observing instrumentation.

General Conditions:

1. The plant is shutdown for a refueling outage.
2. The CRD system has been placed in service.
3. CRD Drive System Venting per 2.2.8 is scheduled for this shift.
4. The Area over the CRD HCUs is a radiation area and a contaminated area.

Task No.:299007O00304

==

Task Title: Determine the Radiological Protection Requirements

==

General References:

1. Procedure 9.ALARA.4, Radiation Work Permits

General Tools and Equipment:

None

Special Conditions, References, Tools, Equipment:

1. Critical checks denoted by "*."

Task Standards:

1. 100% of critical elements successfully completed without error.
2. 100% of safety and radiological work practices.
3. All RWP requirements checked.

Initiating Cue(s):

Evaluate RWP 20031010 and checkmark the **minimum** radiological protection requirements needed in order to perform CRD drive system venting.

Task No.:299007O0304

Task Title: Determine the Radiological Protection Requirements

Performance Checklist	Standards	Initials
NOTE: **Some candidates may not need to perform the first two elements of this JPM due to familiarity with this particular CRD system evolution.**		
1. Obtains procedure 2.2.8.	Obtains the current revision of Procedure 2.2.8.	______
2. Evaluates procedure 2.2.8 to determine what actions are required by procedure.	Determines that entry above the CRD HCUs and venting of potentially contaminated fluid is required by the procedure.	______
3. Compares the requirements of RWP20031010 to actions required by the CRD drive system venting.	Compares the actions required by Procedure 2.2.8 to those specified in the RWP.	_______
4. Highlights the requirements on the RWP.	Highlights the minimum requirements on the RWP.	_______*
NOTE: The following step is completed by the JPM evaluator ONLY.		
5. Compare the highlighted items on the students copy to those on Attachment one.	Checkmarked portions of the RWP received from the candidate match the checkmarked portions on Attachment 1. (Highlighted portion indicates item the candidate may check but is not required to pass)	_______*

Task No.:299007O0304

===

Task Title: Determine the Radiological Protection Requirements

===

ATTACHMENT 1

RWP CATEGORY/TYPE	ROUTINE/PSE	ALARMS (Mr)	CONDITION REQUIRING SWP
SWP / Routine	Routine	**RATE:** 1200 **DOSE:** 30	HIGH RADIATION AREA HIGHLY CONTAMINATED AREA

DOSIMETRY REQUIREMENTS

✔ TLD AND DRD REQUIRED

WORKER INSTRUCTIONS

☐ ENSURE HIGH RADIATION AREA BOUNDARY IS IN PLACE UPON ENTRY AND AT EXIT
✔ CONTACT RP PRIOR TO VENTING OR DRAINING CONTAMINATED SYSTEMS
☐ **RP MUST BE PRESENT TO REMOVE ITEMS FROM CONTAMINATED AREA** (There is a toolbox in each area, no need to remove tools)
☐ BREACHING RADIOLOGICAL ROPE BOUNDARY IS NOT ALLOWED UNLESS APPROVED BY RP
☐ CONTACT RP PRIOR TO ACCESSING OVERHEAD AREAS
✔ DRESS REQUIREMENTS FOR CONTAMINATED AREA WORK:
FOLLOW DRESS REQUIREMENTS AT STEP OFF PAD
☐ DRESS REQUIREMENTS FOR HIGHLY CONTAMINATED AREA WORK:
GREENS, HI-TOPS, SHOE RUBBERS, COTTON LINERS, SURGICALS, RUBBER GLOVES, DOUBLE COTTON COVERALLS. A SINGLE FRAM/NYLON SUIT MAY BE USED IN LIEU OF 2 PAIR OF COVERALLS. A SINGLE FRAM/NYLON SUIT OR COTTON COVERALLS WITH A WATER RESISTANT SUIT IS REQUIRED FOR WORKING IN WET ENVIRONMENTS.
✔ TAPE WRIST AND ANKLES
☐ **IF DIGITAL DOSIMETER ALARMS, PLACE EQUIPMENT IN A SAFE CONDITION, INFORM OTHERS, EXIT AREA AND CONTACT RP.** (conditional)
☐ DOUBLE STEP-OFF-PAD -ADD A PAIR OF HI-TOPS PLACE PD-1 & TLD IN BAG W/BETA WINDOW OUTWARD AND SECURE ON OUTSIDE OF COVERALLS DRESS REQUIREMENTS FOR TOURS IN HCA' S: FULL SET OF PC'S W/FLIGHT CAP.
✔ WORKERS MUST PERIODICALLY CHECK THEIR DOSIMETER

RP PERSONEL INSTRUCTIONS

CONTINUOUS COVERAGE REQUIRED FOR WORK IN AREAS > 1,000 MREM/HOUR AT 12" LOCKED HIGH RADIATION AREA -ENSURE DOOR IS LOCKED
LOCKED HIGH RADIATION AREA -CONTROLLED ACCESS TO AREA REQUIRED
ATTENDANCE COVERAGE REQUIRED FOR ENTRY INTO LOCKED HIGH RADIATION AREAS
IF AIRBORNE CONCENTRATIONS EXCEED 30% OF A DAC, STOP WORK AND EVALUATE ENGINEERING CONTROLS FOR ADEQUACY AND POSSIBLE RWP REVISION. ALL ENTRIES INTO HRA'S AND LHRA'S REQUIRE A PRE-JOB BRIEFING BY AN RP TECHNICIAN USING THE RP-800 FORM PRIOR TO ENTRY. BRIEFINGS FOR ENTRIES INTO LHRA'S WILL BE DOCUMENTED. THIS BRIEFING WILL INCLUDE ESTABLISHED STAY TIMES, AND WORK AREAS WILL HAVE TURN BACK DOSE RATES.

COMMENTS

PD-l ALARM SETPOINTS LISTED ARE DEFAULT SETTINGS ONLY. ACTUAL SETPOINTS USED MAY BE DIFFERENT BASED ON TASK(S) BEING PERFORMED. ACTUAL SETPOINTS WILL BE DISPLAYED WHEN LOGGING ONTO THE RWP. NO SPOT OR MINOR MAINTENANCE TO BE PERFORMED ON THIS SWP. REV #4 -BASED ON CHANGES TO RP & WORKER *I* S REQUIREMENTS. .

ATTACHMENT 2

Directions to Trainee:

When I tell you to begin, you are to determine the radiation protection requirements required to perform CRD Drive System Venting above the HCUs in the Reactor Building 903 South. This includes removal of any tools from the area following task completion.

During task performance, state the actions you are taking, e.g.: repositioning controls and observing instrumentation.

General Conditions:

1. The plant is shutdown for a refueling outage.
2. The CRD system has been placed in service.
3. CRD Drive System Venting per 2.2.8 is scheduled for this shift.
4. The Area over the CRD HCUs is a <u>radiation area</u> and a <u>contaminated area</u>.

Initiating Cues:

Evaluate RWP 20031010 and checkmark the **<u>minimum</u>** radiological protection requirements needed in order to perform CRD drive system venting.

ATTACHMENT 2

RWP#	TASK#	REVISION	BUILDING	ELEVATION	AREA
20031010	01	04	RCA	ALL	VARIOUS

RWP TITLE
OPERATOR ROUNDS IN SWP AREAS

TASK TITLE
MIGRATED TASK DATA

WORK DESCRIPTION
OPERATOR ROUNDS IN SWP AREAS

MAX RAD LEVELS (mrem/hr) 2200@CONTACT 1400@12" 200@GENERAL ROOM

CONTACT RP FOR CURRENT RADIOLOGICAL CONDITIONS!!!

RWP CATEGORY/TYPE	ROUTINE/PSE	ALARMS (Mr)	CONDITION REQUIRING SWP
SWP / Routine	Routine	**RATE:** 1200 **DOSE:** 30	HIGH RADIATION AREA HIGHLY CONTAMINATED AREA

DOSIMETRY REQUIREMENTS
☐ TLD AND DRD REQUIRED

WORKER INSTRUCTIONS
☐ ENSURE HIGH RADIATION AREA BOUNDARY IS IN PLACE UPON ENTRY AND AT EXIT
☐ CONTACT RP PRIOR TO VENTING OR DRAINING CONTAMINATED SYSTEMS
☐ RP MUST BE PRESENT TO REMOVE ITEMS FROM CONTAMINATED AREA
☐ BREACHING RADIOLOGICAL ROPE BOUNDARY IS NOT ALLOWED UNLESS APPROVED BY RP
☐ CONTACT RP PRIOR TO ACCESSING OVERHEAD AREAS
☐ DRESS REQUIREMENTS FOR CONTAMINATED AREA WORK:
FOLLOW DRESS REQUIREMENTS AT STEP OFF PAD
☐ DRESS REQUIREMENTS FOR HIGHLY CONTAMINATED AREA WORK:
GREENS, HI-TOPS, SHOE RUBBERS, COTTON LINERS, SURGICALS, RUBBER GLOVES, DOUBLE COTTON COVERALLS. A SINGLE FRAM/NYLON SUIT MAY BE USED IN LIEU OF 2 PAIR OF COVERALLS. A SINGLE FRAM/NYLON SUIT OR COTTON COVERALLS WITH A WATER RESISTANT SUIT IS REQUIRED FOR WORKING IN WET ENVIRONMENTS.
☐ TAPE WRIST AND ANKLES
☐ IF DIGITAL DOSIMETER ALARMS, PLACE EQUIPMENT IN A SAFE CONDITION, INFORM OTHERS, EXI T AREA AND CONTACT RP.
☐ DOUBLE STEP-OFF-PAD -ADD A PAIR OF HI-TOPS PLACE PD-1 & TLD IN BAG W/BETA WINDOW OUTWARD AND SECURE ON OUTSIDE OF COVERALLS DRESS REQUIREMENTS FOR TOURS IN HCA' S: FULL SET OF PC'S W/FLIGHT CAP.
☐ WORKERS MUST PERIODICALLY CHECK THEIR DOSIMETER

RP PERSONEL INSTRUCTIONS
CONTINUOUS COVERAGE REQUIRED FOR WORK IN AREAS > 1,000 MREM/HOUR AT 12" LOCKED HIGH RADIATION AREA -ENSURE DOOR IS LOCKED
LOCKED HIGH RADIATION AREA -CONTROLLED ACCESS TO AREA REQUIRED
ATTENDANCE COVERAGE REQUIRED FOR ENTRY INTO LOCKED HIGH RADIATION AREAS
IF AIRBORNE CONCENTRATIONS EXCEED 30% OF A DAC, STOP WORK AND EVALUATE ENGINEERING CONTROLS FOR ADEQUACY AND POSSIBLE RWP REVISION. ALL ENTRIES INTO HRA'S AND LHRA'S REQUIRE A PRE-JOB BRIEFING BY AN RP TECHNICIAN USING THE RP-800 FORM PRIOR TO ENTRY. BRIEFINGS FOR ENTRIES INTO LHRA'S WILL BE DOCUMENTED. THIS BRIEFING WILL INCLUDE ESTABLISHED STAY TIMES, AND WORK AREAS WILL HAVE TURN BACK DOSE RATES.

COMMENTS
PD-l ALARM SETPOINTS LISTED ARE DEFAULT SETTINGS ONLY. ACTUAL SETPOINTS USED MAY BE DIFFERENT BASED ON TASK(S) BEING PERFORMED. ACTUAL SETPOINTS WILL BE DISPLAYED WHEN LOGGING ONTO THE RWP. NO SPOT OR MINOR MAINTENANCE TO BE PERFORMED ON THIS SWP. REV #4 -BASED ON CHANGES TO RP & WORKER *I* S REQUIREMENTS. .

PREPARED BY	DATE	APPROVED BY	DATE
R. ENGEBRETSON	3/01/03	TJFRANCIS	3/01/03

Task No.: 200152P0501

==

Task Title: Perform Dose Assessment

==

Trainee: ____________________________ Examiner: ______________________________

Pass:___ Fail:____ Examiner Signature:__________________________ Date:_________

Additional Program Information:

1. Appropriate Performance Locations: Any
2. Appropriate Trainee level: RO / SRO / STE
3. Evaluation Method: Perform ___
4. Performance Time: minutes
5. NRC K/A 2.4.39 (3.3/3.1)

Directions to Examiner:

1. This JPM evaluates the trainee's ability to perform a Dose Projection on a PMIS terminal per 5.7.17.
2. All blanks must be filled out with either initials or an "NP" for "not performed"; an explanation may also be written in the space, if desired, by the examiner.
3. Give the trainee his copy of the Directions to the Trainee (Attachment 2) when ready to start the JPM.
4. Brief the trainee and tell the trainee to begin.

Directions to Trainee:

When I tell you to begin, you are to perform a Dose Projection on a PMIS terminal per 5.7.17. Before you start, I will state the general plant conditions, the Initiating Cues, and answer any questions you may have.

During task performance, state the actions you are taking, e.g.: repositioning controls and observing instrumentation.

==

Task No.: 200152P0501

===

Task Title: Perform Dose Assessment

===

Task No.: 200152P0501

==

Task Title: Perform Dose Assessment

==

General Conditions:

1. A **General Emergency** has been declared under EAL 2.4.1.
2. The plant scrammed **1 hour** ago following a major accident.
3. Containment venting is in progress through SGT due to primary containment pressure approaching PCPL. Venting is expected for **1 hour**.
4. The core **is** degraded.
5. The stability class is **"D"**.
6. The wind is at **15 mph**.
7. The wind is from **090 °**
8. ERP release rate is **1E6 µci/sec.**

General References:

1. Procedure 5.7.17, Dose Assessment

General Tools and Equipment:

1. PMIS Terminal

Special Conditions, References, Tools, Equipment:

1. Critical steps denoted by "*".

Task Standards:

1. 100% of critical elements successfully completed without error.
2. 100% of safety and radiological work practices.
3. Values obtained for the dose assessment match those on Attachment 1.

Initiating Cue(s):

The Shift Manager directs you to perform a Computer Dose Projection per 5.7.17 using PMIS and to provide him with a printed copy of the dose projection when completed.

Task No.: 200152P0501

Task Title: Perform Dose Assessment

Performance Checklist	Standards	Initials
1. Obtain procedure 5.7.17.	Current revision of procedure 5.7.17 obtained.	_______
2. Starts dose projection program on PMIS terminal.	**DOSE** entered on a logged on PMIS terminal.	_______*

NOTE: **The following step is NOT critical if PMIS has defaulted to Elevated Release Point as the origin of the release.**

3. Selects/Ensures origin of the release.	**Elevated Release Point (ERP)** selected as the release point.	_______*
4. Enters the release rate.	**1E6 µCi/sec** release rate entered.	_______*
5. Selects release path through SGT.	**YES** entered for release path through SGT.	_______*
6. Enters the length of the release.	**1 hour** entered for length of release.	_______*
7. Selects Degraded Core.	**YES** is selected for degraded core.	_______*
8. Enters wind direction.	**090°** entered for wind direction.	_______*

Task No.: 200152P0501

Task Title: Perform Dose Assessment

Performance Checklist	Standards	Initials
9. Enters Wind Speed.	**15 mph** entered for wind speed.	_______*
10. Enters stability class.	**"D"** is entered for stability class.	_______*
11. Enters time since shutdown.	**1 hour** entered for time since shutdown.	_______*

NOTE: The following step is NOT critical if PMIS has defaulted to NO for reactor building release path.

Performance Checklist	Standards	Initials
12. Determines if release is through the Reactor Building.	"**NO**" entered for release path through the reactor building.	_______*
13. Selects RESULTS.	**RESULTS** Selected using F3 key.	_______*
14. Selects PRINT.	**PRINT** selected by using F5 key. **CUE: Hard copy prints.**	_______*
15. Informs the Shift Manager that the DOSE projection is complete and provides hard copy of the dose projection.	Shift Manager informed. **CUE: Shift Manager informed and he has accepted the printed copy of the dose projection.**	_______

NOTE: The following step is completed by the JPM evaluator ONLY.

Task No.: 200152P0501

==

Task Title: Perform Dose Assessment

==

Performance Checklist	Standards	Initials
16. Compare the values on the hard copy provided by the candidate to the values in ATTACHMENT 1.	Data provided on the hard copy matches the data in ATTACHMENT 1.	_______*

SIMULATOR SET-UP

A. Materials Required

None

B. Initialize the Simulator in any power IC (IC-20 recommended).

Batch File Name - None

C. Change the simulator conditions as follows:

1. Triggers

   None

2. Malfunctions

   None

3. Remotes

   None

4. Overrides

   None

5. Panel Setup

   a. If using the simulator PMIS enter substitute value for ERP release rate of 1E6 µCi/sec.

Task No.: 200152P0501

==

Task Title: Perform Dose Assessment

==

ATTACHMENT 1

DOSE Projection Data				
Distance From Plant	Projected Integrated Dose (Rem)		Projected Dose Rate (Rem/hr)	
	TEDE	CDE (Thyroid)	TEDE	CDE (Thyroid)
1 Mile	3.66E-04	9.86E-04	3.66E-04	9.86E-04
2 Miles	3.60E-04	9.69E-04	3.60E-04	9.69E-04
5 Miles	1.65E-04	4.45E-04	1.65E-04	4.45E-04
10 Miles	7.98E-05	2.15E-04	7.98E-05	2.15E-04

ATTACHMENT 2

Directions to Trainee:

When I tell you to begin, you are to perform a Dose Projection on a PMIS terminal per 5.7.17. Before you start, I will state the general plant conditions, the Initiating Cues, and answer any questions you may have.

During task performance, state the actions you are taking, e.g.: repositioning controls and observing instrumentation.

General Conditions:

1. A **General Emergency** has been declared under EAL 2.4.1.
2. The plant scrammed **1 hour** ago following a major accident.
3. Containment venting is in progress through SGT due to primary containment pressure approaching PCPL. Venting is expected for **1 hour**.
4. The core **is** degraded.
5. The stability class is **"D"**.
6. The wind is at **15 mph**.
7. The wind is from **090 °**
8. ERP release rate is **1E6 µci/sec.**

Initiating Cue(s):

The Shift Manager directs you to perform a Computer Dose Projection per 5.7.17 using PMIS and to provide him with a printed copy of the dose projection when completed.

Task No. 344050O0503

==

Task Title: Security Emergency

==

Trainee: ______________________________ Examiner: ___________________________

Pass:___ Fail:___ Examiner Signature: ________________________ Date: ________

Additional Program Information:

1. Appropriate Performance Locations: SIM / CR
2. Appropriate Trainee Level: SRO / STE
3. Evaluation Method: _____Perform
4. Performance Time: 10 minutes
6. NRC K/A 2.4.28 (2.3/3.3)

Directions to Examiner:

1. This JPM evaluates the trainee's ability to perform the required actions for a Security Emergency.
2. All blanks must be filled out with either initials or an "NP" for "not performed"; an explanation may also be written in the space, if desired, by the examiner.
3. Give the trainee his copy of the Directions to the Trainee (Attachment 1) when ready to start the JPM.
4. Brief the trainee, place the Simulator in RUN, and tell the trainee to begin.

Directions to Trainee:

When I tell you to begin, you are to respond to a Security Emergency as appropriate to the provided conditions. Before you start, I will state the general plant conditions, the Initiating Cues, and answer any questions you may have.

When simulating, physically point to any meters, gauges, recorders and controls you would be using. State the position of controls as you would have manipulated them in order to complete the assigned task. **ALL PLANT ANNOUNCEMENTS WILL BE SIMULATED!**

Task No. 344050O0503

==

Task Title: Security Emergency

==

General Conditions:

1. The plant is operating at 100% power.
2. The Shift Supervisor has delegated you to direct the plant response.
3. The board operators are available ONLY to support control room actions.
4. You must personally perform any communications required for this situation.
5. The Shift Supervisor will perform all Emergency Director required actions.

General References:

1. Emergency Procedure 5.5SECURITY

General Tools and Equipment:

1. Site communication System.
2. Emergency Telephone number book.

Special Conditions, References, Tools, Equipment:

1. Critical checks denoted by "*".

Task Standards:

1. 100% of critical elements successfully completed without error.
2. 100% of safety and radiological work practices.

Initiating Cue(s):

At 9:35 p.m. you are informed by the Radwaste Supervisor that armed intruders have been seen in the protected area. The intruders last known location was the 903 corridor outside the door to the reactor building. The intruders are carrying automatic rifles, pistols and grenades.

Task No. 344050O0503

==

Task Title: Security Emergency

==

You are to perform the required control room actions for this situation.

Task No. 344050O0503

==

Task Title: Security Emergency

==

Performance Checklist	Standards	Initials
1. Obtain Procedure 5.5SECURITY.	Current revision of 5.5SECURITY obtained.	_________
2. Contact the Security Shift Supervisor to confirm threat.	Security Shift Supervisor contacted and asked to confirm the threat. #**CUE**: As the Security Shift Supervisor, indicate that the intruders posses explosive devices and automatic weapons. **An actual Security Threat exists with high potential for significant structural damage and personnel injury**.	_________
1. Determines that an Actual Threat exists and enters 5.5Security Attachment 2.	Enters Attachment 2 # **CUE**: If asked, a credible insider threat **DOES NOT** exist.	_________
2. Notifies plant personnel of the threat and the nature of the threat.	Plant personnel are informed that: - Intruders have entered the protected area and, - all personnel should take cover.	_________*
3. Directs the reactor be scrammed.	RO directed to scram the reactor. # **CUE**: As the RO, acknowledge the order and report that the reactor is scrammed.	_________*

Task No. 344050O0503

==

Task Title: Security Emergency

==

Performance Checklist	Standards	Initials
4. Directs the BOP to ensure that BF-C-1A, EMERGE BSTR FAN is running.	BOP directed to ensure that BF-C-1A, EMERGE BSTR FAN is running. # **CUE**: As the BOP acknowledge the order and report that BF-C-1A, EMERG BSTR FAN is running.	_________*
5. Directs the BOP to start and run unloaded both DGs.	BOP directed to start and run unloaded both DGs. # **CUE**: As the BOP acknowledge the order and report that both DGs are running unloaded.	_________*
6. Directs or Starts Electric Fire pump 1E or 1D.	Directs the BOP to start the 1E or 1D Fire Pump or places the 1E or 1D fire pump control switch to start. # **CUE**: If BOP directed then acknowledge the order as the BOP and report that the motor driven Fire Pump is running. (If the CRS starts the pump the RED light is ON and the GREEN light is OFF for the pump started.)	_________ *

Task No. 344050O0503

Task Title: Security Emergency

Performance Checklist	Standards	Initials
7. Directs Station Operators to obtain fire fighting gear and report to the Control Room.	Station Operators directed to obtain fire fighting gear and to report to the Control Room. # **CUE**: Indicate that the Reactor Building Station Operator and the Radwaste Station Operator have reported to the Control Room. #**CUE**: The Turbine Building operator contacts the Control Room and reports that he is unable to get to the Control Room at this time. He is just outside the #1 DG room and the 903 area just outside the doors to the control building is blocked by the armed intruders. The intruders are currently unaware of his presence. The Station Operator requests direction from you.	__________ *
8. Directs the TB Station Operator to card into the DG room.	Station Operator directed to enter the DG room. # **CUE**: TB operator acknowledges the order and reports that he has entered the DG room.	__________*
9. Notifies the Operations Manager and the Operations Supervisor.	Phones the Operations Manager and the Operations Supervisor and informs them of the Security Emergency. # **CUE**: Acknowledge report. Inform the student the JPM is complete.	__________

ATTACHMENT 1

Directions to Trainee:

When I tell you to begin, you are to respond to a Security Emergency as appropriate to the provided conditions. Before you start, I will state the general plant conditions, the Initiating Cues, and answer any questions you may have.

When simulating, physically point to any meters, gauges, recorders and controls you would be using. State the position of controls as you would have manipulated them in order to complete the assigned task. **ALL PLANT ANNOUNCEMENTS WILL BE SIMULATED!**

General Conditions:

1. The plant is operating at 100% power.
2. The Shift Supervisor has delegated you to direct the plant response.
3. The board operators are available ONLY to support control room actions.
4. You must personally perform any communications required for this situation.
5. The Shift Supervisor will perform all Emergency Director required actions.

Initiating Cue(s):

At 9:35 p.m. you are informed by the Radwaste Supervisor that armed intruders have been seen in the protected area. The intruders last known location was the 903 corridor outside the door to the reactor building. The intruders are carrying automatic rifles, pistols and grenades.

You are to perform the required control room actions for this situation.

Task No.: 200226A0501

==

Task Title: AVERAGE DRYWELL TEMPERATURE CALCULATION (Alternate Path)

==

Trainee: ___________________________ Examiner: ______________________________

Pass:______ Fail:______ Examiner Signature: ____________________ Date: _________

ALTERNATE PATH

Additional Program Information:

1. Appropriate Performance Locations: **SIMULATOR ONLY!!**
2. Appropriate Trainee level: RO / SRO / STE
3. Evaluation Method: __ **Perform in simulator only**
4. Performance Time: 10 minutes
5. NRC K/A 216000 K5.07 (3.6/3.8), A1.01 (3.2/3.3)

Directions to Examiner:

NOTE: THIS IS AN **ALTERNATE PATH** JPM. ONE OF THE TEMPERATURES IS UNAVAILABLE AND THE ALTERNATE METHOD OF CALCULATION MUST BE USED.

NOTE: The standards for the values collected from the panels will be based on the evaluator's reading of the same parameter. The average value calculated standard will be based on the values collected.

1. This JPM evaluates the trainee's ability to perform an average drywell temperature calculation.
2. Only the cues preceded by "#" should be given.
3. All blanks must be filled out with either initials or an "NP" for "not performed"; an explanation may also be written in the space if desired by the examiner.
4. Give the trainee his copy of the Directions to the Trainee (Attachment 2) when ready to start the JPM.
5. Brief the trainee, place the simulator in run, and tell the trainee to begin.

Task No.: 200226A0501

==

Task Title: AVERAGE DRYWELL TEMPERATURE CALCULATION (Alternate Path)

==

Directions to Trainee:

When I tell you to begin, you are to perform an average drywell temperature calculation. Before you start, I will state the general plant conditions, the Initiating Cues and answer any questions you may have.

During task performance, state the actions you are taking, e.g.: repositioning controls and observing instrumentation.

==

General Conditions:

1. PMIS is unavailable for accurate average drywell temperature calculations.

General References:

1. Procedure 5.8.10, Average Drywell Temperature Calculation

General Tools and Equipment:

1. Calculator

Special Conditions, References, Tools, Equipment:

1. Simulator Setup: See Attachment 1.
2. Critical checks denoted by "*".
3. Simulator cues denoted by "#".

Task No.: 200226A0501

==

Task Title: AVERAGE DRYWELL TEMPERATURE CALCULATION (Alternate Path)

==

Task Standards:

1. 100% of critical elements successfully completed without error.
2. 100% of safety and radiological work practices.

Initiating Cue(s):

The CRS has directed you to perform an average drywell temperature calculation per procedure 5.8.10. Inform the CRS of the average drywell temperature when complete.

NOTE: Place the Simulator in RUN and tell the trainee to begin.

Task No.: 200226A0501

Task Title: AVERAGE DRYWELL TEMPERATURE CALCULATION (Alternate Path)

Performance Checklist	Standards	Initials
1. Refer to Procedure 5.8.10.	Refers to Procedure 5.8.10, Section 3.0, and Attachment 1.	_________
2. Record time.	Record time on Attachment 1 (± 1 minute of simulator time on panel 9-5).	_________
3. Record PC-TI-508A, RX HEAD FLANGE TEMP (0˚)	Record PC-TI-508A, RX HEAD FLANGE TEMP (0˚) on Attachment 1 (± 5˚F of indicated value).	_________
4. Record PC-TI-505E, ZONE 2B TEMP (288˚)	Record PC-TI-505E, ZONE 2B TEMP (288˚) on Attachment 1 (± 5˚F of indicated value).	_________
5. Record PC-TI-510A, ZONE 2C TEMP (0˚)	Record PC-TI-510A, ZONE 2C TEMP (0˚) on Attachment 1 (± 5˚F of indicated value).	_________
6. Record PC-TR-502, RECIRC PUMPS AREA TEMP (RED: PUMP A)	Record PC-TR-502, RECIRC PUMPS AREA TEMP (RED: PUMP A) on Attachment 1 (± 5˚F of indicated value).	_________
7. Recognize Bad Data.	Recognize PC-TR-502, RECIRC PUMPS AREA TEMP (RED PUMP A) has bad data. Proceed to Attachment 2. **#CUE**: If asked, inform candidate to perform the actions specified in 5.8.10.	_________
8. Reference Attachment 2.	Refers to Procedure 5.8.10, Section 3.2, and Attachment 2.	_________

Task No.: 200226A0501

Task Title: AVERAGE DRYWELL TEMPERATURE CALCULATION (Alternate Path)

Performance Checklist	Standards	Initials
9. Record time.	Record time on Attachment 2 (± 1 minute of simulator time on panel 9-5).	________
10. Record temp.	Record PC-TI-505A TEMP on Attachment 2 (± 5˚F of indicated value).	________*
11. Record temp.	Record PC-TI-505B TEMP on Attachment 2 (± 5˚F of indicated value).	________*
12. Record temp.	Record PC-TI-505C TEMP on Attachment 2 (± 5˚F of indicated value).	________*
13. Record temp.	Record PC-TI-505D TEMP on Attachment 2 (± 5˚F of indicated value).	________*
14. Total the 4 readings.	Adds the readings for PC-TI-505A-D (SUM) and records total on Attachment 2 (Actual sum of the 4 values).	________*
15. Determine average.	Divides (SUM of the 4 readings) by 4 and records AVG DW TEMP on Attachment 2 (Actual number obtained by calculation).	________*
16. Informs CRS.	Informs the CRS of the AVG DW TEMP. **#CUE**: As the CRS, acknowledge the report.	________

ATTACHMENT 1

SIMULATOR SET-UP

A. Materials Required

None

B. Initialize the Simulator in any power IC (IC-20 recommended).

Batch File Name - None

C. Change the simulator conditions as follows:

1. Triggers

   None

2. Malfunctions (suggested. Any may be used that results in high drywell pressure)

Number	Title	Trigger	TD	Severity	Ramp	Initial
RR20A	Reactor Coolant Leakage Inside Containment.	A	0	10	0	N/A

3. Remotes

   None

===

Task Title: AVERAGE DRYWELL TEMPERATURE CALCULATION (Alternate Path)

===

4. Overrides

Instrument	Tag	Trigger	TD	Value	Ramp
PC-TR-502 Recirc Pumps Area Temp Recorder Pump A	ZAOPCTR502[1]	A	0	50	0

5. Panel Setup

   a. Insert override and malfunction above.

   b. Allow drywell conditions to stabilize.

   c. Place the Simulator in FREEZE.

Note: If this JPM is to be performed more than once, snap the simulator into an IC after the panel setup is complete.

ATTACHMENT 2

Directions to Trainee:

When I tell you to begin, you are to perform an average drywell temperature calculation per procedure 5.8.10. Before you start, I will state the general plant conditions, the Initiating Cues and answer any questions you may have.

During task performance, state the actions you are taking, e.g.: repositioning controls and observing instrumentation.

General Conditions:

1. PMIS is unavailable for accurate average drywell temperature calculations.

Initiating Cues:

The CRS has directed you to perform an average drywell temperature calculation per procedure 5.8.10. Inform the CRS of the average drywell temperature when complete.

Task No.: 200302G0203

==

Task Title: Determine Post-Maintenance Testing Requirements

==

Trainee: ____________________________ Examiner: _______________________________

Pass:___ Fail:____ Examiner Signature:__________________________ Date:_________

Additional Program Information:

1. Appropriate Performance Locations: Any
2. Appropriate Trainee level: SRO / STE
3. Evaluation Method: Perform ___ Simulate ___
4. Performance Time: minutes
5. NRC K/A 2.2.7 (2.0/3.2)

Directions to Examiner:

1. This JPM evaluates the trainee's ability to determine the post maintenance testing for RHR-MO-26A following bonnet gasket replacement per Maintenance Procedure 7.0.5, Post-maintenance Testing.
2. All blanks must be filled out with either initials or an "NP" for "not performed"; an explanation may also be written in the space, if desired, by the examiner.
3. Give the trainee his copy of the Directions to the Trainee (Attachment 1) when ready to start the JPM.
4. Brief the trainee and tell the trainee to begin.

Directions to Trainee:

When I tell you to begin, you are to determine the post maintenance testing for RHR-MO-26A following bonnet gasket replacement. Before you start, I will state the general plant conditions, the Initiating Cues, and answer any questions you may have.

When simulating, physically point to any meters, gauges, recorders and controls you would be using. State the position of controls as you would have manipulated them in order to complete the assigned task.

==

Task No.: 200302G0203

==

Task Title: Determine Post-Maintenance Testing Requirements

==

Task No.: 200302G0203

Task Title: Determine Post-Maintenance Testing Requirements

General Conditions:

1. The Reactor is shutdown with an outage in progress.

General References:

1. Procedure 7.0.5, POST MAINTENANCE TESTING
2. Procedure 0.26, SURVEILLANCE PROGRAM

General Tools and Equipment:

1. None

Special Conditions, References, Tools, Equipment:

1. Critical steps denoted by "*".

Task Standards:

1. 100% of critical elements successfully completed without error.
2. 100% of safety and radiological work practices.

Initiating Cue(s):

The Shift Manager directs you to determine the post maintenance testing requirements to assign to RHR-MO-26A following bonnet gasket replacement. Inform the Shift Manager when you have determined the requirements.

Task No.: 200302G0203

Task Title: Determine Post-Maintenance Testing Requirements

Performance Checklist	Standards	Initials
1. Obtain a copy of procedure 7.0.5, Post Maintenance Testing.	Current revision of procedure 7.0.5 is obtained.	_______
2. Identify component type for RHR-MO-26A.	Determines that RHR-MO-26A is a motor operated gate valve	_______
3. Locate the general component from Procedure 7.0.5 Attachment 1 index.	Candidate locates the Component Test Matrices for Motor Operated Valve (Gate/Globe).	_______
4. Identify, on the matrices, the type of corrective and/or preventive maintenance to be performed on RHR-MOV-26A.	Candidate Identifies Bonnet Gasket Replacement on the Matrices.	_______
5. Determine the test activities for the bonnet gasket replacement on RHR-MOV-26A.	From attachment 1, the candidate assigns Leak Test, Static VOTES test. (Open/Closed Flow test is not required for this valve.	_______*

NOTE: **In the following step the student MAY elect to perform only portions of the following surveillances as indicated in Procedure 0.26, Surveillance Program.**

6. Determine the test procedures indicated by Procedure 7.0.5, Attachment 2.	Candidate assigns 6.PC.501, 6.1RHR.201, and 6.MISC.401 to post maintenance testing.	_______*

Task No.: 200302G0203

Task Title: Determine Post-Maintenance Testing Requirements

Performance Checklist	Standards	Initials
7. Informs the Shift Manager of the post maintenance testing requirements to assign to the work package.	Shift manager informed. **#CUE:** Shift Manager acknowledges the report.	_______

ATTACHMENT 1

Directions to Trainee:

When I tell you to begin, you are to determine the post maintenance testing for RHR-MO-26A following bonnet gasket replacement. Before you start, I will state the general plant conditions, the Initiating Cues, and answer any questions you may have.

When simulating, physically point to any meters, gauges, recorders and controls you would be using. State the position of controls as you would have manipulated them in order to complete the assigned task.

General Conditions:

1. The Reactor is shutdown with an outage in progress.

Initiating Cue(s):

The Shift Manager directs you to determine the post maintenance testing requirements to assign to RHR-MO-26A following bonnet gasket replacement. Inform the Shift Manager when you have determined the requirements.

Task No.: 344022O0303

==

Task Title: Authorize Stable Iodine Thyroid Blocking

==

Trainee: ____________________________ Examiner: _______________________________

Pass:___ Fail:____ Examiner Signature:__________________________ Date:_________

Additional Program Information:

1. Appropriate Performance Locations: Any
2. Appropriate Trainee level: SRO
3. Evaluation Method: Perform ___ Simulate ___
4. Performance Time: 8 minutes
5. NRC K/As: 2.3.10 (2.9/3.3)

Directions to Examiner:

1. This JPM evaluates the trainee's ability to determine the need to authorize stable iodine thyroid blocking per 5.7.14, Stable Iodine Thyroid Blocking (KI).
2. If this JPM is performed on the Simulator, only the cues preceded by "#" should be given.
3. All blanks must be filled out with either initials or an "NP" for "not performed"; an explanation may also be written in the space, if desired, by the examiner.
4. Give the trainee his copy of the Directions to the Trainee (Attachment 1) when ready to start the JPM.
5. Brief the trainee, place the Simulator in RUN, and tell the trainee to begin.

Task No.: 344022O0303

==

Task Title: Authorize Stable Iodine Thyroid Blocking

==

Directions to Trainee:

When I tell you to begin, you are to perform the actions of the Emergency Director. Before you start, I will state the general plant conditions, the Initiating Cues, and answer any questions you may have.

When simulating, physically point to any meters, gauges, recorders and controls you would be using. State the position of controls as you would have manipulated them in order to complete the assigned task.

==

General Conditions:

1. The Reactor is shutdown in a refueling outage.
2. An accident due to a failure of the refuel floor crane has resulted in personnel injuries to two refuel floor workers and severe damage to several fuel bundles.
3. The two injured refuel floor workers have no immediate life threatening injuries but they are unable to leave the area on their own. No other personnel are currently present on the refueling floor.
4. The Emergency Director has declared a Site Area Emergency.
5. RMA-RA-1, FUEL POOL AREA, indicates **6E5 mrem/hr** and RMA-RA-2 FUEL POOL AREA is **upscale**.
6. No survey data or air samples are presently available from the refuel floor.
7. A Team of EMTs and RPs are standing by to evacuate the injured workers.

DOSE Projection Data				
Distance From Plant	Projected Integrated Dose (Rem)		Projected Dose Rate (Rem/hr)	
	TEDE	CDE (Thyroid)	TEDE	CDE (Thyroid)
1 Mile	2.75E-01	9.41E-07	6.88E-02	2.35E-07
2 Miles	1.33E+00	4.55E-06	3.33E-01	1.14E-06
5 Miles	1.24E+00	4.26E-06	3.11E-01	1.06E-06

===

Task Title: Authorize Stable Iodine Thyroid Blocking

===

10 Miles	8.36E-01	2.86E-06	2.09E-05	7.15E-07

General References:

1. Procedure 5.7.14, Stable Iodine Thyroid Blocking
2. Procedure 5.7.2, Shift Supervisor EPIP

General Tools and Equipment:

1. None

Special Conditions, References, Tools, Equipment:

1. Critical steps denoted by "*".

Task Standards:

1. 100% of critical elements successfully completed without error.
2. 100% of safety and radiological work practices.

Initiating Cue(s):

You are the Emergency Director, Attachment 3 of Procedure 5.7.2, Shift Supervisor EPIP, has been completed through step 1.16 (Initial Accountability complete). Determine if Potassium Iodide (KI) use should be authorized **AND** if authorized, determine who should it be authorized for.

Task No.: 344022O0303

Task Title: Authorize Stable Iodine Thyroid Blocking

Performance Checklist	Standards	Initials
1. Obtain copy of Procedure 5.7.2 Attachment 3.	Current revision of Procedure 5.7.2, Shift Supervisor EPIP, Attachment 3 obtained.	_______
2. Evaluate Attachment 3 Step 1.17 of 5.7.2, Shift Supervisor EPIP.	Determines that Stable Iodine Thyroid Blocking is indicated for the emergency workers involved in the rescue and Procedure 5.7.14 is required to be entered.	_______*
3. Obtain the current copy of Procedure 5.7.14 Stable Iodine Thyroid blocking.	Current revision of Procedure 5.7.14, Stable Iodine Thyroid Blocking is obtained.	_______

NOTE: **This JPM is not intended to evaluate the candidate distributing KI, ONLY to authorize the distribution of KI.**

NOTE: **The candidate may or may not authorize distribution of KI to the injured personnel on the refuel floor.**

Performance Checklist	Standards	Initials
4. Authorizes Stable Iodine Thyroid Blocking.	Radiological Manager is directed to distribute KI to the following: S Rescue personnel that will enter the refuel floor area. S Injured personnel on the refuel floor. (Authorization of KI to injured personnel not required to complete this critical step) **#CUE:** Radiological Manager acknowledges the order.	_______*

Task No.: 344022O0303

==

Task Title: Authorize Stable Iodine Thyroid Blocking

==

ATTACHMENT 1

Directions to Trainee:

When I tell you to begin, you are to perform the actions of the Emergency Director. Before you start, I will state the general plant conditions, the Initiating Cues, and answer any questions you may have.

When simulating, physically point to any meters, gauges, recorders and controls you would be using. State the position of controls as you would have manipulated them in order to complete the assigned task.

General Conditions:

1. The Reactor is shutdown in a refueling outage.
2. An accident due to a failure of the refuel floor crane has resulted in personnel injuries to two refuel floor workers and severe damage to several fuel bundles.
3. The two injured refuel floor workers have no immediate life threatening injuries but they are unable to leave the area on their own. No other personnel are currently present on the refueling floor.
4. The Emergency Director has declared a Site Area Emergency.
5. RMA-RA-1, FUEL POOL AREA, indicates **6E5 mrem/hr** and RMA-RA-2 FUEL POOL AREA is **upscale**.
6. No survey data or air samples are presently available from the refuel floor.
7. A Team of EMTs and RPs are standing by to evacuate the injured workers.

DOSE Projection Data				
Distance From Plant	Projected Integrated Dose (Rem)		Projected Dose Rate (Rem/hr)	
	TEDE	CDE (Thyroid)	TEDE	CDE (Thyroid)
1 Mile	2.75E-01	9.41E-07	6.88E-02	2.35E-07
2 Miles	1.33E+00	4.55E-06	3.33E-01	1.14E-06
5 Miles	1.24E+00	4.26E-06	3.11E-01	1.06E-06
10 Miles	8.36E-01	2.86E-06	2.09E-05	7.15E-07

Initiating Cue(s):

You are the Emergency Director, Attachment 3 of Procedure 5.7.2, Shift Supervisor EPIP, has been completed through step 1.16 (Initial Accountability complete). Determine if Potassium Iodide (KI) use should be authorized **AND** if authorized, determine who should it

be authorized for.

Task No.: 200335G0503

==

Task Title: PAR Table Top #4

==

Trainee: ______________________________ Examiner: ____________________________

Pass ___ Fail ___ Examiner Signature: _________________________ Date: ________

Additional Program Information:

1. Appropriate Performance Locations: Classroom / Simulator
2. Appropriate Trainee Levels: SRO / STE
3. Evaluation Method: __ Simulate __ Perform
4. Performance Time: 10 minutes
5. NRC K/A: 2.4.44 (2.1 / 4.0)

Directions to Examiner:

NOTE: THIS IS A TIME CRITICAL JPM

- 10 minute time limit
- The time starts when in the control room or simulator and the examiner has told the student to begin.
- The clock ends when the completed Attachment 1 is returned to the examiner.

1. This JPM evaluates the trainee's ability to make a PAR Recommendation.
2. All blanks must be filled out with either initials or an “NP” for “not performed”; an explanation may also be written in the space, if desired, by the examiner.
3. Give the trainee his copy of the Directions to the Trainee (Attachment 1) when ready to start the JPM.
4. DO NOT allow the trainee to pre-review the associated procedures.
5. Brief the trainee and tell the trainee to begin.

Directions to Trainee:

When I tell you to begin, you are to determine Protective Action Recommendations for the provided conditions. Before you start, I will state the general plant conditions, the Initiating Cues, and answer any questions you may have.

During task performance, it is recommended that you limit your discussion/demonstration to the minimum required. The examiner will ask questions of you if necessary to understand your actions..

Task No.: 200335G0503

Task Title: PAR Table Top #4

THIS IS A TIME CRITICAL JPM.

General Conditions:

1. A General Emergency has been declared under EAL 2.4.1.
2. The plant scrammed 1 hour ago following a major accident.
3. Containment venting is in progress through SGT due to primary containment pressure approaching PCPL. Venting is expected for 1 hour.
4. The core **is** degraded.
5. The stability class is “D”.
6. The wind is at 15 mph.
7. The wind is from 090 ˚
8. There is NO precipitation
9. PMIS is not available.

DOSE Projection Data				
Distance From Plant	Projected Integrated Dose (Rem)		Projected Dose Rate (Rem/hr)	
	TEDE	CDE (Thyroid)	TEDE	CDE (Thyroid)
1 Mile	3.66E-04	9.86E-04	3.66E-04	9.86E-04
2 Miles	3.60E-04	9.69E-04	3.60E-04	9.69E-04
5 Miles	1.65E-04	4.45E-04	1.65E-04	4.45E-04
10 Miles	7.98E-05	2.15E-04	7.98E-05	2.15E-04

General References:

1. Procedure 5.7.1, EAL Matrix
2. Procedure 5.7.6, Attachment 3
3. Procedure 5.7.20

General Tools and Equipment:

1. None

Task No.: 200335G0503

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Task Title: PAR Table Top #4

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Task No.: 200335G0503

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Task Title: PAR Table Top #4

==

Special Conditions, References, Tools, Equipment:

1. Critical checks denoted by "*".

Task Standards:

1. 100% of critical elements successfully completed without error.
2. 100% of safety and radiological work practices.
3. ***This JPM must be completed within 10 minutes.*** The time starts when the examiner tell the student to begin and ends when the completed Attachment 1 has been returned to the evaluator.

Initiating Cue(s):

You are to determine the appropriate Protective Action Recommendations (PARs) for the provided conditions. Complete the PAR table below and return this Attachment to the examiner when you have completed this task.

Task No.: 200335G0503

Task Title: PAR Table Top #4

Performance Checklist	Standards	Initials
RECORD START TIME __________________________		
NOTE:	**Per 5.7.2, the trainee must reference 5.7.20 for the provided conditions.**	
1. Refer to Procedure 5.7.2.	The operator refers to Procedure 5.7.2. Determine 5.7.20 must be used for PARs.	_________
2. Refer to Procedure 5.7.20.	The operator refers to Procedure 5.7.20.	_________
3. Determines affected sectors.	The operator determines the affected sectors for wind direction of 090°. **Sectors M, N, P**	_________*
4. Determines Sectors to Evacuate	The Operator determines the following Evacuation sectors: **<u>Evacuate</u>** **0 - 2 miles** **All sectors** **2 - 5 miles** **Sectors M, N, P**	_________*
5. Determines Sectors that must Go Indoors and monitor EAS/EBS	The Operator determines the following GO INDOORS sectors: **<u>GO INDOORS</u>** **All remaining sectors not evacuated.**	_________*
6. Turns in completed paperwork.	Returns completed Attachment 1 to evaluator.	_________

Task No.: 200335G0503

Task Title: PAR Table Top #4

Performance Checklist	Standards	Initials
RECORD STOP TIME ______________________		TOTAL TIME _______________
7. JPM completion time	JPM completed in 10 minutes or less.	_________*

ATTACHMENT 1 (Page 1 of 2)

When I tell you to begin, you are to determine Protective Action Recommendations for the provided conditions. Before you start, I will state the general plant conditions, the Initiating Cues, and answer any questions you may have.

During task performance, it is recommended that you limit your discussion/demonstration to the minimum required. The examiner will ask questions of you if necessary to understand your actions..

THIS IS A TIME CRITICAL JPM.

General Conditions:

1. A General Emergency has been declared under EAL 2.4.1.
2. The plant scrammed 1 hour ago following a major accident.
3. Containment venting is in progress through SGT due to primary containment pressure approaching PCPL. Venting is expected for 1 hour.
4. The core <u>**is**</u> degraded.
5. The stability class is “D”.
6. The wind is at 15 mph.
7. The wind is from 090 ˚
8. There is NO precipitation
9. PMIS is not available.

DOSE Projection Data				
Distance From Plant	Projected Integrated Dose (Rem)		Projected Dose Rate (Rem/hr)	
	TEDE	CDE (Thyroid)	TEDE	CDE (Thyroid)
1 Mile	3.66E-04	9.86E-04	3.66E-04	9.86E-04
2 Miles	3.60E-04	9.69E-04	3.60E-04	9.69E-04
5 Miles	1.65E-04	4.45E-04	1.65E-04	4.45E-04
10 Miles	7.98E-05	2.15E-04	7.98E-05	2.15E-04

ATTACHMENT 1 (Page 2 of 2)

Initiating Cue(s):

You are to determine the appropriate Protective Action Recommendations (PARs) for the provided conditions. Complete the PAR table below and return this Attachment to the examiner when you have completed this task.

Protective Action Recommendations (PARS)			
	None	**Evacuate Sectors**	**Go indoors and monitor EAS/EBS in Sectors**
0-2 Miles			
2-5 Miles			
5-10 Miles			

REMARKS for 5.7.6 Attachment 3: ____

____

____

____

Task No.: 201035P0401

==

Task Title: Respond To A Control Rod Drifting Out (Alternate Path)

==

Trainee: ____________________________ Examiner: _______________________________

Pass:___ Fail:____ Examiner Signature:__________________________ Date:_________

ALTERNATE PATH

Additional Program Information:

1. Appropriate Performance Locations: CR / SIM
2. Appropriate Trainee level: RO / SRO
3. Evaluation Method: Perform ___ Simulate ___
4. Performance Time: 10 minutes
5. NRC K/As 201002 A2.02 (3.2/3.3)

Directions to Examiner:

NOTE: THIS IS A **ALTERNATE PATH** JPM. THE CONTROL ROD WILL CONTINUE TO MOVE OUT AFTER INITIALLY BEING NOTCHED IN.

1. This JPM evaluates the trainee's ability to respond to a control rod drifting out per 2.4CRD..
2. If this JPM is performed on the Simulator, only the cues preceded by "#" should be given.
3. All blanks must be filled out with either initials or an "NP" for "not performed"; an explanation may also be written in the space, if desired, by the examiner.
4. Give the trainee his copy of the Directions to the Trainee (Attachment 2) when ready to start the JPM.
5. Brief the trainee, place the Simulator in RUN, and tell the trainee to begin.

Task No.: 201035P0401

==

Task Title: Respond To A Control Rod Drifting Out (Alternate Path)

==

Directions to Trainee:

When I tell you to begin, you are to perform actions as appropriate to panel 9-5 indications. Before you start, I will state the general plant conditions, the Initiating Cues, and answer any questions you may have.

<u>If being simulated In-Plant or Control Room:</u>

When simulating, physically point to any meters, gauges, recorders and controls you would be using. State the position of controls as you would have manipulated them in order to complete the assigned task.

<u>If being performed in the Simulator:</u>

During task performance, state the actions you are taking, e.g.: repositioning controls and observing instrumentation.

==

General Conditions:

1. The plant is operating at power.
2. You are the CRO - Reactor Operator (RO) (9-5 PNL).

General References:

1. Procedure 2.4CRD, CRD Trouble.

Task No.: 201035P0401

==

Task Title: Respond To A Control Rod Drifting Out (Alternate Path)

==

General Tools and Equipment:

1. None

Special Conditions, References, Tools, Equipment:

1. Simulator Setup: See Attachment 1.
2. Critical steps denoted by “*”.
3. Simulator cues denoted by “#”.

Task Standards:

1. 100% of critical elements successfully completed without error.
2. 100% of safety and radiological work practices.

Initiating Cue(s):

The Control Room Supervisor directs you to perform actions as appropriate to panel 9-5 indications.

Task No.: 201035P0401

Task Title: Respond To A Control Rod Drifting Out (Alternate Path)

Performance Checklist	Standards	Initials
1. Assume the watch at panel 9-5.	The operator positions himself in a position to monitor panel 9-5.	_______

ACTION: After the candidate assumes the watch, DELETE malfunction RD12 to cause control rod 18-27 to drift out.

Performance Checklist	Standards	Initials
2. Respond to alarm **9-5-1/C-4**, ROD DRIFT.	The operator pulls the alarm card for **9-5-1/C-4**. **CUE**: Alarm **9-5-1/C-4**, ROD DRIFT is alarming	_______
3. Check full core display for rod drift light(s) to determine which rod(s) is drifting.	The operator Check full core display for rod drift light(s) to determine which rod(s) is drifting. **CUE**: The rod drift light for rod 18-27 is illuminated. **CUE**: IF asked, NO other control rods have drift lights illuminated.	_______
4. Enter procedure 2.4CRD.	The operator enters procedure 2.4CRD.	_______

Task No.: 201035P0401

Task Title: Respond To A Control Rod Drifting Out (Alternate Path)

Performance Checklist	Standards	Initials
NOTE: **The operator may have selected the control rod previously.**		
5. Select control rod 18-27.	Operator selects control rod 18-27 on panel 9-5. **CUE**: Control rod 18-27 white light is illuminated on the bench board and vertical portion of panel 9-5. **#CUE**: If asked, inform the operator that the control rod was at notch 26.	________*
6. Attempt to restore control rod position with Rod Movement Control or Emergency In switch.	Operator attempts to restore control rod position with Rod Movement Control or Emergency In switch. **CUE**: Control rod 18-27 moves in when driven, but does not latch.	________
NOTE: **After the control rod is fully inserted, malfunction RD10 will automatically DELETE to allow control rod 18-27 to latch.**		
7. Fully insert control rod 18-27.	Operator fully inserts control rod 18-27. **CUE**: Control rod 18-27 is fully inserted.	________*

Task No.: 201035P0401

Task Title: Respond To A Control Rod Drifting Out (Alternate Path)

Performance Checklist	Standards	Initials
8. Dispatch an operator to panel 9-16.	Operator dispatches operator to panel 9-16 to scram control rod 18-27 if it does not latch. **#CUE**: An operator is present at panel 9-16 and is prepared to scram control rod 18-27	_______
9. Releases continuous insert switch.	Operator releases continuous insert switch. **CUE**: Control rod 18-27 settles at position “00".	_______*
10. Contact Reactor Engineer and System Engineer to evaluate.	Operator asks CRS to have Reactor Engineer and System Engineer to evaluate. **#CUE**: CRS acknowledges report. Inform candidate that the JPM is complete.	_______

==
Task Title: Respond To A Control Rod Drifting Out (Alternate Path)
==

ATTACHMENT 1

SIMULATOR SET-UP

A. Materials required

None

B. Initialize the Simulator in any IC that will support full power operation (IC-18, 19 or 20 suggested)

Batch File name - none.

C. Change the Simulator conditions from those of the IC as follows:

1. Triggers

Number	File Name	Description
3	None	trgset 3 “ycacr067 == 0" (Control rod 18-27 at position 0) trg 3 “dmf rd101827" (Trigger 3 deletes malfunction rd101827 when activated)

2. Malfunctions

Number	Title	Trigger	TD	Severity	Ramp	Initial
RD10	Control Rod 18-27 drift out	A	N/A	N/A	N/A	N/A
RD12	Control Rod 18-27 Stuck	E1	0	N/A	N/A	N/A

3. Remotes

Number	**Title**	**Trigger**	**TD**	**Value**	**Ramp**
	None				

4. Overrides

Instrument	**Tag**	**Trigger**	**TD**	**Value**	**Ramp**
None					

5. Panel Set-up (suggested. Any setup is allowed that supports control rod 18-27 drifting out of the core.)

   a. Insert listed malfunction.

   b. Place the Simulator in RUN.

   c. Notch control rod 18-27 in one notch.

   d. When control rod reaches previous even notch, activate TRIGGER E1 to stick the control rod

Note: If this JPM is to be performed more than once, snap the Simulator into an IC after the panel set-up is complete.

ATTACHMENT 2

Directions to Trainee:

When I tell you to begin, you are to perform actions as appropriate to panel 9-5 indications. Before you start, I will state the general plant conditions, the Initiating Cues, and answer any questions you may have.

If being simulated In-Plant or Control Room:

When simulating, physically point to any meters, gauges, recorders and controls you would be using. State the position of controls as you would have manipulated them in order to complete the assigned task.

If being performed in the Simulator:

During task performance, state the actions you are taking, e.g.: repositioning controls and observing instrumentation.

General Conditions:

1. The plant is operating at power.
2. You are the CRO - Reactor Operator (RO) (9-5 PNL).

Initiating Cue(s):

The Control Room Supervisor directs you to perform actions as appropriate to panel 9-5 indications.

Task No.: 241013O0401

==

Task Title: Transfer Governor Valve Control from Manual to Auto with DEH in Mode IV

==

Trainee: ____________________________ Examiner: _______________________________

Pass:___ Fail:____ Examiner Signature:___________________________Date:_________

Additional Program Information:

1. Appropriate Performance Locations: CR / SIM
2. Appropriate Trainee level: RO / SRO
3. Evaluation Method: Perform ___ Simulate ___
4. Performance Time: 10 minutes
5. NRC K/As 241000 A4.08 (3.5/3.4)

Directions to Examiner:

1. This JPM evaluates the trainee's ability to transfer governor valves from manual to automatic per procedure 2.2.77.1, "Digital Electro-Hydraulic (DEH) Control System."
2. If this JPM is performed on the Simulator, only the cues preceded by "#" should be given.
3. All blanks must be filled out with either initials or an "NP" for "not performed"; an explanation may also be written in the space, if desired, by the examiner.
4. Give the trainee his copy of the Directions to the Trainee (Attachment 2) when ready to start the JPM.
5. Brief the trainee, place the Simulator in RUN, and tell the trainee to begin.

Task No.: 241013O0401

==

Task Title: Transfer Governor Valve Control from Manual to Auto with DEH in Mode IV

==

Directions to Trainee:

When I tell you to begin, you are to transfer main turbine governor valves to automatic per 2.2.77.1, "Digital Electro-Hydraulic (DEH) Control System." Before you start, I will state the general plant conditions, the Initiating Cues, and answer any questions you may have.

<u>If being simulated In-Plant or Control Room:</u>

When simulating, physically point to any meters, gauges, recorders and controls you would be using. State the position of controls as you would have manipulated them in order to complete the assigned task.

<u>If being performed in the Simulator:</u>

During task performance, state the actions you are taking, e.g.: repositioning controls and observing instrumentation.

==

General Conditions:

1. The Reactor operating at power.
2. Last shift, DEH failed causing Governor valves to transfer to manual.
3. Bypass valves are in automatic.
4. DEH has been repaired.

General References:

1. Procedure 2.4DEH, DEH Abnormal.
2. Procedure 2.2.77.1, Digital Electro-Hydraulic (DEH) Control System.

Task No.: 241013O00401

==

Task Title: Transfer Governor Valve Control from Manual to Auto with DEH in Mode IV

==

General Tools and Equipment:

1. None

Special Conditions, References, Tools, Equipment:

1. Simulator Setup: See Attachment 1.
2. Critical steps denoted by "*".
3. Simulator cues denoted by "#".

Task Standards:

1. 100% of critical elements successfully completed without error.
2. 100% of safety and radiological work practices.

Initiating Cue(s):

The Control Room Supervisor directs you to return DEH to automatic control in MODE IV per procedure 2.2.77.1, "Digital Electro-Hydraulic (DEH) Control System." Inform the CRS when DEH is in automatic in MODE IV.

Task No.: 241013O0401

Task Title: Transfer Governor Valve Control from Manual to Auto with DEH in Mode IV

Performance Checklist	Standards	Initials
1. Obtain copy of 2.2.77.1.	The operator obtains a copy of 2.2.77.1.	______
2. Press OPER AUTO button.	The operator presses the OPER AUTO button. **CUE**: The OPER AUTO button has been depressed.	_______*
3. Check OPER AUTO button is on.	Operator checks OPER AUTO button is on. **CUE**: OPER AUTO button is on..	_______
4. Check GV AUTO button is on.	Operator checks GV AUTO button is on. **CUE**: GV AUTO button is on.	_______
5. Press LOAD RATE MW/MIN button and check it backlights.	Operator presses LOAD RATE MW/MIN button and check it backlights. **CUE**: The LOAD RATE MW/MIN button backlights.	_______
6. Check existing load rate appears in TURBINE DISPLAY window.	Operator checks existing load rate appears in TURBINE DISPLAY window. **CUE**: Existing load rate appears in TURBINE DISPLAY window. **#CUE**: If asked as CRS, existing load rate is acceptable.	_______

Task No.: 241013O0401

Task Title: Transfer Governor Valve Control from Manual to Auto with DEH in Mode IV

Performance Checklist	Standards	Initials
7. Press REF button and check it backlights.	Operator presses REF button and check it backlights. **CUE**: The REF button backlights.	_______
8. Check load setpoint is displayed in TURBINE REFERENCE and TURBINE DEMAND windows.	Operator checks load setpoint is displayed in TURBINE REFERENCE and TURBINE DEMAND windows. **CUE**: The load setpoint is displayed in TURBINE REFERENCE and TURBINE DEMAND windows.	_______
9. Press DEMAND RAISE button until 839 MWe is displayed in TURBINE DEMAND window.	Operator presses DEMAND RAISE button until 839 MWe is displayed in TURBINE DEMAND window. **CUE**: Initially: 800 is displayed in the TURBINE DEMAND WINDOW. After adjustment: 839 MWe is displayed.	_______*
10. Press ENTER button.	Operator presses ENTER button. **CUE**: The ENTER button has been depressed.	_______*

Task No.: 241013O0401

Task Title: Transfer Governor Valve Control from Manual to Auto with DEH in Mode IV

Performance Checklist	Standards	Initials
11. Check 839 MWe is indicated in TURBINE DEMAND window.	Operator checks 839 MWe is indicated in TURBINE DEMAND window. **CUE**: 839 MWe is indicated in TURBINE DEMAND window.	_______
12. Check old load setpoint remains in TURBINE REFERENCE window.	Operator checks old load setpoint remains in TURBINE REFERENCE window. **CUE**: The old load setpoint remains in TURBINE REFERENCE window.	_______
13. Check HOLD button backlights.	Operator checks HOLD button backlights. **CUE**: The HOLD button backlights.	_______
14. Press GO button.	Operator presses the GO button. **CUE**: The GO button has been depressed.	_______*
15. Check GO button backlights and HOLD button turns off.	Operator checks GO button backlights and HOLD button turns off. **CUE**: The GO button backlights and HOLD button turns off.	_______

Task No.: 241013O0401

Task Title: Transfer Governor Valve Control from Manual to Auto with DEH in Mode IV

Performance Checklist	Standards	Initials
16. Check Load setpoint in TURBINE REFERENCE window changes at requested rate.	Operator checks Load setpoint in TURBINE REFERENCE window changes at requested rate. **CUE**: The Load setpoint in TURBINE REFERENCE window changes at requested rate.	_______
17. When TURBINE REFERENCE is equal to TURBINE DEMAND, check GO button turns off.	Operator checks GO button turns off. **CUE**: The GO button turns off.	_______
18. Operator reports DEH is in MODE IV in automatic control.	Operator reports DEH is in MODE IV in automatic control. **#CUE**: The CRS acknowledges the report.	_______

Task No.: 241013O0401

Task Title: Transfer Governor Valve Control from Manual to Auto with DEH in Mode IV

ATTACHMENT 1

SIMULATOR SET-UP

A. Materials required

None

B. Initialize the Simulator in any IC that will support DEH operation in MODE IV. (IC-18, 19 or 20 suggested)

Batch File name - none.

C. Change the Simulator conditions from those of the IC as follows:

1. Triggers

Number	File Name	Description
	None	

2. Malfunctions

Number	Title	Trigger	TD	Severity	Ramp	Initial
	None					

3. Remotes

Number	Title	Trigger	TD	Value	Ramp
	None				

===

Task Title: Transfer Governor Valve Control from Manual to Auto with DEH in Mode IV

===

4. Overrides

Instrument	**Tag**	**Trigger**	**TD**	**Value**	**Ramp**
None					

5. Panel Set-up (suggested. Any setup is allowed that supports operation of DEH in MODE IV)

   a. Place the Simulator in RUN.

   b. Transfer main turbine governor valves to manual.

   c. Close the governor valves until bypass valves are open ~ 10% (this value may be increased as necessary to support other simultaneous JPMs.)

Note: If this JPM is to be performed more than once, snap the Simulator into an IC after the panel set-up is complete.

ATTACHMENT 2

Directions to Trainee:

When I tell you to begin, you are to transfer main turbine governor valves to automatic per 2.2.77.1, "Digital Electro-Hydraulic (DEH) Control System." Before you start, I will state the general plant conditions, the Initiating Cues, and answer any questions you may have.

<u>If being simulated In-Plant or Control Room:</u>

When simulating, physically point to any meters, gauges, recorders and controls you would be using. State the position of controls as you would have manipulated them in order to complete the assigned task.

<u>If being performed in the Simulator:</u>

During task performance, state the actions you are taking, e.g.: repositioning controls and observing instrumentation.

General Conditions:

1. The Reactor operating at power.
2. Last shift, DEH failed causing Governor valves to transfer to manual.
3. Bypass valves are in automatic.
4. DEH has been repaired.

Initiating Cue(s):

The Control Room Supervisor directs you to return DEH to automatic control in MODE IV per procedure 2.2.77.1, "Digital Electro-Hydraulic (DEH) Control System." Inform the CRS when DEH is in automatic in MODE IV.

Task No.: 202022C0401

==

Task Title: Respond to a Trip of a Reactor Recirc pump (Alternate Path)

==

Trainee: ____________________________ Examiner: _______________________________

Pass:___ Fail:____ Examiner Signature:___________________________Date:_________

ALTERNATE PATH

Additional Program Information:

1. Appropriate Performance Locations: CR / SIM
2. Appropriate Trainee level: RO / SRO
3. Evaluation Method: Perform ___ Simulate ___
4. Performance Time: 20 minutes
5. NRC K/As 202001 A2.03 (3.6/3.7)

Directions to Examiner:

NOTE: THIS IS A **ALTERNATE PATH** JPM. THE FLOW SUBTRACTING NETWORK WILL FAIL AND REQUIRE MANUAL INPUT OF TOTAL CORE FLOW.

1. This JPM evaluates the trainee's ability to respond to a Reactor Recirculation pump trip per 2.4RR, "Reactor Recirculation Abnormal."
2. If this JPM is performed on the Simulator, only the cues preceded by "#" should be given.
3. All blanks must be filled out with either initials or an "NP" for "not performed"; an explanation may also be written in the space, if desired, by the examiner.
4. Give the trainee his copy of the Directions to the Trainee (Attachment 2) when ready to start the JPM.
5. Brief the trainee, place the Simulator in RUN, and tell the trainee to begin.

Task No.: 202022C0401

==

Task Title: Respond to a Trip of a Reactor Recirc pump (Alternate Path)

==

Directions to Trainee:

When I tell you to begin, you are to perform actions as appropriate to panel 9-4 and 9-5 indications. Before you start, I will state the general plant conditions, the Initiating Cues, and answer any questions you may have.

<u>If being simulated In-Plant or Control Room:</u>

When simulating, physically point to any meters, gauges, recorders and controls you would be using. State the position of controls as you would have manipulated them in order to complete the assigned task.

<u>If being performed in the Simulator:</u>

During task performance, state the actions you are taking, e.g.: repositioning controls and observing instrumentation.

==

General Conditions:

1. The plant is operating at power.
2. You are the CRO - Reactor Operator (RO) (9-5 PNL).

General References:

1. Procedure 2.4RR, Reactor Recirculation Abnormal
2. Procedure 2.2.68.1, Reactor Recirculation System Operations

Task No.: 202022C0401

==

Task Title: Respond to a Trip of a Reactor Recirc pump (Alternate Path)

==

General Tools and Equipment:

1. None

Special Conditions, References, Tools, Equipment:

1. Simulator Setup: See Attachment 1.
2. Critical steps denoted by "*".
3. Simulator cues denoted by "#".

Task Standards:

1. 100% of critical elements successfully completed without error.
2. 100% of safety and radiological work practices.

Initiating Cue(s):

The Control Room Supervisor directs you to perform actions as appropriate to panel 9-4 and panel 9-5 indications.

Task No.: 202022C0401

Task Title: Respond to a Trip of a Reactor Recirc pump (Alternate Path)

Performance Checklist	Standards	Initials
1. Assume the watch at panel 9-5.	The operator positions himself in a position to monitor panel 9-5.	_______
ACTION:	**After the candidate assumes the watch, activate TRIGGER E1 to trip "B" Reactor Recirculation Pump.**	
2. Recognize and report trip of "B" Reactor Recirculation pump.	The operator recognizes and reports the trip of "B" Reactor Recirculation pump. **#CUE**: Acknowledge the report as CRS.	_______
3. Take appropriate immediate actions.	The operator evaluates 2.4RR immediate actions; determines none apply.	_______
4. Obtain procedure 2.4RR.	The operator obtains a copy of 2.4RR.	_______
5. Enter Attachment 1 of 2.4RR	Operator enters Attachment 1 of 2.4RR.	_______
6. Evaluate need to enter Attachment 4 for stability exclusion region.	Operator evaluates the need to enter Attachment 4 for stability exclusion region; determines entry is not required. **CUE**: Indicate a value of core flow outside the stability exclusion region.	_______

Task No.: 202022C0401

Task Title: Respond to a Trip of a Reactor Recirc pump (Alternate Path)

Performance Checklist	Standards	Initials
7. Ensure RRMG Set B GEN FIELD BKR open.	Operator ensures RRMG Set B GEN FIELD BKR open. **CUE**: RRMG Set B GEN FIELD BKR green light is illuminated, the red light is out.	_______
8. Close RR-MO-53B, PUMP DISCHARGE VLV.	Operator closes RR-MO-53B, PUMP DISCHARGE VLV. **CUE**: RR-MO-53B green light is lit and the red light is out.	_______

NOTE: **The operator should continue with the remaining 2.4RR steps while waiting for the RR-MO-53B valve.**

Performance Checklist	Standards	Initials
9. After RR-MO-53B has been closed for 5 minutes, open valve.	Operator opens RR-MO-53B after it has has been closed for 5 minutes (-0, + 5 minutes). **CUE**: Inform operator that 5 minutes has elapsed since RR-MO-53B was closed. **CUE**: RR-MO-53B green light is out and red light is lit (after valve has been opened).	_______

Task No.: 202022C0401

Task Title: Respond to a Trip of a Reactor Recirc pump (Alternate Path)

Performance Checklist	Standards	Initials
10. Ensure operating RRMG is transferred to Startup Transformer per Procedure 2.2.18.	Operator ensures "A" RRMG is powered by the Startup Transformer. **CUE**: "A" RRMG is powered by the Startup Transformer (Breaker 1CS is closed).	_______
11. Maintain oil outlet temperature for tripped RRMG 90°F to 130°F.	Operator directs Station Operator to maintain oil outlet temperature for tripped RRMG 90°F to 130°F. **#CUE**: Acknowledge/repeat back order as Station Operator.	_______
12. Monitor loop cooldown rate on RR-TR-165, RR SUCTION & FEEDWATER TEMP.	Operator monitors loop cooldown rate on RR-TR-165, RR SUCTION & FEEDWATER TEMP. **CUE**: Loop temperature has dropped 6˚F over the last 5 minutes.	_______
13. Concurrently enter Single Loop Operation per Procedure 2.2.68.1.	Operator obtains a copy of 2.2.68.1.	_______
14. Dispatch Operators to R-976-W and Non-Critical Switchgear Room to record lockout relays and targets for tripped pump.	Operator dispatches Operators to R-976-W and Non-Critical Switchgear Room to record lockout relays and targets for tripped pump. **#CUE**: Overcurrent (51 relay) is tripped for breaker 1DN.	_______

Task No.: 202022C0401

Task Title: Respond to a Trip of a Reactor Recirc pump (Alternate Path)

Performance Checklist	Standards	Initials
15. If total core flow < 20%, concurrently enter Attachment 2.	Operator determines Core Flow is above 20% rated. **CUE**: Indicated core flow is 49 Mlbm/hr.	_______
16. Align RRMG H&V System per Procedure 2.2.85.	Operator align RRMG H&V System per Procedure 2.2.85. **#CUE**: When operator initiates action to align ventilation, inform him that another operator will perform the ventilation alignment.	_______
17. Raise core flow to > 29.5×10^6 lbs/hr, if possible.	Operator determines core flow is > 29.5×10^6 lbs/hr **CUE**: Indicated core flow is 49 Mlbm/hr.	_______
18. Determine if reverse flow summer is functioning.	Operator determines reverse flow summer is NOT functioning as annunciator 9-4-3/E-7 is NOT in and indicated core flow is NOT approximately equal to difference between NBI-FI-92A and NBI-FI-92B **CUE**: 9-4-3/E-7 is ***NOT*** alarming. **CUE**: NBI-FI-92A reads 44 Mlbm/hr. NBI-FI-92B reads 5 Mlbm/hr.	_______*

Task No.: 202022C0401

Task Title: Respond to a Trip of a Reactor Recirc pump (Alternate Path)

Performance Checklist	Standards	Initials
19. Initiate an emergency Work Order to repair or replace reverse flow summer (NBI-SUM-97).	Operator initiate an emergency Work Order to repair or replace reverse flow summer (NBI-SUM-97). **#CUE**: Another operator will perform this task.	_______
20. Determine difference between NBI-FI-92A and NBI-FI-92B loop flows.	Operator determines difference between NBI-FI-92A and NBI-FI-92B loop flows is $\geq 24x10^6$ lbs/hr,	_______*

NOTE: **Entering substitute value for point B012 is accomplished using an IDT terminal and pressing the "Bogey Value" button. On screen prompts will guide the rest of the substitution.**

Performance Checklist	Standards	Initials
21. Enter substitute value for PMIS Point B012.	Operator enters difference between NBI-FI-92A and NBI-FI-92B loop flows as substitute for PMIS Point B012. **CUE**: PMIS Point B012 has been substituted for. **#CUE**: Inform candidate that this JPM is complete.	_______*

ATTACHMENT 1

SIMULATOR SET-UP

A. Materials required

None

B. Initialize the Simulator in any full power IC (IC-18, 19 or 20 suggested)

Batch File name - none.

C. Change the Simulator conditions from those of the IC as follows:

1. Triggers

Number	File Name	Description
E11	None	trgset 11 "zlorrmgfbb[1]==1" "B" Recirc MG Field Breaker green light on

2. Malfunctions

Number	Title	Trigger	TD	Severity	Ramp	Initial
RR04D	"B" Reactor Recirculation Pump Drive Motor Breaker Trip (1DN)	E1	N/A	N/A	N/A	N/A

3. Remotes

Number	Title	Trigger	TD	Value	Ramp
	None				

Task Title: Respond to a Trip of a Reactor Recirc pump (Alternate Path)

4. Overrides

Instrument	Tag	Trigger	TD	Value	Ramp
9-4-3/E-7, RECIRC LOOP B OUT OF SERVICE	RA:MUX08C108	A	0	OFF	0
Total Core Flow	ZAONBIDPRFR95[2]	E11	0	49	0

5. Panel Set-up (suggested.)

   a. Insert listed overrides and malfunctions.

   b. Place the Simulator in RUN.

   c. Ensure “A” Reactor Recirculation pump is aligned to the Startup Transformer.

   b. Ensure “B” Reactor Recirculation pump is aligned to the Normal Transformer.

Note: If this JPM is to be performed more than once, snap the Simulator into an IC after the panel set-up is complete.

ATTACHMENT 2

Directions to Trainee:

When I tell you to begin, you are to perform actions as appropriate to panel 9-4 and 9-5 indications. Before you start, I will state the general plant conditions, the Initiating Cues, and answer any questions you may have.

<u>If being simulated In-Plant or Control Room:</u>

When simulating, physically point to any meters, gauges, recorders and controls you would be using. State the position of controls as you would have manipulated them in order to complete the assigned task.

<u>If being performed in the Simulator:</u>

During task performance, state the actions you are taking, e.g.: repositioning controls and observing instrumentation.

General Conditions:

1. The plant is operating at power.
2. You are the CRO - Reactor Operator (RO) (9-5 PNL).

Initiating Cue(s):

The Control Room Supervisor directs you to perform actions as appropriate to panel 9-4 and panel 9-5 indications.

Task No.: 200043C0401

==

Task Title: Vent Primary Containment per 2.4PC

==

Trainee: ____________________________ Examiner: _______________________________

Pass:___ Fail:____ Examiner Signature:___________________________Date:_________

Additional Program Information:

1. Appropriate Performance Locations: CR / SIM
2. Appropriate Trainee level: RO / SRO
3. Evaluation Method: Perform ___ Simulate ___
4. Performance Time: 15 minutes
5. NRC K/As 223001 A2.07 (4.2/4.3)

Directions to Examiner:

1. This JPM evaluates the trainee's ability to vent the primary containment per the guidance of procedure 2.4PC, "Primary Containment Control" and 2.2.60, "Primary Containment Cooling and Nitrogen Inerting System."
2. If this JPM is performed on the Simulator, only the cues preceded by "#" should be given.
3. All blanks must be filled out with either initials or an "NP" for "not performed"; an explanation may also be written in the space, if desired, by the examiner.
4. Give the trainee his copy of the Directions to the Trainee (Attachment 2) when ready to start the JPM.
5. Brief the trainee, place the Simulator in RUN, and tell the trainee to begin.

Task No.: 200043C0401

==

Task Title: Vent Primary Containment per 2.4PC

==

Directions to Trainee:

When I tell you to begin, you are to maintain Drywell pressure by venting the primary containment. Before you start, I will state the general plant conditions, the Initiating Cues, and answer any questions you may have.

If being simulated In-Plant or Control Room:

When simulating, physically point to any meters, gauges, recorders and controls you would be using. State the position of controls as you would have manipulated them in order to complete the assigned task.

If being performed in the Simulator:

During task performance, state the actions you are taking, e.g.: repositioning controls and observing instrumentation.

==

General Conditions:

1. Primary containment pressure has risen over the last shift.
2. The crew has begun implementing procedure 2.4PC, Primary Containment Control.

General References:

1. Procedure 2.4PC Primary Containment Control
2. Procedure 2.2.60 Primary Containment Cooling and Nitrogen Inerting System

Task No.: 200043C0401

Task Title: Vent Primary Containment per 2.4PC

General Tools and Equipment:

1. None

Special Conditions, References, Tools, Equipment:

1. Simulator Setup: See Attachment 1.
2. Critical steps denoted by "*".
3. Simulator cues denoted by "#".

Task Standards:

1. 100% of critical elements successfully completed without error.
2. 100% of safety and radiological work practices.

Initiating Cue(s):

The Control Room Supervisor directs you to maintain Drywell pressure per 2.4PC. Inform the CRS when drywell pressure is lowering.

Task No.: 200043C0401

Task Title: Vent Primary Containment per 2.4PC

Performance Checklist	Standards	Initials
1. Obtain copy of 2.4PC.	The operator obtains a copy of 2.4PC.	______
2. Inform crew of required scram action.	The operator informs the CRS/crew that a scram is required if drywell pressure cannot be maintained below 1.5 psig. **#CUE**: Acknowledge information.	______
3. Determine 2.4PC requirements for maintenance of Drywell pressure.	The operator determines he must vent containment through SGT System per Procedure 2.2.60 to maintain drywell pressure between 0.25 psig and 0.45 psig.	______
4. Obtain a copy of 2.2.60.	Operator obtains a copy of 2.2.60 or hard card.	______

NOTE: **The following step is not on the hard card for containment venting. If the hard card is used, the following step is N/A.**

Performance Checklist	Standards	Initials
5. Ensure lineup verification requirements are met.	The operator ensure lineup verification requirements of ODAM DSR 3.2.7.1 are completed **#CUE**: Report requirements of ODAM DSR 3.2.7.1 are completed.	______

Task No.: 200043C0401

Task Title: Vent Primary Containment per 2.4PC

Performance Checklist	Standards	Initials
6. Ensure DAMPER AD-R-1A AND AD-R-1B CONTROL switch is in SGT	The operator ensures DAMPER AD-R-1A AND AD-R-1B CONTROL switch is in SGT. **CUE**: DAMPER AD-R-1A AND AD-R-1B CONTROL switch is in SGT.	_______
7. Check AD-R-1A is closed	The operator checks AD-R-1A is closed. **CUE**: AD-R-1A green light is lit, red light is out.	_______
8. Check AD-R-1B is open.	The operator **CUE**: AD-R-1B green light is out, red light is lit.	_______
9. Start preferred SGT fan, EF-R-1E(1F), SGT A(B) EXHAUST FAN	The operator starts preferred SGT fan, EF-R-1E(1F). **CUE**: The preferred SGT fan, EF-R-1E(1F) red light is lit, green light is out.	_______*
10. Open DPCV associated with running SGT fan, SGT-DPCV-546A(B), SGT A(B) FLOW/RX BLDG DP CONT.	The operator opens DPCV associated with running SGT fan, SGT-DPCV-546A(B) **CUE**: SGT-DPCV-546A(B) green light is out, red light is lit.	_______*

NOTE **Steps to vent the drywell and torus may be performed in any order or concurrently, depending on plant conditions.**

Task No.: 200043C0401

Task Title: Vent Primary Containment per 2.4PC

Performance Checklist	Standards	Initials
11. Open PC-AO-246, DW EXH OUTBD ISOL VLV	The operator opens PC-AO-246. **CUE**: PC-AO-246 green light is out, red light is lit.	_______*
12. Open PC-MO-306, VALVE MO 231 BYPASS VLV.	While ensuring Torus pressure does <u>not</u> exceed Drywell pressure by > 0.1 psig, the operator opens PC-MO-306 **CUE**: PC-MO-306 green light is out, red light is lit.	_______*
13. At VBD-P2, ensure PC-MO-1308, TORUS VENT ISOLATION VLV, is closed	The operator ensures PC-MO-1308, TORUS VENT ISOLATION VLV, is closed **CUE**: PC-MO-1308 green light is lit, red light is out.	_______
14. At VBD-H, open PC-AO-245, TORUS EXH OUTBD ISOL	The operator opens PC-AO-245. **CUE**: PC-AO-245 green light is out, red light is lit.	_______*
15. At VBD-H, open PC-MO-305, VALVE MO 230 BYPASS VLV.	The operator opens PC-MO-305. **CUE**: PC-MO-305 green light is out, red light is lit.	_______*
16. Operator reports drywell pressure is lowering.	Operator reports drywell pressure is lowering. **#CUE**: The CRS acknowledges the report.	_______

==

Task Title: Vent Primary Containment per 2.4PC

==

Task No.: 200043C0401

Task Title: Vent Primary Containment per 2.4PC

ATTACHMENT 1

SIMULATOR SET-UP

A. Materials required

None

B. Initialize the Simulator in any IC that will support venting the primary containment (IC-18, 19 or 20 suggested)

Batch File name - none.

C. Change the Simulator conditions from those of the IC as follows:

1. Triggers

Number	File Name	Description
	None	

2. Malfunctions

Number	Title	Trigger	TD	Severity	Ramp	Initial
RR20A	Coolant Leakage Inside Primary Containment	A	0	4	0	0

3. Remotes

Number	Title	Trigger	TD	Value	Ramp
	None				

Task No.: 200043C0401

Task Title: Vent Primary Containment per 2.4PC

4. Overrides

Instrument	**Tag**	**Trigger**	**TD**	**Value**	**Ramp**
None					

5. Panel Set-up (suggested. Any setup is allowed that supports venting the primary containment per 2.4PC and 2.2.60)

   a. Place the Simulator in RUN.

   b. Insert malfunctions as listed.

   c. Adjust RR20A as necessary to obtain and maintain drywell pressure ~ 0.7 to 1.0 psig.

Note: If this JPM is to be performed more than once, snap the Simulator into an IC after the panel set-up is complete.

ATTACHMENT 2

Directions to Trainee:

When I tell you to begin, you are to maintain Drywell pressure by venting the primary containment. Before you start, I will state the general plant conditions, the Initiating Cues, and answer any questions you may have.

If being simulated In-Plant or Control Room:

When simulating, physically point to any meters, gauges, recorders and controls you would be using. State the position of controls as you would have manipulated them in order to complete the assigned task.

If being performed in the Simulator:

During task performance, state the actions you are taking, e.g.: repositioning controls and observing instrumentation.

General Conditions:

1. Primary containment pressure has risen over the last shift.
2. The crew has begun implementing procedure 2.4PC, Primary Containment Control.

Initiating Cue(s):

The Control Room Supervisor directs you to maintain Drywell pressure per 2.4PC. Inform the CRS when drywell pressure is lowering.

Task No.: 259058G401

==

Task Title: Perform a Quick Restart of RFPT B (Hard Card) (Alternate Path)

==

Trainee: ______________________________ Examiner: ___________________________

Pass:___ Fail:___ Examiner Signature: _________________________ Date: _________

ALTERNATE PATH

Additional Program Information:

1. Appropriate Performance Locations: CR / SIM
2. Appropriate Trainee level: RO / SRO
3. Evaluation Method: __ Simulate __ Perform
4. Performance Time: 15 minutes
5. NRC K/A 259001 A4.02 (3.9/3.7)

Directions to Examiner:

NOTE: THIS IS AN **ALTERNATE PATH** JPM. THE INITIAL ATTEMPT TO RESET THE RFPT TRIP WILL FAIL AND REQUIRE USE OF THE ALTERNATIVE METHOD TO RESET THE TRIP.

1. This JPM evaluates the trainee's ability to perform a quick start of a RFPT
2. If this JPM is performed on the Simulator, only the cues preceded by "#" should be given.
3. All blanks must be filled out with either initials or an “NP” for “not performed”; an explanation may also be written in the space if desired by the examiner.
4. ***DO NOT place the simulator in RUN until the student is ready to perform the JPM. (RFP must be coasting down or on turning gear < 5 minutes to use this procedure section.)***
5. Give the trainee his copy of the Directions to the Trainee (Attachment 2) when ready to start the JPM.
6. Brief the trainee, place the simulator in run, and tell the trainee to begin.

Task No.: 259058G401

==

Task Title: Perform a Quick Restart of RFPT B (Hard Card) (Alternate Path)

==

Directions to Trainee:

When I tell you to begin, you are to perform a quick start of the “B” RFPT. Before you start, I will state the general plant conditions, the Initiating Cues and answer any questions you may have.

If being simulated In-Plant or Control Room:

When simulating, physically point to any meters, gauges, recorders and controls you would be using. State the position of controls as you would have manipulated them in order to complete the assigned task.

If being performed in the Simulator:

During task performance, state the actions you are taking, e.g.: repositioning controls and observing instrumentation.

==

General Conditions:

1. The Reactor has scrammed.
2. Both RFPs have tripped on high level.

General References:

1. Procedure 2.2.28
8. Procedure 2.2.28.1

General Tools and Equipment:

1. None

Special Conditions, References, Tools, Equipment:

1. Simulator Setup: See Attachment 1.
2. Critical checks denoted by "*".
3. Simulator cues denoted by "#".

Task No.: 259058G401

==

Task Title: Perform a Quick Restart of RFPT B (Hard Card) (Alternate Path)

==

Task Standards:

1. 100% of critical elements successfully completed without error.
2. 100% of safety and radiological work practices.

Initiating Cue(s):

The Control Room Supervisor directs you to restart “B” RFP and raise RPV water level using the quick start hard card and procedure 2.2.28.1.

Task No.: 259058G401

==

Task Title: Perform a Quick Restart of RFPT B (Hard Card) (Alternate Path)

==

Performance Checklist	Standards	Initials
1. Ensure RFPT time limit is met.	Ensure RFPT coasting from trip or not on turning gear > 5 minutes after trip. **CUE**: RFPTs are still coasting down.	__________
2. Ensure RFPT trips are reset.	Ensure all RFPT trips (except high water level) are reset. **CUE**: All trips (except high RPV water level) are reset.	__________
3. Ensure High RPV water level RFPT trips are reset.	At Panel 9-5, reset at least 2 of the HIGH WATER LEVEL TRIPS. **CUE**: All 3 High RPV water level trips were in and now have been reset.	__________*
4. Ensure RFP-CS-RFPTB RFPT B Control Station is in MDVP	Operator ensures RFP-CS-RFPTB is in MDVP. **CUE**: RFP-CS-RFPTB is in MDVP	__________
5. Ensure OUTPUT on RFP-CS-RFPTB is adjusted to minimum.	Operator ensures OUTPUT on RFP-CS-RFPTB is adjusted to minimum. **CUE**: OUTPUT on RFP-CS-RFPTB is adjusted to minimum.	__________

Task No.: 259058G401

==

Task Title: Perform a Quick Restart of RFPT B (Hard Card) (Alternate Path)

==

Performance Checklist	Standards	Initials
6. Ensure RF-11B is OPEN	Ensure RF-FCV-11B, MIN FLOW VALVE, is open. **CUE**: RF-FCV-11B Red light is **ON** and Green light **OFF.**	_________
7. Attempt to reset B RFPT	Press and hold RFPT B TRIP RESET button. Note the RFPT B HP and LP STOP valves are NOT open. **CUE**: HP and LP STOP valves red light are **OFF** Green lights are **ON**.	_________
8. Check the Reset light.	Check light above RFPT B TRIP RESET button is on. **CUE**: The light is **ON**.	_________
NOTE:	**The operator may not perform the next step as the hard card does not specify as to how to reset the trip.**	
9. Recognize and report the trip reset failure.	Recognize and report the B RFPT trip reset failure. **#CUE**: As CRS, direct the operator to continue with the reset of B RFPT per procedure 2.2.28.	_________

Task No.: 259058G401

==

Task Title: Perform a Quick Restart of RFPT B (Hard Card) (Alternate Path)

==

Performance Checklist	Standards	Initials
10. Reset B RFPT	Press and hold RFPT B OVERSPEED TRIP BLOCK and RFPT B OVERSPEED TRIP RESET. **CUE**: HP and LP STOP valves red light are **ON** Green lights are **OFF**.	_________*
11. Check the Reset light	Check light above RFPT B TRIP RESET button is on. **CUE**: The trip reset light is **ON**. **#CUE**: As CRS, direct resumption of hard card use.	_________
12. Raise "B" RFPT speed.	Raise RFPT B speed by adjusting output on RFC-CS-RFPTB. **CUE**: Speed is 520 RPM	_________
13. Transfer control mode.	When RFP speed is > 500 rpm, place RFC-CS-RFPTB in MDEM. **CUE**: RFC-CS-RFPTB is in MDEM.	_________*
14. Raise RFP speed.	Raise speed of "B" RFP using UP/DOWN arrows on RFC-CS-RFPTB until the RFP is feeding the RPV. **CUE**: "B" RFP flow is rising.	_________*

Task No.: 259058G401

==

Task Title: Perform a Quick Restart of RFPT B (Hard Card) (Alternate Path)

==

Performance Checklist	Standards	Initials
15. Inform the CRS that the task is Complete.	Inform the Control Room Supervisor that 1B RFP has been restarted and injecting into the RPV #**CUE**: CRS acknowledges the report.	_________

Task No.: 259058G401

==

Task Title: Perform a Quick Restart of RFPT B (Hard Card) (Alternate Path)

==

ATTACHMENT 1

SIMULATOR SET-UP

A. Materials Required

None

B. Initialize the Simulator to any full power IC.

Batch File Name - none.

C. Change the Simulator conditions from those of the IC as follows:

1. Triggers

Number	**File Name**	**Description**
E2	None	Default to False

2. Malfunctions

Number	**Title**	**Trigger**	**TD**	**Severity**	**Ramp**	**Initial**
RR19	Shutdown Cooling Loss of Mass	E2	0	100	0	N/A

3. Remotes

Number	**Title**	**Trigger**	**Value**	**Ramp**
	None			

Task Title: Perform a Quick Restart of RFPT B (Hard Card) (Alternate Path)

4. Overrides

Instrument	**Tag**	**Trigger**	**TD**	**Value**	**Ramp**
RFPT-B Trip Reset Pushbutton	ZDIRFSWRFTRB[1]	A	0	OFF	0

5. Panel Setup (recommended. Any setup may be used that supports quick restart of "B" Reactor Feedwater Pump.)

   a. Place the Simulator in run.

   b. Place place RFC-CS-RFPTA in MDEM.

   c. Place the Mode Switch to Shutdown.

   d. Secure two Condensate and Condensate Booster pumps.

   e. After both RFPTs trip on high RPV water level, insert malfunction RR19 (Shutdown Cooling Loss of Mass) at 100% (Trigger E2).

   f. Ensure the "B" RFP is still coasting down or has just gone on the turning gear.

   g. Ensure RPV water level is < +50" (narrow range).

   h. Insert listed switch override.

   h. Place the Simulator in FREEZE until the operator is ready to begin.

Note: If this JPM is to be performed more than once, snap the simulator into an IC after the panel setup is complete.

ATTACHMENT 2

Directions to Trainee:

When I tell you to begin, you are to perform a quick start of the "B" RFPT. Before you start, I will state the general plant conditions, the Initiating Cues and answer any questions you may have.

<u>If being simulated In-Plant or Control Room:</u>

When simulating, physically point to any meters, gauges, recorders and controls you would be using. State the position of controls as you would have manipulated them in order to complete the assigned task.

<u>If being performed in the Simulator:</u>

During task performance, state the actions you are taking, e.g.: repositioning controls and observing instrumentation.

General Conditions:

1. The Reactor has scrammed.
2. Both RFPs have tripped on high level.

Initiating Cues:

The Control Room Supervisor directs you to restart "B" RFP and raise RPV water level using the quick start hard card and procedure 2.2.28.1.

Task No.: 245056G0101

==

Task Title: Transfer of 4160V Bus 1G from DG2 to Emergency Transformer

==

Trainee: ______________________________ Examiner: __________________________

Pass:____ Fail:____ Examiner Signature: _________________________ Date: ________

Additional Program Information:

1. Appropriate Performance Locations: CR / SIM
2. Appropriate Trainee level: RO / SRO
3. Evaluation Method: __ Simulate __ Perform
4. Performance Time: 20 minutes
5. NRC K/A:262001 A4.04 (3.6/3.7)

Directions to Examiner:

1. This JPM evaluates the trainee's ability to perform transfer of 4160 bus 1G from DG2 to the Emergency Transformer.
2. If this JPM is performed on the Simulator, only the cues preceded by “#” should be given.
3. All blanks must be filled out with either initials or an “NP” for “not performed”; an explanation may also be written in the space, if desired, by the examiner.
4. Give the trainee his copy of the Directions to the Trainee (Attachment 2) when ready to start the JPM.
5. Brief the trainee, place the Simulator in RUN, and tell the trainee to begin.

Task No.: 245056G0101

===

Task Title: Transfer of 4160V Bus 1G from DG2 to Emergency Transformer

===

Directions to Trainee:

When I tell you to begin, you are to perform the required actions to transfer 4160V bus 1G from DG2 to the Emergency Transformer. Before you start, I will state the general plant conditions, the Initiating Cues, and answer any questions you may have.

<u>If being simulated In-Plant or Control Room:</u>

When simulating, physically point to any meters, gauges, recorders and controls you would be using. State the position of controls as you would have manipulated them in order to complete the assigned task.

<u>If being performed in the Simulator:</u>

During task performance, state the actions you are taking, e.g.: repositioning controls and observing instrumentation.

==

General Conditions:

1. DG2 is supplying 4160 VAC bus 1G.
2. The Emergency Transformer is available.
3. Breaker 1FS has failed.

General References:

1. Procedure 2.2.18, 4160V Auxiliary Power Distribution System
2. Procedure 2.2.20.1, Diesel Generator Operations

General Tools and Equipment:

1. None

Special Conditions, References, Tools, Equipment:

Task No.: 245056G0101

==

Task Title: Transfer of 4160V Bus 1G from DG2 to Emergency Transformer

==

1. Simulator Setup: See Attachment 1.
2. Critical steps denoted by "*".
3. Simulator cues denoted by "#".

Task Standards:

1. 100% of critical elements successfully completed without error.
2. 100% of safety and radiological work practices.

Initiating Cue(s):

The Control Room Supervisor directs you to perform transfer of 4160V bus 1G from DG2 to the Emergency Transformer. The diesel will remain running unloaded for I&C maintenance. Inform Control Room Supervisor when 4160V bus 1G is being carried by the Emergency Transformer.

Task No.: 245056G0101

==

Task Title: Transfer of 4160V Bus 1G from DG2 to Emergency Transformer

==

Performance Checklist	Standards	Initials
1. Obtain copy of 2.2.18.	Operator obtains a copy of procedure 2.2.18.	_________
2. Ensure risk has been assessed, per Procedure 0.49, for placing Breaker 1FS, EMERGENCY XFMR BKR, for Bus 1F in PULL-TO-LOCK	The operator determines that risk need not be assessed as breaker is already open due to a failure. #**CUE**: IF asked as CRS, direct the operator to "N/A" the step.	_________
3. Ensure the Emergency Transformer is energized	The operator verifies voltage available from the Emergency Transformer. **CUE**: Emergency Transformer secondary voltage is 4440VAC.	_________
4. Inform Shift Supervisor both off-site circuits are inoperable.	Inform Shift Supervisor to declare both off-site circuits inoperable and to enter the appropriate Condition and Required Action of LCO 3.8.1, AC Sources - Operating. **#CUE**: SS acknowledges the information.	_________
5. Verify 1GS racked in.	Ensure Breaker 1GS, EMERGENCY TRANSFORMER FEED TO 4160V BUS 1G, is racked in. **CUE**: 1GS Breaker RED light OFF, GREEN light ON. **#CUE**: IF asked, breaker 1GS is racked in.	_________

Task No.: 245056G0101

Task Title: Transfer of 4160V Bus 1G from DG2 to Emergency Transformer

Performance Checklist	Standards	Initials
6. Start signals clear	Ensure DG2 auto start signals clear. **CUE**: Drywell pressure is .5 pig and steady Reactor water level being controlled 15" to 40". **CUE**: 4160V Bus 1F/1G are energized	_________
7. Depress local RESET button	Direct SO to locally at DG2 Control Panel, press and release EMERGENCY to NORMAL RESET button #**CUE**: Station Operator reports EMERGENCY to NORMAL RESET button has been depressed and released	_________*
8. Place into Droop Parallel to Parallel.	Direct SO to locally at DG2 Control Panel, ensure DROOP PARALLEL switch is in PARALLEL. #**CUE**: Station Operator reports DROOP PARALLEL switch is in PARALLEL.	_________*
9. Speed control check	Adjust speed of DG2 by placing DIESEL GEN 2 GOVERNOR SWITCH to RAISE or LOWER to verify DG2 frequency control. **CUE**: DG2 frequency is 59.6 Hz **CUE**: DG2 frequency rises to 60.0 Hz	_________

Task No.: 245056G0101

Task Title: Transfer of 4160V Bus 1G from DG2 to Emergency Transformer

Performance Checklist	Standards	Initials
10. Voltage control check	Adjust voltage of DG2 by placing DIESEL GEN 2 VOLTAGE REGULATOR switch RAISE or LOWER to maintain DG2 voltage at 4160 volts. **CUE**: DG voltage is 4050 volts. **CUE**: DG voltage rises to 4160 volts.	________
11. Synch switch to 1GS	Place SYNCH SWITCH 1GS to 1GS. **CUE**: SYNCH SWITCH is in 1GS position	________ *
12. Adjust speed	Using DIESEL GEN 2 GOVERNOR switch, adjust engine speed so SYNCHROSCOPE is rotating very slowly in counter-clockwise (slow) direction. **CUE**: SYNCHROSCOPE is rotating very slowly in the clockwise (fast) direction. **CUE**: SYNCHROSCOPE is rotating very slowly in the counter-clockwise (slow) direction.	________

Task No.: 245056G0101

Task Title: Transfer of 4160V Bus 1G from DG2 to Emergency Transformer

Performance Checklist	Standards	Initials
13. Adjust voltage	Using DIESEL GEN 2 VOLTAGE REGULATOR switch, adjust Bus 1G voltage to slightly lower than EMERGENCY XFMR VOLTAGE **CUE**: (Before adjustment) DG2 Voltage reads 4160 volts. **CUE**: XFMR Transformer voltage is 4449 volts. **CUE**: (after adjustment) DG2 voltage reads 4430 volts.	_________ *
14. Close 1GS	When SYNCHROSCOPE is at 1 o'clock, close Breaker 1GS and check switch spring returns to NORMAL AFTER CLOSE (red flagged) **CUE**: Switch returns to center position (red flagged). 1GS Breaker , GREEN light is OFF, RED light is ON.	_________*
15. Adjust kVARS	Adjust DG2 kVARS so that they are slightly positive (~200 kVARS) using DIESEL GEN 2 VOLTAGE REGULATOR switch. (100 - 300 KVARs) **CUE**: DG2 kVARS reads 40 kVARS **CUE**: DG2 kVARS reads 200 kVARS	_________

Task No.: 245056G0101

==

Task Title: Transfer of 4160V Bus 1G from DG2 to Emergency Transformer

==

Performance Checklist	Standards	Initials
16. Bkr 1FS Pull-To-Lock.	Verify switch for 1FS in PULL-TO-LOCK (PTL) **CUE**: 1FS Breaker is still tagged	________
17. Lower DG2 Load	Reduce load on DG2 to 1000 KW using DIESEL GEN 2 GOVERNOR switch. **CUE**: DG2 load is 2200 KW **CUE**: DG2 load is 1000 KW	________
18. Place SYNCH SWITCH to OFF	Place SYNCH SWITCH 1GS to OFF **CUE**: SYNCH SWITCH is in OFF	________
19. DG2 Cooldown	After engine has cooled and cylinder exhaust temperature have dropped (~5 minutes), remove DG2 from services per 2.2.20.1. #**CUE**: Five minutes have elapsed. #**CUE**: Station Operator reports engine temperatures are cooling down.	________
20. Lower DG1 load (2.2.20.1)	Lower DG1 load to ≥ 400 kW and ≤ 1000 kW **CUE**: DG2 load is 600 kW #**CUE**: 15 minutes have elapsed.	________

Task No.: 245056G0101

==

Task Title: Transfer of 4160V Bus 1G from DG2 to Emergency Transformer

==

Performance Checklist	Standards	Initials
21. Lower to 400 KW	Lower DG2 load to 400 KW **CUE**: DG2 load is 400 kW	________
22. Lower kVARs	Reduce DG2 kVARs as low as possible. **CUE**: DG2 kVARs is 50 KVA	________
23. Open EG2	Open DIESEL GEN 2 BKR EG2 **CUE**: BKR EG2 is NORMAL AFTER TRIP (green flagged). Green light is ON. Red light is OFF.	________*
24. Inform CRS	Inform CRS that the Emergency Transformer is carrying 4160 V Bus 1G. **CUE**: CRS acknowledges. This JPM is complete. (Stop JPM at this point even if candidate continues.)	________

Task No.: 245056G0101

==

Task Title: Transfer of 4160V Bus 1G from DG2 to Emergency Transformer

==

ATTACHMENT 1

SIMULATOR SET-UP

A. Materials required

None

B. Initialize the Simulator in any IC that will support HPCI injection mode after a scram (IC-18, 19 or 20 suggested)

Batch File name - none.

C. Change the Simulator conditions from those of the IC as follows:

1. Triggers

Number	File Name	Description
	None	

2. Malfunctions

Number	Title	Trigger	TD	Severity	Ramp	Initial
ED05	Loss of Power (Startup Transformer)	E1	0	N/A	N/A	N/A
EG09	Main Generator Trip	E1	0	N/A	N/A	N/A

3. Remotes

Number	Title	Trigger	TD	Value	Ramp
	None				

Task No.: 245056G0101

==

Task Title: Transfer of 4160V Bus 1G from DG2 to Emergency Transformer

==

4. Overrides

Instrument	**Tag**	**Trigger**	**TD**	**Value**	**Ramp**
None					

5. Panel Set-up (suggested. Any setup is allowed that supports performance of the bus transfer)

a. Place Bkr 1FS in PTL and place Danger Tag on C/S

b. Place Bkr 1GS in PTL

c. Insert malfunctions (if desired)

d. Place Mode switch to Shutdown

e. Restore REC per 2.2.65.1 following loss of power, start 3rd REC Pump on Div 2 side (if necessary).

f. Place 2nd SW pump B in service

g. Place 'B' RHR loop in Suppression Pool Cooling using Div 2 power sources

h. Place Bkr 1GS to NORMAL AFTER TRIP

i. Stabilize Reactor pressure and level with HPCI and RCIC

j. At DG2 local panel, depress & release EMERGENCY to NORMAL RESET pushbutton.

k. At DG2 local panel, place DROOP PARALLEL switch in PARALLEL.

l. Open SW-36 & SW-37.

Note: If this JPM is to be performed more than once, snap the simulator into an IC after the panel setup is complete.

Task No.: 245056G0101

==

Task Title: Transfer of 4160V Bus 1G from DG2 to Emergency Transformer

==

ATTACHMENT 2

Directions to Trainee:

When I tell you to begin, you are to perform the required actions to transfer 4160V bus 1G from DG2 to the Emergency Transformer. Before you start, I will state the general plant conditions, the Initiating Cues, and answer any questions you may have.

If being simulated In-Plant or Control Room:

When simulating, physically point to any meters, gauges, recorders and controls you would be using. State the position of controls as you would have manipulated them in order to complete the assigned task.

If being performed in the Simulator:

During task performance, state the actions you are taking, e.g.: repositioning controls and observing instrumentation.

General Conditions:

1. DG2 is supplying 4160 VAC bus 1G.
2. The Emergency Transformer is available.
3. Breaker 1FS has failed.

Initiating Cues:

The Control Room Supervisor directs you to perform transfer of 4160V bus 1G from DG2 to the Emergency Transformer. The diesel will remain running unloaded for I&C maintenance. Inform Control Room Supervisor when 4160V bus 1G is being carried by the Emergency Transformer.

Task No.: 299051A0104

==

Task Title: Install EOP PTM 97-100

==

Trainee: ____________________________ Examiner: ____________________________

Pass:___ Fail:____ Examiner Signature:__________________________ Date:_________

Additional Program Information:

1. Appropriate Performance Locations: Plant
2. Appropriate Trainee Levels: SO / RO / SRO
3. Evaluation Method: ___ Simulate ___ Perform
4. Performance Time: 20 minutes
5. NRC K/A: 216000 A1.02(2.9*/3.1*) 295037 2.1.30 (3.9/3.4)

Directions to Examiner:

1. This JPM evaluates the trainee's ability to Defeat automatic opening of Outboard LPCI Injection Valve on low Reactor Pressure during ATWS using PTM #97 through #100 per 5.8.20.
2. **ENSURE THE OPERATOR COMPLIES WITH ALL ELECTRICAL SAFETY PRECAUTIONS.**
3. Brief the trainee and tell the trainee to begin.
4. All blanks must be filled out with either initials or an "NP" for "not performed", and an explanation may also be written in the space if desired by the examiner.

Directions to Trainee:

When I tell you to begin, you are to perform the steps necessary to defeat automatic opening of Outboard LPCI Injection Valve on low Reactor Pressure during ATWS. Before you start, I will state the general plant conditions, the initiating cues, and answer any questions you may have.

When simulating, physically point to any meters, gauges, recorders and controls you would be using. State the position of controls as you would have manipulated them in order to complete the assigned task.

==

Task No.: 299051A0104

==

Task Title: Install EOP PTM 97-100

==

General Conditions:

1. The plant has scrammed.
2. A high power ATWS event is in progress.
3. All injection into the RPV except for RCIC, CRD and boron injection must be stopped and prevented.

General References:

4. Emergency Support Procedure 5.8.20, EOP Plant Temporary Modifications

General Tools and Equipment:

1. Flat tipped screwdriver.
2. Key for Aux Relay Room (Grand Master will work)
3. Electrical tape.
4. Key for PTM Box in Aux Relay Room (Master Lock J423)
5. Flashlight.

Special Conditions, References, Tools, Equipment:

1. Critical checks denoted by "*".

Task Standards:

1. 100% of critical elements successfully completed without error.
2. 100% of safety and radiological work practices.

Initiating Cue(s):

The Reactor Operator has directed you to defeat automatic opening of Outboard LPCI Injection Valve on low Reactor Pressure during ATWS by installing EOP PTMs #97 through #100 per ESP 5.8.20. You are to inform the Reactor Operator when the EOP PTMs #97 through #100 have been installed.

Task No.: 299051A0104

Task Title: Install EOP PTM 97-100

Performance Checklist		**Standards**	**Initials**
NOTE:	**Procedure 5.8.20 is staged in the Relay Room.**		
NOTE:	**Key J423 is required to open the PTM Box in the Aux Relay room.**		
NOTE:	**The student must show you a flat-tipped screwdriver and electrical tape. These are in the PTM box, but a J423 key is required to open the box. Any flat-tipped screwdriver and electrical tape will work.**		
1.	Obtain flat-tipped screwdriver.	The operator obtains flat-tipped screwdriver and electrical tape.	_______*
2.	Obtain electrical tape.	The operator obtains electrical tape.	_______*
3.	Obtain jumper for EOP PTM 98.	The operator obtains jumper for EOP PTM 98.	_______*
4.	Obtain jumper for EOP PTM 100.	The operator obtains jumper for EOP PTM 100.	_______*
5.	Identify wire RH197-13 from GG-84 (BAY-2, PNL 9-32).	The operator correctly identifies wire RH197-13.	_______
6.	Install EOP PTM Number 97 by lifting wire RH197-13 from GG-84.	The operator loosens the screw and removes wire RH197-13 from GG-84. **CUE**: The lead is lifted.	_______*
7.	Insulate the lifted lead	The operator wraps tape around the end of the wire lifted. **CUE**: The end of the wire has electrical tape wrapped around it.	_______

Task No.: 299051A0104

Task Title: Install EOP PTM 97-100

Performance Checklist		Standards	Initials
8.	Hang PTM tag.	The operator hangs the PTM 97 tag on the lifted lead. **CUE**: The PTM 97 tag is attached to the lifted lead.	_______
9.	Identify terminals GG-85 and GG-86 in BAY-2 of PNL 9-32.	The operator correctly identifies terminals GG-85 and GG-86.	_______
10.	Loosen screws.	The operator loosens screws on terminals GG-85 and GG-86. **CUE**: The screws rotate counterclockwise and the wire ends are loose.	_______*
11.	Install jumper between terminals GG-85 and GG-86.	The operator installs jumper between terminals GG-85 and GG-86. **CUE**: A jumper is installed between terminals GG-85 and GG-86.	_______*
12.	Tighten screw.	The operator tightens the screws and removes wire RH197-13 from GG-84. **CUE**: The screw rotates clockwise ½ turn, then stops turning. The wire ends are tight.	_______*
13.	Hang PTM tag.	The operator hangs the PTM 98 tag on the jumper. **CUE**: The PTM 98 tag is attached to the jumper.	_______
14.	Identify wire RH22-13 from GG-84 (BAY-2, PNL 9-33).	The operator correctly identifies wire RH22-13 from GG-84.	_______

Task No.: 299051A0104

Task Title: Install EOP PTM 97-100

Performance Checklist		Standards	Initials
15.	Install EOP PTM Number 99 by lifting wire RH22-13 from GG-84.	The operator loosens the screw and removes wire RH22-13 from GG-84. **CUE**: The lead is lifted.	_______*
16.	Insulate the lifted lead	The operator wraps tape around the end of the wire lifted. **CUE**: The end of the wire has electrical tape wrapped around it.	_______
17.	Hang PTM tag.	The operator hangs the PTM 99 tag on the lifted lead. **CUE**: The PTM 99 tag is attached to the lifted lead.	_______
18.	Identify terminals GG-85 and GG-86 (BAY-2, PNL 9-33).	The operator correctly identifies terminals GG-85 and GG-86.	_______
19.	Loosen screws.	The operator loosens screws on terminals GG-85 and GG-86. **CUE**: The screws rotate counterclockwise and the wire ends are loose.	_______*
20.	Install jumper between Terminals GG-85 and GG-86.	The operator installs jumper between Terminals GG-85 and GG-86. **CUE**: A jumper is installed between Terminals GG-85 and GG-86.	_______*
21.	Tighten screw.	The operator tightens the screws. **CUE**: The screw rotates clockwise ½ turn, then stops turning. The wire ends are tight.	_______*

Task No.: 299051A0104

Task Title: Install EOP PTM 97-100

Performance Checklist		Standards	Initials
22.	Hang PTM tag.	The operator hangs the PTM 100 tag on the jumper. **CUE**: The PTM 100 tag is attached to the jumper	_______
23.	Inform RO (CRS) of status.	The operator informs the RO that EOP PTMs #97 through #100 have been installed **CUE**: The RO (CRS) acknowledges.	_______

ATTACHMENT 1

Directions to Trainee:

When I tell you to begin, you are to perform the steps necessary to defeat automatic opening of Outboard LPCI Injection Valve on low Reactor Pressure during ATWS. Before you start, I will state the general plant conditions, the initiating cues, and answer any questions you may have.

When simulating, physically point to any meters, gauges, recorders and controls you would be using. State the position of controls as you would have manipulated them in order to complete the assigned task.

General Conditions:

1. The plant has scrammed.
2. A high power ATWS event is in progress.
3. All injection into the RPV except for RCIC, CRD and boron injection must be stopped and prevented.

Initiating Cue(s):

The Reactor Operator has directed you to defeat automatic opening of Outboard LPCI Injection Valve on low Reactor Pressure during ATWS by installing EOP PTMs #97 through #100 per ESP 5.8.20. You are to inform the Reactor Operator when the EOP PTMs #97 through #100 have been installed.

Task No.: 286005I0102

==

Task Title: Operate the Diesel Fire Pump Manually (Alternate Path)

==

Trainee: ____________________________ Examiner: ____________________________

Pass:___ Fail:____ Examiner Signature:_________________________ Date:_________

ALTERNATE PATH

Additional Program Information:

1. Appropriate Performance Locations: Plant
2. Appropriate Trainee Level: SO / RO / SRO
3. Evaluation Method: __ Simulate __ Perform
4. Performance Time: 12 minutes
5. NRC K/A 286000 A2.06 3.1/3.2

Directions to Examiner:

NOTE: THIS IS AN **ALTERNATE PATH** JPM. THE FIRST METHOD OF STARTING THE DIESEL WILL BE UNSUCCESSFUL.

1. This JPM evaluates the trainee's ability operate the diesel fire pump manually.
2. All blanks must be filled out with either initials or an “NP” for “not performed”; an explanation may also be written in the space, if desired, by the examiner.
3. Give the trainee his copy of the Directions to the Trainee (Attachment 1) when ready to start the JPM.
4. Brief the trainee and tell the trainee to begin.

Directions to Trainee:

When I tell you to begin, you are to operate the diesel fire pump manually. Before you start, I will state the general plant conditions, the Initiating Cues and answer any questions you may have.

When simulating, physically point to any meters, gauges, recorders and controls you would be using. State the position of controls as you would have manipulated them in order to complete the assigned task.

===

Task No.: 286005I0102

===

Task Title: Operate the Diesel Fire Pump Manually (Alternate Path)

===

General Conditions:

1. The plant has experienced a fire.
2. The "C" fire pump is out of service.
3. The electric fire pump is unable to maintain system pressure.
4. The Diesel Fire Pump has failed to auto start.
5. The Diesel Fire Pump cannot be started at panel FA.

General References:

1. Procedure 2.2.30

General Tools and Equipment:

1. Master key for building access

Special Conditions, References, Tools, Equipment:

1. Critical checks denoted by "*."

Task Standards:

1. 100% of critical elements successfully completed without error.
2. 100% of safety and radiological work practices.

Initiating Cue(s):

The Control Room Supervisor directs you to start the diesel fire pump at FP-PNL-F2 in the diesel fire house **AND** check engine parameters per procedure 2.2.30. Notify the CRS when the task is complete.

Task No.: 286005I0102

Task Title: Operate the Diesel Fire Pump Manually (Alternate Path)

Performance Checklist	Standards	Initials
1. Check the engine oil level.	Check the engine oil level is normal. **CUE:** Engine oil level is at normal.	______
2. Check the engine cooling reservoir.	Check the engine cooling water reservoir is full. **CUE:** The water level is at the top of the reservoir.	______
3. Position selector switch.	Position the selector switch on the local control cabinet in the MANUAL 1 or MANUAL 2 position. **CUE:** (As selected) switch is positioned to Manual 1 or Manual 2.	______
4. Start the pump.	Depress the start button. **CUE:** Sound of engine cranking sluggishly.	______
5. Place selector switch in opposite position.	Position the selector switch to the OPPOSITE manual position. **CUE:** (As selected) switch is positioned to Manual 2 or Manual 1.	______*

Task No.: 286005I0102

Task Title: Operate the Diesel Fire Pump Manually (Alternate Path)

Performance Checklist	Standards	Initials
6. Depress the START pushbutton.	Depress the start button. **CUE**: Sound of engine cranking briskly. **CUE**: Sound of engine running. **CUE**: (When checked) state that the diesel engine has warmed up.	_______*
7. Check circ water temperature.	Check circ water temperature is between 165°F and 195°F. **CUE**: Temperature reads 170°F.	_______
8. Check oil pressure.	Check oil pressure is between 30 and 85 psig. **CUE**: Pressure reads 80 psig.	_______
9. Check RPM.	Check engine RPM is between 1740 and 1840. **CUE**: RPM reading is 1750.	_______
10. Inform the Control Room Supervisor of completion.	Notify the CRS that the diesel fire pump has started and the engine parameters have been checked. **CUE**: The CRS acknowledges the report.	_______

ATTACHMENT 1

When I tell you to begin, you are to operate the diesel fire pump manually. Before you start, I will state the general plant conditions, the Initiating Cues and answer any questions you may have.

When simulating, physically point to any meters, gauges, recorders and controls you would be using. State the position of controls as you would have manipulated them in order to complete the assigned task.

General Conditions:

1. The plant has experienced a fire.
2. The "C" fire pump is out of service.
3. The electric fire pump is unable to maintain system pressure.
4. The Diesel Fire Pump has failed to auto start.
5. The Diesel Fire Pump cannot be started at panel FA.

Initiating Cues:

The Control Room Supervisor directs you to start the diesel fire pump at FP-PNL-F2 in the diesel fire house **AND** check engine parameters per procedure 2.2.30. Notify the CRS when the task is complete.

Task No.: 200078A0504

===

Task Title: Conduct Manual Draining of the SDV (Alternate Rod Insertion)

===

Trainee: ____________________________ Examiner: ____________________________

Pass:____ Fail:_____ Examiner Signature:__________________________ Date:_________

Additional Program Information:

1. Appropriate Performance Locations: Plant
2. Appropriate Trainee Level: SO / RO / SRO
3. Evaluation Method **Simulate**
4. Performance Time: 30 minutes
5. NRC K/As: 201001 A2.02 (3.2/3.3), 295037 EA1.05 (3.9/4.0), 295037 2.1.30 (3.9/3.4)

Directions to Examiner:

1. This JPM evaluates the trainee's ability to conduct manual draining of the SDV.
2. All blanks must be filled out with either initials or an “NP” for “not performed”; an explanation may also be written in the space, if desired, by the examiner.
3. Give the trainee his copy of the Directions to the Trainee (Attachment 1) when ready to start the JPM.
4. Brief the trainee and tell the trainee to begin.

Directions to Trainee:

When I tell you to begin, you are to manually drain the south Scram Discharge Volume. Before you start, I will state the general plant conditions, the initiating cues, and answer any questions you may have.

When simulating, physically point to any meters, gauges, recorders and controls you would be using. State the position of controls as you would have manipulated them in order to complete the assigned task.

===

Task No.: 200078A0504

Task Title: Conduct Manual Draining of the SDV (Alternate Rod Insertion)

General Conditions:

1. The plant scrammed; however, all rods have not been inserted to or beyond 02.
2. The scram has been Reset.
3. The SDV vent and drain valves remain closed and cannot be reopened.
4. EOP PTMs #31, 32, 33, and 34 have been installed.
5. TSC is not yet operational.
6. No ARMs are alarming.
7. The In-Containment Rad Monitors are reading 100 Rem/hr.
8. An operator has been dispatched to drain the North SDV.

General References:

1. Emergency Operating Procedure 5.8.3

General Tools and Equipment:

1. Special (yellow) wrench hanging by lanyard above drain valves.

Special Conditions, References, Tools, Equipment:

1. Critical checks denoted by "*".

Task Standards:

1. 100% of critical elements successfully completed without error.
2. 100% of safety and radiological work practices.

Initiating Cue(s):

The Control Room Supervisor directs you to manually drain the south scram discharge volume per Emergency Operating Procedure 5.8.3 and to secure the south SDV drain lineup when draining is complete. You are to inform the CRS when the task is complete.

Task No.: 200078A0504

Task Title: Conduct Manual Draining of the SDV (Alternate Rod Insertion)

Performance Checklist	Standard	Initials
1. Unlock CRD-AOV-CV35	Loosen the locknut below the handwheel of CRD-AOV-CV35 with the special wrench. **CUE**: Locknut turns freely.	___________*
2. OPEN CRD-AOV-CV35	Turn the hand-wheel counter-clockwise until CRD-AOV-CV35 is full OPEN. **CUE**: (First) The handwheel has engaged the upper nut. **CUE**: (Then) The handwheel stops turning. The stem indicator is at the top of the scale. The upper limit switch is made up.	___________*
3. Unlock CRD-AOV-CV33	Loosen the locknut below the hand-wheel of CRD-AOV-CV33 with the special wrench. **CUE**: Locknut turns freely.	___________*
4. OPEN CRD-AOV-CV33	Turn the hand-wheel counter-clockwise until CRD-AOV-CV33 is full open. **CUE**: (First) The handwheel has engaged the upper nut. **CUE**: (Then) The handwheel stops turning. The stem indicator is at the top of the scale. The upper limit switch is made up.	___________*

Task No.: 200078A0504

Task Title: Conduct Manual Draining of the SDV (Alternate Rod Insertion)

Performance Checklist	Standard	Initials
5. OPEN CRD-V-279	Open CRD-V-279, SDV VENT ROOT VALVE, (East end of SDV above HCUs). **CUE**: The handwheel is fully counter clockwise and the valve stem is out.	__________*
6. Vent the Scram Discharge Volume	Throttle OPEN CRD-V-280, SDV VENT SHUTOFF VALVE, (East end of SDV above HCUs). **CUE**: (First) the handwheel is turning counter-clockwise and the valve stem is rising. Sound of rushing air. **CUE**: (Then) the handwheel is fully counter-clockwise and the valve stem is out. **CUE**: (When operator begins timing) state that 3 minutes have elapsed. **CUE**: (If asked) the control room states that the south SDV NOT DRAINED alarm has reset and directs the operator to secure draining the south scram discharge volume.	__________*
7. CLOSE CRD-V-279	Close CRD-V-279, SDV VENT ROOT VALVE (East end of SDV above HCUs). **CUE**: The handwheel is fully clockwise and the valve stem is in.	__________*

Task No.: 200078A0504

Task Title: Conduct Manual Draining of the SDV (Alternate Rod Insertion)

Performance Checklist	Standard	Initials
8. CLOSE CRD-V-280	Close CRD-V-280, SDV VENT SHUTOFF VALVE, (East end of SDV above HCUs). **CUE**: The handwheel is fully clockwise and the valve stem is in.	__________*
9. CLOSE CRD-AOV-CV33	Turn the handwheel for CRD-AOV-CV33 clockwise until it will no longer turn. **CUE**: (First) The handwheel has disengaged upper nut. **CUE**: (Then) The handwheel stops turning. The stem indicator is at the bottom of the scale. The bottom limit switch is made up.	__________*
10. Lock the valve operator	Tighten the locknut below the handwheel for CRD-AOV-CV33 with the special wrench. **CUE**: The locknut is tight.	_________*
11. CLOSE CRD-AOV-CV35	Turn the handwheel for CRD-AOV-CV35 clockwise until it will no longer turn. **CUE**: (First) The handwheel has disengaged upper nut. **CUE**: (Then) The handwheel stops turning. The stem indicator is at the bottom of the scale. The bottom limit switch is made up.	__________*

Task No.: 200078A0504

==

Task Title: Conduct Manual Draining of the SDV (Alternate Rod Insertion)

==

Performance Checklist	Standard	Initials
12. Lock the valve operator	Tighten the locknut below the hand-wheel for CRD-AOV-CV35 with the special wrench. **CUE**: The locknut is tight.	__________*
13. Inform CRS the task is complete.	Inform the Control Room Supervisor that the South SDV has been drained. **CUE**: The Control Room operator acknowledges the report.	__________

ATTACHMENT 1

Directions to Trainee:

When I tell you to begin, you are to manually drain the south Scram Discharge Volume. Before you start, I will state the general plant conditions, the initiating cues, and answer any questions you may have.

When simulating, physically point to any meters, gauges, recorders and controls you would be using. State the position of controls as you would have manipulated them in order to complete the assigned task.

General Conditions:

1. The plant scrammed; however, all rods have not been inserted to or beyond 02.
2. The scram has been Reset.
3. The SDV vent and drain valves remain closed and cannot be reopened.
4. EOP PTMs #31, 32, 33, and 34 have been installed.
5. TSC is not yet operational.
6. No ARMs are alarming.
7. The In-Containment Rad Monitors are reading 100 Rem/hr.
8. An operator has been dispatched to drain the North SDV.

Initiating Cue(s):

The Control Room Supervisor directs you to manually drain the south scram discharge volume per Emergency Operating Procedure 5.8.3 and to secure the south SDV drain lineup when draining is complete. You are to inform the CRS when the task is complete.

Task No.: 233016O00104

===

Task Title: Refill FPC Skimmer Surge Tank

===

Trainee: ___________________________ Examiner: ___________________________

Pass:___ Fail:____ Examiner Signature:_________________________ Date:_________

Additional Program Information:

1. Appropriate Performance Locations: Plant
2. Appropriate Trainee Level: SO / RO / SRO
3. Evaluation Method: __ Simulate __ Perform
4. Performance Time: 12 minutes
5. NRC K/A 233000 A2.03 (2.8/3.0)

Directions to Examiner:

1. This JPM evaluates the trainee's ability to refill the Fuel Pool Cooling Skimmer Surge Tank.
2. All blanks must be filled out with either initials or an “NP” for “not performed”; an explanation may also be written in the space, if desired, by the examiner.
3. Give the trainee his copy of the Directions to the Trainee (Attachment 1) when ready to start the JPM.
4. Brief the trainee and tell the trainee to begin.

Directions to Trainee:

When I tell you to begin, you are to refill the Fuel Pool Cooling Skimmer Surge Tank. Before you start, I will state the general plant conditions, the Initiating Cues and answer any questions you may have.

When simulating, physically point to any meters, gauges, recorders and controls you would be using. State the position of controls as you would have manipulated them in order to complete the assigned task.

==

Task No.: 233016O0104

==

Task Title: Refill FPC Skimmer Surge Tank

==

General Conditions:

1. The plant is operating at power.
2. Alarm **9-4-2/C-3** "Skimmer Tank Low Level" has been received.
3. You are the Reactor Building Station Operator.

General References:

1. Procedure 2.2.32, Fuel Pool Cooling and Demineralizer System

General Tools and Equipment:

1. None

Special Conditions, References, Tools, Equipment:

1. Critical steps denoted by "*."

Task Standards:

1. 100% of critical elements successfully completed without error.
2. 100% of safety and radiological work practices.

Initiating Cue(s):

The Control Room operator directs you to refill the Skimmer Surge Tank per procedure 2.2.32. Notify the CRO when the task is complete.

Task No.: 233016O00104

==

Task Title: Refill FPC Skimmer Surge Tank

==

Performance Checklist	Standards	Initials
1. Open CM-85, CONDENSATE SUPPLY TO SKIMMER SURGE TANK 1B SHUTOFF (R-1001-SE skimmer surge tank valve box).	The operator opens CM-85. **CUE**: CM-85 handwheel rotates counter-clockwise several turns, then stops rotating.	_______*
2. Open CM-47, FUEL POOL & SERVICE BOXES SUPPLY ROOT (R-958-SE).	The operator opens CM-47. **CUE**: CM-47 handwheel rotates counter-clockwise several turns, then stops rotating.	_______*
3. When level on LI-70 is just below high level alarm level switch (R-976-SE), close CM-47.	The operator closes CM-47. **CUE**: Provide indication on LI-70 that is is just below high level alarm level switch. **CUE**: CM-47 handwheel rotates clockwise several turns, then stops rotating.	_______*
4. Close CM-85.	The operator closes CM-85. **CUE**: CM-85 handwheel rotates clockwise several turns, then stops rotating.	_______
5. Inform the Control Room Operator of completion.	Notify the CRO that the skimmer surge tank has been refilled. **CUE**: The CRS acknowledges the report.	_______

ATTACHMENT 1

When I tell you to begin, you are to refill the Fuel Pool Cooling Skimmer Surge Tank. Before you start, I will state the general plant conditions, the Initiating Cues and answer any questions you may have.

When simulating, physically point to any meters, gauges, recorders and controls you would be using. State the position of controls as you would have manipulated them in order to complete the assigned task.

General Conditions:

1. The plant is operating at power.
2. Alarm **9-4-2/C-3** "Skimmer Tank Low Level" has been received.
3. You are the Reactor Building Station Operator.

Initiating Cues:

The Control Room operator directs you to refill the Skimmer Surge Tank per procedure 2.2.32. Notify the CRO when the task is complete.

2003 CNS NRC Written Exam References

The following are the references to be provided to the license candidates during the NRC written examination:

Reactor Operator Exam
Non-programmable calculator
Ruler
Tablet
Steam Tables with attached Mollier diagram (Question #25, 14027)
GFES Forumula sheet (Question #25, 14027)
All EOP flowcharts and graphs with entry conditions, cautions, combustible limits and EOP-7A lowering level reasons removed. (Numerous questions, standard exam reference)
T. S. LCO Applicability (Section 3.0) and associated bases (Multiple questions)
T. S. 3.4.4 & 3.4.5 and associated bases (Question 44, 14042)
T. S. 3.6.1.4 & 3.6.1.5 and associated bases (Question 45, 14051)
ESP 5.8.3 Flowchart (Question 51, 14001)
T. S. 3.3.1.1 and associated bases (Question 95, 14003)
T. S. 3.3.3.1 and associated bases (Question 96, 14026)

2003 CNS NRC Written Exam References

Senior Reactor Operator Exam
Non-programmable calculator
Ruler
Tablet
Steam Tables with attached Mollier diagram (Question #25, 14027)
GFES Forumula sheet (Question #25, 14027)
All EOP flowcharts and graphs with entry conditions, cautions, combustible limits and EOP-7A lowering level reasons removed. (Numerous questions, standard exam reference)
T. S. LCO Applicability (Section 3.0) and associated bases (Multiple questions)
T. S. 3.4.4 & 3.4.5 and associated bases (Question 44, 14042)
T. S. 3.6.1.4 & 3.6.1.5 and associated bases (Question 45, 14051)
ESP 5.8.3 Flowchart (Question 51, 14001)
T. S. 3.3.1.1 and associated bases (82, 4002)
T. S. 3.3.3.1, 3.8.1 & 3.8.3 and bases (Question 83, 38)
T. S. 3.1.3 & 3.1.4 and associated bases (Question 84, 48)
T. S. 3.1.6 & 3.3.2.1 and associated bases (Question 85, 114)
T.S. section 3.0, 3.6.1.3, 3.6.2.3 and bases, 6.1RHR.201 (Question 87, 16415)
TRM section 3.11 and bases & 0.23 "CNS FIRE PROTECTION PLAN". (Question 90, 46)
0.50, Outage Management Program.(Question 100, 9684)

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

Applicant Information

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

Instructions

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

Applicant Certification

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

____
Applicant's Signature

Results

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

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
1	1111	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0010102 AC Electrical Distribution

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

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

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

QUESTION: 1111

Given the following conditions:

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

Which breaker(s) will trip?

a. **ONLY** EG1

b. **ONLY** 1AF

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

d. **BOTH** EG1 and 1FA

ANSWER: 1111

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

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

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

Distractors:

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

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
2	14036	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0010102 AC Electrical Distribution

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

Related References	
2.2.63	Procedure 2.2.63, PMIS Uninterruptible Power Supply System

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

QUESTION: 14036

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

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

PMIS would be . . .

a. deenergized.

b. powered directly from MDP-1.

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

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

ANSWER: 14036

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

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

Answer source: 2.2.63, p. 10, step 1.2.1

Distractors:

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

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
3	1099	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0010102 AC Electrical Distribution

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

Related References	
2.2.18	4160V Auxiliary Power Distribution System

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

QUESTION: 1099

Given the following conditions:

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

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

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

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

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

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

ANSWER: 1099

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

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

Answer source: 2.2.20, p. 29, step 2.7.1.5

Distractors:

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

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
4	14035	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0011302 OPS MAIN GENERATOR AND AUXILIARIES

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

Related References	
2.2.14	22 KV Electrical System

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

QUESTION: 14035

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

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

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

MVARs become . . .

a. less negative and field amperage increases.

b. less negative and field amperage decreases.

c. more negative and field amperage increases.

d. more negative and field amperage decreases.

ANSWER: 14035

d. more negative and field amperage decreases.

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

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

Distractors:

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

Source: *New*

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
5	14670	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0011402 OPS MAIN TURBINE

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

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

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

QUESTION: 14670

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

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

a. Reheat stop valves.

b. Extraction steam dump valves.

c. Extraction Steam Non-Return valves.

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

ANSWER: 14670

b. Extraction steam dump valves.

Answer source: 2.2.29, p. 19, step 2.1

Distractors:

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

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
6	19124	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0022802 OPS STANDBY GAS TREATMENT COR0011802 OPS Radiation Monitoring

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

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

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

QUESTION: 19124

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

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

NO group isolations or automatic initiations occur.

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

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

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

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

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

ANSWER: 19124

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

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

Answer source: 4.7.5, pp. 5 & 6, steps 1.2.3, 1.2.4, 1.2.5, & 1.3.1.1

Distractors:

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

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
7	5084	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0012002 OPS Reactor Water Cleanup

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

Related References	
2.1.22	Procedure 2.1.22, Recovering From A Group Isolation

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

QUESTION: 5084

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

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

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

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

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

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

ANSWER: 5084

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

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

Answer source: 2.1.22, p. 10, step 6.4

Distractors:

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

Source: ***Modified from 5084***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
8	14043	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0012402 OPS TURBINE EQUIPMENT COOLING SYSTEM

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

Related References	
2.2.76	Procedure 2.2.76, Turbine Equipment Cooling Water System

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

QUESTION: 14043

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

The normal supply is from the . . .

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

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

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

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

ANSWER: 14043

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

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

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

Distractors:

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

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
9	14045	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0013001 HWC Gas Generation System

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

Related References	
2.2.98	Hydrogen/Oxygen Generation System

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

QUESTION: 14045

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

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

ERP release rate . . .

a. increases and MSL radiation levels increase.

b. decreases and MSL radiation level decrease.

c. is unchanged and MSL radiation level decrease.

d. is unchanged and MSL radiation level is unchanged.

ANSWER: 14045

c. is unchanged and MSL radiation level decrease.

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

<u>Answer source:</u> COR012-03-01, p. 4

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

Distractors:

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

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

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
10	2931	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0020302 CONTAINMENT

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

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

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

QUESTION: 2931

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

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

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

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

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

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

ANSWER: 2931

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

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

Distractors:

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

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
11	10081	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0020302 CONTAINMENT

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

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

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

QUESTION: 10081

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

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

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

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

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

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

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

ANSWER: 10081

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

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

Distractors:

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

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

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
12	5155	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0020402 CONTROL ROD DRIVE HYDRAULICS

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

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

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

QUESTION: 5155

Given the following conditions:

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

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

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

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

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

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

ANSWER: 5155

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

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

Answer source: 2.2.8A, p. 10

Distractors:

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

Source: *Direct from bank*

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
13	14040	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0020602 CORE SPRAY

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

Related References	
2.2.9	Procedure 2.2.9, Core Spray System

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

QUESTION: 14040

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

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

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

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

a. remains the same.

b. increases to pump shut-off head.

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

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

ANSWER: 14040

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

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

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

Distractors:

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

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
14	14050	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0020702 OPS DC ELECTRICAL DISTRIBUTION

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

Related References	
2.2.25.2	125 VDC ELECTRICAL SYSTEM (DIV 2)

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

QUESTION: 14050

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

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

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

a. **only**.

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

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

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

ANSWER: 14050

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

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

Distractors:

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

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
15	19090	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0020702 OPS DC ELECTRICAL DISTRIBUTION COR0021202 INTERMEDIATE RANGE MONITOR

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

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

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

QUESTION: 19090

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

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

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

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

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

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

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

ANSWER: 19090

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

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

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

Distractors:

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

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

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
16	1507	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0020802 DIESEL GENERATORS COR0010102 AC Electrical Distribution

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

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

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

QUESTION: 1507

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

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

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

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

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

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

ANSWER: 1507

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

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

Distractors:

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

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
17	14032	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0021102 OPS High Pressure Coolant Injection (HPCI)

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

Related References	
2.2.33	Procedure 2.2.33, High Pressure Coolant Injection System

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

QUESTION: 14032

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

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

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

a. open closed

b. closed closed

c. open open

d. closed open

ANSWER: 14032

a. open closed

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

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

Distractors:

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

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

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
18	19051	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

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

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

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

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

QUESTION: 19051

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

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

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

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

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

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

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

ANSWER: 19051

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

Per 2.2.33.1:

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

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

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

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

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

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
19	16513	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0020902 Digital Electro-Hydraulic Control

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

Related References	
2.2.77.1	Procedure 2.2.77.1, DEH Control System

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

QUESTION: 16513

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

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

a. The reactor scram due to high reactor pressure.

b. The MSIVs isolate due to low reactor pressure.

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

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

ANSWER: 16513

b. The MSIVs isolate due to low reactor pressure.

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

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

Distractors:

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

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
20	1058	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0021502 NUCLEAR BOILER INSTRUMENTATION

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

Related References	
COR0021502	Nuclear Boiler Instrumentation

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

QUESTION: 1058

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

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

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

a.	lower	larger
b.	higher	larger
c.	lower	smaller
d.	higher	smaller

ANSWER: 1058

d. higher smaller

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

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

Distractors:

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

Source: *Direct from bank*

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
21	5425	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0021602 OPS NUCLEAR PRESSURE RELIEF

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

Related References	
2.2.1	Automatic Depressurization System

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

QUESTION: 5425

Given the following:

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

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

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

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

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

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

ANSWER: 5425

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

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

Distractors:

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

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
22	5608	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0021602 OPS NUCLEAR PRESSURE RELIEF

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

Related References	
791E253	Automatic Blowdown System
2.2.1	Automatic Depressurization System

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

QUESTION: 5608
An accident has occurred, resulting in the following conditions:

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

If present conditions continue, how will ADS respond?

ADS valves will . . .

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

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

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

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

ANSWER: 5608

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

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

Distractors:

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

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
23	14679	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0021702 PLANT MANAGEMENT INFORMATION SYSTEM

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

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

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

QUESTION: 14679

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

What is the impact of these conditions on plant operation?

Official Cases are . . .

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

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

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

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

ANSWER: 14679

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

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

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

Distractors:

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

Source: ***Direct from Bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
24	14451	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0021802 OPS Reactor Core Isolation Cooling (RCIC)

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

Related References	
2.2.67	Procedure 2.2.67, Reactor Core Isolation Cooling System

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

QUESTION: 14451

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

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

The RCIC turbine will start and . . .

a. run normally.

b. trip on overspeed.

c. run continuously at minimum speed.

d. run continuously at approximately 4500 rpm.

ANSWER: 14451

d. run continuously at approximately 4500 rpm.

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

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

Distractors:

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

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
25	14027	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

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

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

Related References	
ACD0060507	Reactor Operational Physics

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

QUESTION: 14027

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

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

A station blackout then occurs.

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

a. 83 psig

b. 90 psig

c. 165 psig

d. 180 psig

ANSWER: 14027

c. 165 psig

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

Answer source: Steam Tables, Generic Fundamentals

Distractors:

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

Source: ***New***

Provide to Candidate: Calculator, Steam Tables, GFES Formula sheet.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
26	2521	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0021902 REACTOR EQUIPMENT COOLING

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

Related References	
2.2.65	Procedure 2.2.65, Reactor Equipment Cooling Water System
5.2REC	LOSS OF REC

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

QUESTION: 2521

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

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

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

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

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

ANSWER: 2521

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

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

Distractors:

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

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
27	14039	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0021902 REACTOR EQUIPMENT COOLING

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

Related References	
2.2.65	Procedure 2.2.65, Reactor Equipment Cooling Water System

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

QUESTION: 14039

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

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

a. "A" Drywell Fan Coil Unit

b. "A" Station Air Compressor

c. "A" Control Rod Drive pump

d. Northwest Quad Fan Coil Unit.

ANSWER: 14039

d. Northwest Quad Fan Coil Unit.

With an isolation signal present REC-MO-702MV can be reopened, however, the REC-MO-712 and 713 will auto close on the low pressure and cannot be overriden. This will isolate REC to the non-critical loops/components. The fan coil unit is the only load listed supplied from the critical loop.

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

Distractors:

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

Source: Modified *Original Question 5279*

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
28	14024	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0022002 OPS REACTOR MANUAL CONTROL SYSTEM

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

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

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

QUESTION: 14024

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

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

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

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

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

ANSWER: 14024

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

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

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

Distractors:

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

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

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

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
29	1208	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0022102 REACTOR PROTECTION SYSTEM

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

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

Related Skills (K/A)	
212000.A2.09	Ability to (a) predict the impacts of the following on the REACTOR PROTECTION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the conseq...: (CFR: 41.5 / 45.6) High containment/drywell pressure (4.1*/4.3*)

QUESTION: 1208

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

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

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

A reactor scram will occur . . .

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

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

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

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

ANSWER: 1208

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

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

Distractors:

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

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

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
30	19096	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
SKL0124222 REACTOR RECIRCULATION SYSTEM COR0022202 REACTOR RECIRCULATION

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

Related References	
2.2.68	Procedure 2.2.68, Reactor Recirculation System Operations

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

QUESTION: 19096

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

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

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

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

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

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

ANSWER: 19096

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

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

Distractors:

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

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
31	1744	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0022302 RESIDUAL HEAT REMOVAL

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

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

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

QUESTION: 1744

Following a LOCA, the following conditions are present :

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

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

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

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

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

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

ANSWER: 1744

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

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

Distractors:

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

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

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
32	4029	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0022302 RESIDUAL HEAT REMOVAL

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

Related References	
2.2.69	Procedure 2.2.69, Residual Heat Removal System

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

QUESTION: 4029

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

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

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

RHR . . .

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

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

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

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

ANSWER: 4029

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

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

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

Distractors:

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

<u>Source</u>: ***Modified original question 4029.***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
33	14028	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0022302 RESIDUAL HEAT REMOVAL

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

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

Related Skills (K/A)	
203000.A2.02	Ability to (a) predict the impacts of the following on the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of...: (CFR: 41.5 / 45.6) Pump trips (3.5/3.5)

QUESTION: 14028

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

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

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

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

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

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

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

ANSWER: 14028

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

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

Distractors:

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

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

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
34	2127	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0022602 OPS ROD WORTH MINIMIZER

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

Related References	
4.2	Procedure 4.2, Rod Worth Minimizer

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

QUESTION: 2127

Why is the RWM required below 10% power but **not** above 10%?

At higher power,

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

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

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

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

ANSWER: 2127

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

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

Distractors:

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

Source: ***Direct from Bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
35	14033	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0022802 OPS STANDBY GAS TREATMENT

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

Related References	
2.2.73	Procedure 2.2.73, Standby Gas Treatment System

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

QUESTION: 14033

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

What is the potential effect on SGT?

a. SGT flow decrease

b. Inlet HEPA filter moisture damage

c. Moisture impingment on the SGT fans

d. Increased iodine carryover at the SGT train outlet

ANSWER: 14033

a. SGT flow decrease.

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

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

Distractors:

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

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
36	14025	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0022902 STANDBY LIQUID CONTROL

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

Related References	
COR0022902	SLC
2.2.9	Procedure 2.2.9, Core Spray System

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

QUESTION: 14025

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

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

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

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

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

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

ANSWER: 14025

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

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

__**Answer source:**__ COR002-29-02 Figure 7

Distractors:

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

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
37	5348	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0023002 SOURCE RANGE MONITOR SUBSYSTEM

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

Related References	
4.1.1	Procedure 4.1.1, Source Range Monitoring System

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

QUESTION: 5348

A Reactor Startup is in progress with the following conditions:

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

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

a. Associated IRMs are on Range 3 or higher.

b. Associated IRMs are on Range 8 or higher.

c. Associated IRMs are on Range 9 or higher.

d. The Mode switch is placed to RUN.

ANSWER: 5348

b. Associated IRMs are on Range 8 or higher.

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

<u>Answer source:</u> 4.1.1 p. 5, step 1.2.2

Distractors:

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

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

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
38	19084	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0023102 OPS TRAVERSING IN-CORE PROBE

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

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

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

QUESTION: 19084

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

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

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

What action is ***required***?

a.	Fire TIP "A" shear valve.

b.	Close TIP "A" ball valve.

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

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

ANSWER: 19084

b.	Close TIP "A" ball valve.

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

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

Distractors:

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

Source: Direct from Bank

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
39	14004	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0023202 OPS REACTOR VESSEL LEVEL CONTROL

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

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

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

QUESTION: 14004

Given the following conditions:

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

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

The RFPT M/A station shifts to __________ mode with controller output ____________ output prior to the event.

a. MDEM the same as

b. MDVP the same as

c. MDEM higher than

d. MDVP higher than

ANSWER: 14004

a. MDEM, the same as

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

Distractors:

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

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
40	14014	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0023402 Alternate Shutdown (LO)

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

Related References	
5.1ASD	Shutdown From Outside The Control Room

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

QUESTION: 14014

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

a. To prevent spurious equipment operation.

b. To ensure automatic operation of ECCS remains available.

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

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

ANSWER: 14014

a. To prevent spurious equipment operation.

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

Distractors:

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

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
41	8970	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0070501	OPS Introduction to Technical Specifications

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

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

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

QUESTION: 8970

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

What is a potential consequence of this event?

a. Increased likelihood of thermal hydraulic instabilities.

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

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

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

ANSWER: 8970

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

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

Answer source: Safety Limit Violation Bases p. B 2.0-5

Distractors:

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

Source: ***Modified from 8970***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
42	3995	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0070502	CNS Tech. Spec. 3.1, Reactivity Control Systems

Related Objectives	
INT0070502001060A	From memory, for MODES 1 and 2, state the actions required in less than one hour for: two or more control rod scram accumulators inoperable with the reactor steam dome pressure less than or equal to 940 psig (LCO 3.1.5 B.1).

Related References	
3.1.5	Control rod scram accumulators

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

QUESTION: 3995

Given the following conditions:

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

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

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

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

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

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

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

ANSWER: 3995

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

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

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

Distractors:

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

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
43	14048	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0070504	CNS Tech. Spec. 3.3, Instrumentation

Related Objectives	
INT00705040010200	Discuss the applicable Safety Analysis in the Bases associated with each Section 3.3 Specification.

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

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

QUESTION: 14048

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

a. To indirectly prevent damage to the Moisture Separators by low enthalpy fluid.

b. To prevent ECCS equipment damage from missiles created by main turbine failure.

c. To ensure flow induced vibration of the main steam lines remains within analytical limits.

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

ANSWER: 14048

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

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

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

Distractors:

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

Source: *New*

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
44	14042	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

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

Related Objectives	
INT00705050010100	Given a set of plant conditions, recognize non-compliance with a Chapter 3.4 LCO.

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

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

QUESTION: 14042

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

What is required?

Place/Leave . . .

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

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

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

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

ANSWER: 14042

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

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

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

Distractors:

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

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

<u>Provide to Candidate:</u> T.S. 3.0 section and bases, T.S. LCO 3.4.4 and bases, T.S. LCO 3.4.5 and bases.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
45	14051	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

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

Related Objectives	
INT00705070010300	Given a set of plant conditions that constitutes non-compliance with a Chapter 3.6 LCO, determine the ACTIONS that are required.
INT00705070010100	Given a set of plant condtions, recognize non-compliance with a Chapter 3.6 LCO.

Related References	
3.6.1.5	Drywell air temperature
3.6.1.4	Drywell pressure

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

QUESTION: 14051

The plant was operating at power when drywell pressure and temperature began rising. The following parameters existed at 0700 on 5/20:

- ➢ Drywell pressure 0.8 psig (rising)
- ➢ Drywell temperature 142°F (rising)

The crew then reduced reactor power using Reactor Recirculation flow and began venting the drywell and the torus. At 0800 on 5/20, the crew discovered the following primary containment parameters:

- ➢ Drywell pressure 0.45 psig (stable)
- ➢ Drywell temperature 151°F (rising)

IF conditions do not improve, what is the LATEST time that the plant is ***allowed*** to enter **MODE 3** by Technical Specifications?

Be in MODE 3 by . . .

a. 1600 on 5/20.

b. 0300 on 5/21.

c. 0400 on 5/21.

d. 0300 on 5/22.

ANSWER: 14051

c. 0400 on 5/21.

Enter 3.6.1.4 at 0700 (RA 3.6.1.A). Exit 3.6.1.4 at 0800. Enter 3.6.1.5 at 0800, RA A.1. Enter RA B.1 (B.2) at 1600 on 5/20. Be in MODE 3 by 0400 on 5/21.

<u>Answer source:</u> Tech Spec LCO 3.6.1.4 p. 3.6-16

Distractors:

a. Time when RA 3.6.1.5.B.1 must be entered, not completed.
b. Time to MODE 3 if use wrong start time (0700).
d. Time to MODE 4 if use wrong start time (0700).

Source: ***New***

Provide to Candidate: T.S. 3.0 section and bases, T.S. LCO 3.6.1.4 and bases, 3.6.1.5 and bases.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
46	5247	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

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

Related Objectives	
INT0080605001010A	List the entry conditions of Flowchart 1A: Describe the importance of each in an emergency situation.

Related References	
EOP 1A, RPV CON	RPV Control

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

QUESTION: 5247

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

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

Which procedure(s) must be executed?

2.1.5 "Reactor Scram" . . .

a. **ONLY**.

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

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

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

ANSWER: 5247

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

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

<u>Answer source:</u> EOP-1A, 2.1.5 p. 1

Distractors:

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

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

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
47	13407	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

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

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

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

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

QUESTION: 13407

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

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

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

a. No additional requirements must be met.

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

c. A RP Technician must accompany the operator.

d. The TSC must be declared operational.

ANSWER: 13407

d. The TSC must be declared operational.

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

Answer source: 5.8.2 p. 12, section 5.1

Distractors:

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

Source: ***Direct from Bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
48	14000	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

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

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

Related References	
2.1.22	Procedure 2.1.22, Recovering From A Group Isolation

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

QUESTION: 14000

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

What group isolations have automatically occurred?

Group 2 . . .

a. **only**.

b. and Group 3 **only**.

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

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

ANSWER: 14000

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

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

Distractors:

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

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

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
49	17991	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

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

Related Objectives	
INT00806050010600	Identify the characteristics of a "cycling SRV" and the reasons the cycling must be stopped.
INT00806050011100	Given an EOP flowchart 1A, RPV CONTROL step, state the reason for the actions contained in the step.

Related References	
INT0080605	Flowchart 1A - RPV Control/RPV Pressure
PSTG	Plant Specific Technical Guideline

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

QUESTION: 17991

When executing EOP flowchart 1A-RPV CONTROL, the operator is directed to open SRVs to reduce RPV pressure to 940 psig if SRVs are cycling.

What is basis for selecting the value of 940 psig?

940 psig is . . .

a. the lowest pressure that will result in fully open Main Turbine bypass valves.

b. low enough to provide a sufficient margin below the high pressure scram setpoint.

c. high enough to provide a sufficient margin above the shutoff head of CS and RHR pumps.

d. low enough to prevent the actuation of SRVs on mechanical setpoints and prevents arming of LLS.

ANSWER: 17991

a. the lowest pressure that will result in fully open Main Turbine bypass valves.

RPV pressure reduction with SRVs is continued until RPV pressure reaches the pressure at which steam flow through the main turbine BPVs is at 100% of BPV capacity. If the MSIVs are open, reducing RPV pressure to below this value results in partial closure of the BPVs and a corresponding rise in the amount of steam discharged to the suppression pool through the SRVs. If the MSIVs are not open, reducing RPV pressure to the lowest pressure at which all BPVs would be fully open, if controlling pressure, provides an adequate operating margin below the setpoint pressure of the lowest lifting SRV.

Answer source: INT008-06-05 p. 15

Distractors:

b. While this would provide a margin to the scram setpoint, it is not the reason why this action is taken in EOP-1A.

c. While this pressure does provide a substantial margin to the shutoff head of the CS and RHR pumps it is not the reason for this action.

d. While this pressure is below the lowest lift setpoint of the relief valves prevention of LLS arming and actuation is not part of the reason. Verification of Low Low Set operation may ensure this action is performed automatically.

Source: ***Direct from Bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
50	19034	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

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

Related Objectives	
INT00806050011000	Given plant conditions and EOP flowchart 1A, RPV CONTROL, determine required actions.
INT00806180010200	For each graph used in the flowcharts, identify the action(s) required if the parameters associated indicate operation in the restricted or prohibited area.
INT00806050011200	Given plant conditions, assess if RPV water level can be determined or not.

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

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

QUESTION: 19034

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

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

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

Wide Range Reactor Level Instrumentation . . .

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

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

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

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

ANSWER: 19034

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

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

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

Distractors:

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

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

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

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
51	14001	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

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

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

Related References	
5.8.3	Procedure 5.8.3, Alternate Rod Insertion Methods

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

QUESTION: 14001

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

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

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

a. Individually scram control rods.

b. Drain the SDV and manually scram.

c. Manually vent CRDM over-piston area.

d. Insert control rods using emergency override switch.

ANSWER: 14001

c. Manually vent CRDM over-piston area.

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

Answer source: 5.8.3 flowchart, path "B"

Distractors:

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

Source: *New*

Provide to Candidate: 5.8.3 Flowchart

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
52	16472	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

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

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

Related References	
5.8.3	Procedure 5.8.3, Alternate Rod Insertion Methods

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

QUESTION: 16472

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

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

Fully insert rod . . .

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

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

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

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

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

ANSWER: 16472

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

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

Answer source: 5.8.3 p. 4, section 4.2

Distractors:

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

Source: ***Direct from Bank***

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

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

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

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

Related References	
PSTG	Plant Specific Technical Guideline

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

QUESTION: 16485

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

Reactor Pressure . . .

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

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

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

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

ANSWER: 16485

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

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

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

Distractors:

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

Source: *Modified*

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

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
54	8939	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

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

Related Objectives	
INT00806100010300	State the basis for intentionally lowering RPV water level and the criteria for the lowered level (LL).
INT00806100010900	Given an EOP flowchart 7A, RPV LEVEL (FAILURE TO SCRAM) step, state the reason for the actions contained in the step.

Related References	
INT0080610	Flowchart 7A RPV Level Failure to Scram
PSTG	Plant Specific Technical Guideline

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

QUESTION: 8939

Why is RPV level lowered during a failure-to-scram event to 100" (corrected fuel zone)?

To prevent or mitigate the consequences of . . .

a. any bundle exceeding the MCPR safety limit.

b. exceeding the peak cladding temperature of 1500 °F.

c. any large irregular neutron flux oscillations induced by neutronic/thermal-hydraulic instabilities.

d. exceeding the Heat Capacity Temperature Limit and challenging the Primary Containment Pressure Limit.

ANSWER: 8939

c. any large irregular neutron flux oscillations induced by neutronic/thermal-hydraulic instabilities.

<u>Answer source:</u> EOP-7A step FS/L-6

Distractors:

a. Lowering level to -100" FZ suppression any large irregular neutron flux oscillations but, there is on assurance that the MCPR safety limit will not be violated on a failure-to-scram event.
b. The lower limit (-25 in. (FZ)) is the Minimum Steam Cooling RPV Water Level (MSCRWL). Which is the basis for the 1500 ºF PCT.
d. 0 inches on Table 17 & 28 is based on the ability of SPC to remove heat and not challenge Pri Cont.

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

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
55	14047	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0080613	OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL

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

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

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

QUESTION: 14047

Given the following conditions:

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

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

a. 195°F

b. 205°F

c. 211°F

d. 215°F

ANSWER: 14047

b. 205°F

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

Distractors:

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

Source: ***New***

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

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
56	5268	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0080613	OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL

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

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

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

QUESTION: 5268

Given the following conditions:

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

What action is required to control Drywell conditions?

a. Initiate drywell sprays.

b. Operate all available drywell cooling.

c. Vent primary containment with torus vent line.

d. Vent primary containment with drywell vent line.

ANSWER: 5268

b. Operate all available Drywell Cooling.

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

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

Distractors:

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

Source: ***Modified 5268***

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

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
57	5332	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0080613	OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL

Related Objectives	
INT00806130011100	Given plant conditions and EOP Flowchart 3A, PRIMARY CONTAINMENT CONTROL, determine required actions.
INT00806130011200	Given plant conditions and EOP flowchart 3A, PRIMARY CONTAINMENT CONTROL, state the reasons for the actions contained in the steps.
INT0080613001040A	State the basis for primary containment control actions as they apply to the following: Specific setpoints

Related References	
PSTG	Plant Specific Technical Guideline

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

QUESTION: 5332

Given the following conditions:

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

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

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

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

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

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

ANSWER: 5332

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

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

Distractors:

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

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

Source: ***Direct from bank***

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

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
58	5333	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0080613	OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL

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

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

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

QUESTION: 5333

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

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

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

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

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

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

ANSWER: 5333

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

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

Distractors:

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

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
59	14023	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0080613	OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL

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

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

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

QUESTION: 14023

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

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

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

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

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

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

ANSWER: 14023

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

Answer source: INT008-06-13 p. 22

Distractors:

a. RCIC exhaust flowrate, factors for basis of PCPL-A.
b. The "lowest energy state" is an Emergency Depressurization basis. There is no more severe strategy than Emergency Depressurization.
d. It is not a "potential" energy addition and scramming the reactor does not remove the source of the energy addition.

Source: *New*

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

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
60	14044	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0080613	OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL

Related Objectives	
INT00806130011200	Given plant conditions and EOP flowchart 3A, PRIMARY CONTAINMENT CONTROL, state the reasons for the actions contained in the steps.
INT00806130011000	Identify any EOP support procedures referenced in Flowchart 3A and apply any associated special operating instructions or cautions.

Related References	
PSTG	Plant Specific Technical Guideline

Related Skills (K/A)	
223001.K6.04	Knowledge of the effect that a loss or malfunction of the following will have on the PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES: (CFR: 41.7 / 45.7) Combustible gas mixing: Plant-Specific (2.8/2.8)

QUESTION: 14044

The plant was operating at rated power for several months when a LOCA with hydrogen generation occurred.

What affect will the inability to initiate drywell sprays have on combustible gas flammability and the primary containment?

The inability to initiate drywell sprays __________ the flammability of combustible gases and _________ the chance of containment damage from hydrogen deflagration.

a.	reduces	reduces
b.	reduces	increases
c.	increases	reduces
d.	increases	increases

ANSWER: 14044

d. increases increases

Answer source: INT008-06-13 p. 9

Distractors:

a., b, c Drywell spray reduces the flamability and potential for deflagration. Loss of Drywell sprays has the opposite affect.

Source: *New*

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

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
61	5730	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0080617	OPS FLOWCHART 5A - SECONDARY CONTAINMENT AND RADIOACTIVITY RELEASE CONTROL

Related Objectives	
INT00806170010400	State the basis for the limits of the maximum safe operating values (MSO) as they apply to personnel protection and equipment operability.

Related References	
PSTG	Plant Specific Technical Guideline

Related Skills (K/A)	
2.3.1	Knowledge of 10 CFR: 20 and related facility radiation control requirements. (CFR: 41.12 / 43.4. 45.9 / 45.10) (2.6/3.0)

QUESTION: 5730

What is the basis for the value selected as the EOP Flowchart 5A Maximum Safe Operating (MSO) Radiation level?

It is based on personnel exposures exceeding the . . .

a. CNS TEDE dose limit of 3 Rem ***per quarter*** for a **one hour** stay time.

b. CNS TEDE dose limit of 4 Rem ***per year*** for a **two hour** stay time.

c. 10CFR20 planned special exposure limit of 3 rem for a **one hour** stay time.

d. 10CFR20 planned special exposure limit of 5 rem for a **two hour** stay time.

ANSWER: 5730

b. CNS TEDE dose limit of 4 Rem ***per year*** for a **two hour** stay time.

Answer source: INT008-06-17 pp. 10 & 11

Distractors:

c, d. MSO Radiation levels are not based on the 10CFR20 limits.
a. The quarterly limit no longer exists, but it was previously the basis for MSO radiation level, and herefore remains a plausible distractor.

2.3.1 Knowledge of 10 CFR: 20 and related facility radiation control requirements as it applies to 290001 Secondary Containment

Source: ***Modified from 5730***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
62	14019	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0080617	OPS FLOWCHART 5A - SECONDARY CONTAINMENT AND RADIOACTIVITY RELEASE CONTROL

Related Objectives	
INT00806170010700	Given plant conditions and EOP flowchart 5A, SECONDARY CONTAINMENT CONTROL and RADIOACTIVITY RELEASE CONTROL, state the reasons for the actions contained in the steps.

Related References	
PSTG	Plant Specific Technical Guideline

Related Skills (K/A)	
295038.EK3.04	Knowledge of the reasons for the following responses as they apply to HIGH OFF-SITE RELEASE RATE: (CFR: 41.5 / 45.6) Emergency depressurization (3.6/3.9)

QUESTION: 14019

What is the basis for performing an Emergency Depressurization during the execution of EOP 5A, **RADIOACTIVITY RELEASE CONTROL**?

Performing an Emergency Depressurization ensures the . . .

a. availability of equipment in the turbine building that may be necessary to mitigate the event is not challenged.

b. energy level of the radiation and the atomospheric dispersion factors fall within the bounds of the accident analysis.

c. isotopic mixture of radioactive materials deposited off-site will be within the bounds of the accident analysis.

d. lowest possible driving head and flow of primary systems that are unisolated and discharging outside of containment.

ANSWER: 14019

d. lowest possible driving head and flow of primary systems that are unisolated and discharging outside of containment.

Answer source: INT008-06-17 p. 14

Distractors:

a. availability of turbine building equipment is not an EOP consideration, but sounds like the reactor building ED reason.
b. This is a reason to use the DOSE program for the projections vice the ODAM calculations, but is not the reason for the ED.
c. This is not the basis for the ED.

Source: ***New***

Provide to Candidate: EOP 5A with entry conditions and cautions removed.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
63	14021	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0080617	OPS FLOWCHART 5A - SECONDARY CONTAINMENT AND RADIOACTIVITY RELEASE CONTROL

Related Objectives	
INT00806170010700	Given plant conditions and EOP flowchart 5A, SECONDARY CONTAINMENT CONTROL and RADIOACTIVITY RELEASE CONTROL, state the reasons for the actions contained in the steps.

Related References	
PSTG	Plant Specific Technical Guideline

Related Skills (K/A)	
295036.EK3.02	Knowledge of the reasons for the following responses as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL: (CFR: 41.5 / 45.6) Reactor SCRAM. (2.8/2.8)

QUESTION: 14021

Why is a reactor scram required before exceeding a Maximum Safe Operating Water Level if a primary system is discharging into the Reactor building (EOP-5A step SC-12)?

Scramming the Reactor . . .

a. provides mitigating action such that the condition does not pose an immediate threat to the health and safety of the public.

b. promptly reduces energy to decay heat levels and reduces the likelihood of requiring rapid depressurization of the RPV.

c. promptly reduces to decay heat levels the energy discharged into primary containment and reduces the driving head and flow of primary systems that are unisolated.

d. promptly places the primary system in its lowest possible energy state and reduces the driving head and flow of primary systems that are unisolated and discharging into the secondary containment.

ANSWER: 14021

b. promptly reduces energy to decay heat levels and reduces the likelihood of requiring rapid depressurization of the RPV.

Answer source: INT008-06-17 p. 10, section 10

Distractors:

a. Not a bases for scramming.
c. Primary Containment Control bases.
d. Bases for emergency depressurization.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
64	16483	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0080617	OPS FLOWCHART 5A - SECONDARY CONTAINMENT AND RADIOACTIVITY RELEASE CONTROL

Related Objectives	
INT00806170010600	Given plant conditions and EOP flowchart 5A, SECONDARY CONTAINMENT CONTROL and RADIOACTIVITY RELEASE CONTROL, determine required actions.

Related References	
EOP 5A, SCCP	EOP 5A, Secondary Containment Control
EOP/SAG PSTG	Plant Specific Technical Guideline

Related Skills (K/A)	
2.4.31	Knowledge of annunciators alarms and indications / and use of the response instructions. (CFR: 41.10 / 45.3) (3.3/3.4)

QUESTION: 16483

The plant is at 90% when a RCIC steam line break occurs. Initial event conditions were:

- ➢ RCIC cannot be isolated
- ➢ RCIC temperatures in the NE Quad are 214°F and rising
- ➢ Reactor Building ventilation exhaust radiation levels are 25 mr/hr
- ➢ RCIC Room radiation levels are 300 mr/hr and rising
- ➢ A reactor scram is inserted and all control rods fully insert

Five (5) minutes later, annunciator 9-4-1/E-4 RX BLDG VENT HI HI RAD clears.

What is the required EOP response to this annunciator clearing and what is the bases for the response?

a. Restart Reactor Bldg. HVAC to ensure all radioactive discharges are elevated.

b. Restart Reactor Bldg. HVAC to help return secondary containment parameters to normal.

c. Do **NOT** restart Reactor Bldg. HVAC because EOP 1A requires a group 6 isolation.

d. Do **NOT** restart Reactor Bldg. HVAC until RP ensures normal radiation levels to minimize the spread of contamination.

ANSWER: 16483

b. Restart Reactor Bldg. HVAC to help return secondary containment parameters to normal.

Answer source: INT008-06-17 p. 7, section "b"

Distractors:

a. This is not a bases for restarting Reactor Bldg. HVAC
c. The Isolation would only occur on low level (3) and if so, it should be bypassed.
d. Per 2.1.22 if a Group 6 Isol had occurred, it should not be reset until Chem. and HP have ensured normal rad levels, but EOPs take precedence.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
65	19068	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0080617	OPS FLOWCHART 5A - SECONDARY CONTAINMENT AND RADIOACTIVITY RELEASE CONTROL

Related Objectives	
INT00806170010700	Given plant conditions and EOP flowchart 5A, SECONDARY CONTAINMENT CONTROL and RADIOACTIVITY RELEASE CONTROL, state the reasons for the actions contained in the steps.

Related References	
EOP/SAG PSTG	Plant Specific Technical Guideline

Related Skills (K/A)	
2.3.11	Ability to control radiation releases. (CFR: 45.9 / 45.10) (2.7/3.2)

QUESTION: 19068

What is the basis for restarting building ventilation in the Turbine Building when executing EOP-5A, RADIOACTIVITY RELEASE CONTROL?

Operation of Turbine Building ventilation . . .

a. maintains equipment availability **AND** assures that radioactivity releases pass through a monitored release point.

b. preserves personnel accessibility **AND** assures that radioactivity releases pass through a monitored release point.

c. maintains equipment availability **AND** assures a minimum amount of radioactivity plates out on turbine building surfaces.

d. preserves personnel accessibility **AND** assures a minimum amount of radioactivity plates out on turbine building surfaces.

ANSWER: 19068

b. preserves personnel accessibility **AND** assures that radioactivity releases pass through a monitored release point.

Continued personnel access to the turbine building, radwaste and augmented radwaste may be essential for responding to emergencies. These structures are not air tight and radioactivity release inside them would not only limit personnel access, but would eventually lead to an unmonitored ground level release. Operation of ventilation in these structures preserves accessibility, and assures that radioactivity is discharged through an elevated, monitored release point.

Answer source: INT008-06-17 p. 13, section B.1

Distractors:

a. The purpose of restarting Turbine Building ventilation is not to preserve equipment availability.
c. The purpose of restarting Turbine Building ventilation is not to preserve equipment availability nor to minimize deposition of radioactivity in the building.
d. The purpose of restarting Turbine Building ventilation is not to minimize deposition of radioactivity in the building.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
66	14046	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
INT0080618 EOP AND SAG GRAPHS

Related Objectives	
INT00806180010100	Using the graphs provided in the EOP and SAG Graphs Flowchart, determine how the shape of each curve or family of curves was determined.

Related References	
PSTG	Plant Specific Technical Guideline
INT0080618	EOP and SAG Graphs and Cautions

Related Skills (K/A)	
295030.EK1.03	Knowledge of the operational implications of the following concepts as they apply to LOW SUPPRESSION POOL WATER LEVEL: (CFR: 41.8 to 41.10) Heat capacity. (3.8/4.1*)

QUESTION: 14046

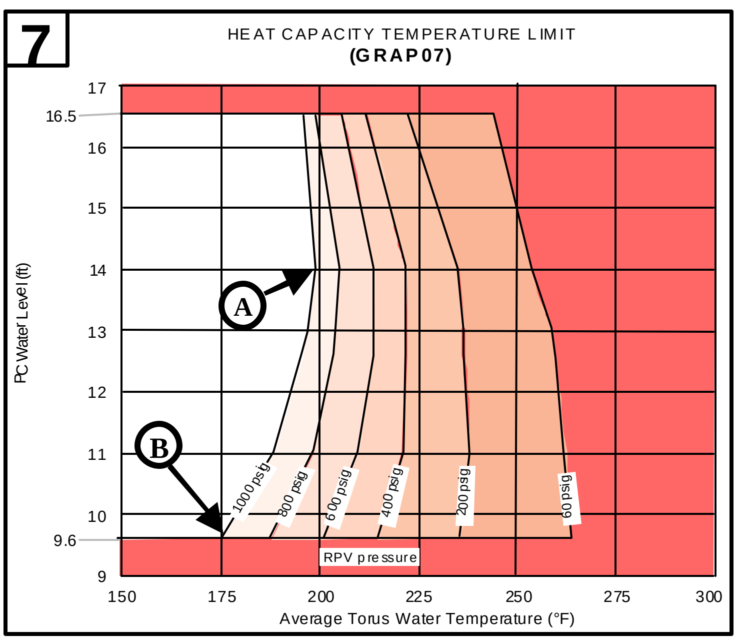
What is the bases for the shape of the Heat Capacity Temperature Limit (GRAPH 07) from point "A" to point "B"?

a. As torus level lowers, there is less backpressure on the SRV tailpipes. With lower SRV backpressure, a lower initial temperature will exceed analyzed blowdown rates during a LOCA.

b. As torus level lowers, there is less backpressure on the downcomers. With lower downcomer backpressure, a lower initial temperature will exceed peak design containment pressure during a LOCA blowdown.

c. Torus airspace volume increase has a greater effect on the limit than torus heat sink mass decrease. As torus airspace volume rises, a lower temperature is needed to heat and pressurize the airspace to PCPL-A.

d. Torus heat sink mass decrease has a greater effect on the limit than torus airspace volume increase. As torus level decreases, there is less mass available to absorb the blowdown energy from the RPV, requiring a more restrictive limit.



ANSWER: 14046

d. Torus heat sink mass decrease has a greater effect on the limit than torus airspace volume increase. As torus level decreases, there is less mass available to absorb the blowdown energy from the RPV, requiring a more restrictive limit.

Answer source: INT008-06-18 pp. 12 & 13

Distractors:

a. HCTL is based on not exceeding PCPL-A and has nothing to do with SRV backpressure.
b. HCTL is based on not exceeding PCPL-A and has nothing to do with downcomer backpressure.
d. Torus airspace volume increase as a lesser effect and acts to make a less restrictive limit rather than a more resrictive limit.

Source: ***New***

Provide to Candidate: EOP Graphs.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
67	5764	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0320101	CNS Administrative Procedures Volume 0, Administrative Procedures (Formal Classroom/Pre-OJT Training)

Related Objectives	
INT032010100G010I	Discuss the following as described in Administrative Procedure 0.26, Surveillance Program: Surveillance test authorization
INT032010100G010J	Discuss the following as described in Administrative Procedure 0.26, Surveillance Program: Surveillance test performance
INT032010100G010N	Discuss the following as described in Administrative Procedure 0.26, Surveillance Program: Surveillance Test Scheduling

Related References	
0.26	Surveillance Program

Related Skills (K/A)	
2.2.12	Knowledge of surveillance procedures. (CFR: 41.10 / 45.13) (3.0/3.4)

QUESTION: 5764

A Standby Gas Treatment operability surveillance test is due to be performed this shift.

Who is responsible for the coordination and authorization of performance of this test?

a. Shift Supervisor

b. Operations Manager

c. Operations Supervisor

d. Surveillance Coordinator

ANSWER: 5764

a. Shift Supervisor

Answer source: 0.26 p. 4, section 5

Distractors:

b. No direct responsibility for 0.26 actions.
c. Only responsible for arranging working load to ensure surveillance completed on time.
d. Only schedules surveillance.

2.2.12 Knowledge of surveillance procedures.   (CFR: 41.10 / 45.13) as it applies to: 261000 SGTS

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
68	12215	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0320102	CNS Administrative Procedures Site Services Procedures (Formal Classroom/Pre-OJT Training)

Related Objectives	
INT032010200C030A	Discuss the requirement associated with the following items: Visitor acces and departure
INT032010200C010B	Discuss the following as described in Administrative Procedure 1.15, Visitor/Tour Station Access: Vital Area access criteria

Related References	
1.15	Visitor/Tour Station Access

Related Skills (K/A)	
2.3.1	Knowledge of 10 CFR: 20 and related facility radiation control requirements. (CFR: 41.12 / 43.4. 45.9 / 45.10) (2.6/3.0)

QUESTION: 12215

Four vendor representatives were provided an escorted tour to perform a walkdown of the RHR system to assist in correcting an emergent problem. TLDs were not issued to the vendors. Following the walkdown, the following dose was received by each of the vendors on their DRDs:

- Rick Cox 106 mrem
- Ira Fox 98 mrem
- Marvin Estes 51 mrem
- Robert Smith 101 mrem

Which of the following are required?

An Exposure History Worksheet must be completed for . . .

a. **all** the visitors prior to departure.

b. **only** Rick Cox before departure.

c. **only** Rick Cox and Robert Smith before departure.

d. **only** Rick Cox, Robert Smith and Ira Fox before departure.

ANSWER: 12215

c. **only** Rick Cox and Robert Smith before departure.

If the dose received is > 100 mrem (1.0 mSv) by DRD, or equivalent, and no TLD was issued, Radiological Protection shall be contacted and a CNS-RP-10, Exposure History Worksheet, shall be completed before allowing the visitor to depart. Both these individuals received greater than 100 mrem and since TLDs were not issued, a CNS-RP-10, Exposure History Worksheet, shall be completed before allowing they depart.

Answer source: 1.15 page 4, step 4.6

Distractors:

a. Ira Fox and Marvin Estes received less than 100 mrem and are not required to complete CNS-RP-10, Exposure History Worksheet.
b. Robert Smith is also required to complete CNS-RP-10, Exposure History Worksheet prior to departure.
d. Ira Fox received less than 100 mrem and is not required to complete CNS-RP-10, Exposure History Worksheet.

Source: ***Direct from Bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
69	12209	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0320103	CNS Administrative Procedures Conduct of Operations and General Alarm Procedures (Formal Classroom/Pre-OJT Training)

Related Objectives	
INT032010300E010B	Discuss the following as described in Alarm Procedure 2.3.1, General Alarm Procedure: Alarm acknowledgement

Related References	
2.3.1	Procedure 2.3.1, General Alarm Procedure
OI-07	Operations Management Expectations

Related Skills (K/A)	
2.4.31	Knowledge of annunciators alarms and indications / and use of the response instructions. (CFR: 41.10 / 45.3) (3.3/3.4)

QUESTION: 12209

The plant is operating at 100% power. Surveillance Procedure 6.2PCIS.302, MAIN STEAM LINE HIGH TEMPERATURE CHANNEL FUNCTIONAL TEST 9 (DIV 2) has just commenced.

STEAM TUNNEL HIGH TEMP CHANNEL A (9-5-1/E-1) then alarms (the first annunciation of this alarm for the shift).

What action(s) is/are required?

Acknowledge the annunciator . . .

a. **only**.

b. **AND** announce "expected annunciator".

c. **AND** announce "STEAM TUNNEL HIGH TEMP CHANNEL A" **only**.

d. **AND** announce "STEAM TUNNEL HIGH TEMP CHANNEL A" **AND** pull the associated alarm card.

ANSWER: 12209

d. **AND** announce "STEAM TUNNEL HIGH TEMP CHANNEL A" **AND** pull the associated alarm card.

This annunciator is not associated with the surveillance and therefore the alarm should be acknowledged, announced and the alarm card pulled.

Answer source: 2.3.1 p. 3, steps 4.12, 4.13 & 4.14

Distractors:

a. This annunciator is not expected for this surveillance.
b. This annunciator is not expected for this surveillance.
c. The alarm card must be pulled for this annunciator.

Source: ***Direct from Bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
70	13673	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0320104	CNS Administrative Procedures General Operating Procedures (Startup and Shutdown) Procedures (Formal Classroom/Pre-OJT Training)

Related Objectives	
INT032010400G0300	State from memory the " Mitigating Task Scram Actions" associated with Procedure 2.1.5, Reactor Scram.

Related References	
2.1.5	Procedure 2.1.5, Reactor Scram

Related Skills (K/A)	
295006.AA2.05	Ability to determine and/or interpret the following as they apply to SCRAM: (CFR: 41.10 / 43.5 / 45.13) Whether a reactor SCRAM has occurred (4.6*/4.6*)

QUESTION: 13673

The plant was operating at full power when a reactor scram occurred.

Per 2.1.5, "Reactor Scram," which methods are to be utilized to determine that all control rods have been fully inserted into the core?

a. REFUEL MODE SELECT PERMISSIVE light ON **ONLY**.

b. The REFUEL MODE SELECT PERMISSIVE light is ON **OR** ***ALL*** green FULL-IN lights on the full core display are ON.

c. All green FULL-IN lights on the full core display are ON **AND** ***ALL*** rod positions indicating 00 on the PMIS RPIS display screen.

d. The REFUEL MODE SELECT PERMISSIVE light is ON **OR** ***ALL*** rod positions indicating 00 on the PMIS RPIS display screen.

ANSWER: 13673

b. The REFUEL MODE SELECT PERMISSIVE light is ON **OR** ***ALL*** green FULL-IN lights on the full core display are ON.

Answer source: 2.1.5 p. 5, Attachment 1, step 1.5

Distractors:

a. Utilizing the green full in lights on the full core display is also allowed.
c. The use of the RPIS PMIS display is not directed by procedure 2.1.5.
d. The use of the RPIS PMIS display is not directed by procedure 2.1.5

Source: ***Direct from Bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
71	16796	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
INT0320126 CNS Abnormal Procedures (RO) Cooling Water

Related Objectives	
INT0320126Q0Q0100	Given plant condition(s), and the applicable Abnormal/Emergency Procedure, determine the correct subsequent actions required to mitigate the event(s).

Related References	
2.4SDC	RHR Loss of Shutdown Cooling

Related Skills (K/A)	
295021.AA2.07	Ability to determine and/or interpret the following as they apply to LOSS OF SHUTDOWN COOLING: (CFR: 41.10 / 43.5 / 45.13) Reactor recirculation flow (2.9/3.1)

QUESTION: 16796

With the plant shutdown in Mode 4 and RHR loop "B" operating in Shutdown Cooling, the following conditions exist:

- Reactor pressure is 0 psig
- Recirc suction temperature is 170°F
- Reactor water level is 36" (NR)
- RHR pumps "A" and "C" have both motors disconnected from their pumps

Subsequently, "B" RHR Loop develops a leak, requiring the Control Room operators to remove RHR loop "B" from service.

What action is required, and for what reason?

a. RPV water level must be raised to > 48" to aid in natural circulation flow.

b. RPV water level must be raised to > 48" in order to maximize Reactor coolant contact with RPV metal for enhanced heat transfer to the Drywell atmosphere.

c. A Reactor Recirculation pump must be started to reduce the possibility of excessive thermal stresses on the CRD stub tubes.

d. A Reactor Recirculation pump must be started in order to reduce the possibility of thermal binding of the RHR-MO-25A/B valves caused by the expected coolant heatup.

ANSWER: 16796

a. RPV water level must be raised to > 48" to aid in natural circulation flow.

Answer source: 2.4SDC p. 10, Attachment 2, step 1.1

Distractors:

b. Water level is raised to enhance natural ciruclation flow, not heat transfer.
c. A Recirculation pump is started, but not to prevent thermal stresses on the stub tubes. Starting a recirc pump causes thermal stress on the stub tubes.
d. A Recirculation pump is started, but not to prevent thermal binding of the gate valves. These valves are cycled if closed during a cooldown.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
72	14426	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0320132	CNS Abnormal Procedures (RO) Off Gas/Vaccuum
COR0011402	OPS MAIN TURBINE

Related Objectives	
COR0011402001060D	Given a specific Main Turbine and Auxilary systems malfunction, determine the effect on any of the following: Condenser vacuum
INT0320132L0L0100	Given plant condition(s), and the applicable Abnormal/Emergency Procedure, determine the correct subsequent actions required to mitigate the event(s).
COR0011402001140E	Given a Main Turbine and Auxiliaries component manipulation, predict and explain the changes in the following parameters: Steam seal pressure

Related References	
2.4VAC	Loss of Condenser Vacuum
2.2.75	Procedure 2.2.75, Steam Sealing System

Related Skills (K/A)	
295002.AK2.11	Knowledge of the interrelations between LOSS OF MAIN CONDENSER VACUUM and the following: (CFR: 41.7 / 45.8) Seal steam: Plant-Specific. (2.6/2.7)

QUESTION: 14426

The plant is at 30% power with the following conditions:

- ➢ Main Condenser vacuum is currently 27"Hg and degrading
- ➢ SJAE air flow (AR-FR-47) has shown a large increase over the last several minutes
- ➢ Gland Steam supply pressure indicator (MS-PI-83) indicates 0 psig

What action will correct the problem?

a. Throttle ***Open*** MS-MO-BMV3, Steam Supply Bypass Valve.

b. Throttle ***Closed*** MS-MO-BMV3, Steam Supply Bypass Valve.

c. Throttle ***Open*** MS-MO-BMV4, Steam Unloader Bypass Valve.

d. Throttle ***Closed*** MS-MO-BMV4, Steam Unloader Bypass Valve.

ANSWER: 14426

a. Throttle ***Open*** MS-MO-BMV3, Steam Supply Bypass Valve.

Regardless of the valve alignment or supply, the correct action to take at this power level is to open the steam bypass from main steam to put pressure on the seals.

Answer source: 2.2.75 p. 3, step 4.19.2

Distractors:

b. Taking action to close BMV3 could only make a low pressure situation worse, and BMV3 is probably already closed to begin with.
c. BMV4 would be opened if pressure was too high and steam was leaking from the seals.
d. Throttling closed BMV4 could make a high pressure situation worse, if the steam unloaders did not have enough capacity to maintain pressure.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
73	4214	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0320134	OPS CNS Abnormal Procedures (RO) - Fire

Related Objectives	
INT0320134H0H0100	Given plant condition(s), and the applicable Abnormal/Emergency Procedure, determine the correct subsequent actions required to mitigate the event(s).
INT0320134D0D0100	Given plant condition(s), determine from memory the appropriate Abnormal/Emergency Procedure(s) to be utilized to mitigate the event(s).

Related References	
5.4.FIRE-SD	FIRE INDUCED SHUTDOWN FROM OUTSIDE CONTROL ROOM

Related Skills (K/A)	
2.4.34	Knowledge of RO tasks performed outside the main control room during emergency operations including system geography and system implications. (CFR: 43.5 / 45.13) (3.8/3.6)

QUESTION: 4214

A shutdown from outside the Control Room is in progress per 5.4FIRE-S/D.

How is Service Water flow established/controlled through RHR HX?

a. The Reactor Building operator manually operates SW-MO-89B from the handwheel.

b. The Reactor Building operator operates the open contacts at MCC-Y for SW-MO-89B.

c. The ASD Operator controls flow by the control switch for SW-MO-89B on the RHR panel in the ASD Room.

d. The Reactor Building operator fully opens SW-MO-89B at MCC-Y and the Control Building operator throttles on the SWBP discharge valve.

ANSWER: 4214

b. The Reactor Building operator operates the open contacts at MCC-Y for SW-MO-89B.

5.4FIRE-SD directs the operator to ensure breaker is closed and to rotate silver screw under switch which defeats mechanical interlock and open cubicle door. Then to remove control power fuse to prevent spurious operation, and to open valve by pressing button on LOWER contactor for time indicated for valve on the procedure attachment.

Answer source: 5.4FIRE-S/D p.22, step 1.1.6

Distractors:

a. 5.4.FIRE-SD directs the use of the open and closed contacts at MCC-Y for SW-MO-89B.
c. 5.4.FIRE-SD directs the use of the open and closed contacts at MCC-Y for SW-MO-89B.
d. 5.4.FIRE-SD directs the use of the open and closed contacts at MCC-Y for SW-MO-89B.

Source: ***Direct from Bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
74	5127	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
SKL0080101 WATCHSTANDING PRINCIPLES

Related Objectives	
SKL00801010011500	State what should be done if an out of limits reading is recorded on the logs or if an error is made recording a reading.

Related References	
1.9	Control and Retention of Records

Related Skills (K/A)	
2.1.18	Ability to make accurate / clear and concise logs / records / status boards / and reports. (CFR: 45.12 / 45.13) (2.9/3.0)

QUESTION: 5127

The RO realizes that an incorrect river level was transcribed by the Station Operator into 6.LOG.601, DAILY SURVEILLANCE LOG - MODES 1, 2, AND 3."

How is the the wrong river level corrected?

a. Correct number should be entered **AND** dated.

b. Circle the number, **AND** write in the correct number with an explanation.

c. Circle, initial **AND** date the old number, **AND** write in the correct number.

d. Draw one line through the incorrect old number, initial **AND** date, **AND** write in the correct number.

ANSWER: 5127

d. Draw one line through the incorrect old number initial **AND** date, **AND** write in the correct number.

Procedure 1.9 requires a single line to be drawn through the information to be corrected such that the information crossed out must remain legible. The person making the correction must initial and date the correction. Additional information may be added.

Distractors:

a. Required to draw a single line through correction, intial & date and write in the correct entry.
b. Action for a reading out of limit. Required to draw a single line through correction, intial & date and write in the correct entry.
c. Action for a reading out of limit. Required to draw a single line through correction, intial & date and write in the correct entry.

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

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
75	2167	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0010502 FIRE PROTECTION SYSTEM

Related Objectives	
COR0010502001130F	Briefly describe the following concepts as they apply to the Fire Protection system: Principle of operation of the smoke detectors
COR0010502001110F	Given plant conditions, determine if the following should occur: Initiation of Total Flooding High Pressure CO2 in associated Diesel Generator Room.

Related References	
2.2.2	Procedure 2.2.2, Carbon Dioxide Systems

Related Skills (K/A)	
286000.K5.07	Knowledge of the operational implications of the following concepts as they apply to FIRE PROTECTION SYSTEM: (CFR: 41.5 / 45.3) Smoke detection (2.6/2.7)

QUESTION: 2167

What conditions must be satisfied to energize the DG room #1 high pressure CO_2 system cylinder solenoid valve with DG room #1 and #2 MAIN-RESERVE switches in MAIN?

a. One of four smoke detectors trip **AND** one of one thermal detector trips (after a time delay).

b. One of four smoke detectors trip **AND** one of one thermal detector trips (no time delay).

c. Two of four smoke detectors trip **OR** one of one thermal detector trips (no time delay).

d. Two of four smoke detectors trip **OR** one of one thermal detector trips (after a time delay).

ANSWER: 2167

d. Two of four smoke detectors trip **OR** one of one thermal detector trips (after a time delay).

The system is automatically actuated by any two smoke detectors in the DG room, and/or by a thermal detector, set at 190°F, in the associated diesel fuel oil day tank room.

Answer source: 2.2.2 p. 16, step 1.2.3

Distractors:

a. System actuation would result from the thermal detector ONLY without any smoke detectors tripped. Any automaitic actuation is preceded by a time delay.
b. System actuation would result from the thermal detector ONLY without any smoke detectors tripped. Any automaitic actuation is preceded by a time delay.
c. Although the logic indicated would actuate the system, automatic actuations are preceded by a 50 sec time delay.

Source: ***Direct from Bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
76	1290	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0010602 FUEL POOL COOLING AND DEMINERALIZING SYSTEM

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

Related References	
2.2.32	Procedure 2.2.32, Fuel Pool Cooling and Demineralizer System
2.4FPC	Fuel Pool Cooling Trouble

Related Skills (K/A)	
233000.A1.07	Ability to predict and/or monitor changes in parameters associated with operating the FUEL POOL COOLING AND CLEAN-UP controls including: (CFR: 41.5 / 45.5) System temperature (2.7/2.8)

QUESTION: 1290

What is HIGHEST fuel pool temperature expected under **maximum normal** heat load?

a. 110 °F.

b. 125 °F.

c. 150 °F.

d. 160 °F.

ANSWER: 1290

b. 125 °F.

Answer source: 2.2.32 p. 45, step 1.2.15

Distractors:

a. This is the upper end of the administrative limit for fuel pool temperature during normal operation.

c. This is the upper end of the expected fuel pool temperature with the core offloaded and the FPC system assisted by RHR.

d. This temperature is above the expected maximum for all FPC operations.

Source: ***Direct from Bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
77	5092	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0010602 FUEL POOL COOLING AND DEMINERALIZING SYSTEM

Related Objectives	
COR0010602001070B	Given a specific FPC malfunction, determine the effect on any of the following: Pool/Rx Well Water Level

Related References	
2.4FPC	Fuel Pool Cooling Trouble
2.2.32	Procedure 2.2.32, Fuel Pool Cooling and Demineralizer System

Related Skills (K/A)	
295023.AA1.02	Ability to operate and/or monitor the following as they apply to REFUELING ACCIDENTS: (CFR: 41.7 / 45.6) Fuel pool cooling and cleanup system (2.9/3.1)

QUESTION: 5092

Given the following conditions:

- ➢ The Fuel Pool Cooling (FPC) System is operating with one pump **AND** one heat exchanger in service
- ➢ The Fuel Pool Gates are installed
- ➢ A leak occurs on the outlet pipe of the Skimmer Surge Tanks
- ➢ **NO** operator action is taken

What effect will these conditions have on FPC System cooling capability **AND** Fuel Pool water level?

FPC System cooling capability . . .

a. **AND** Fuel Pool water level will be unchanged.

b. will be lost **AND** Fuel Pool water level will continuously lower.

c. will be lost **AND** Fuel Pool water level will lower slightly **AND** stabilize.

d. will be unchanged **AND** Fuel Pool water level will lower slightly **AND** stabilize.

ANSWER: 5092

c. will be lost **AND** Fuel Pool water level will lower slightly **AND** stabilize.

Water from the pool overflows the weirs into the Skimmer Surge Tanks and is then pumped from the Skimmer Surge Tanks back to the pool. Since makeup to the system is manual, a leak will drain the tanks. Low level in the tanks will trip the pumps . With no water being pumped to the pool, level will lower until it reaches the top of the weirs and then stabilize.

Answer source: 2.4FPC p. 1, step 2.1

Distractors:

a. The pump trips and pool level will stabilize at the top of the weirs.
b. Pool level will stabilize at the top of the weirs.
d. The pump trips.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
78	3724	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
SKL0124108	HEATING, VENTILATION, AIR CONDITIONING
COR0010802	OPS HEATING, VENTILATION AND AIR CONDITIONING

Related Objectives	
SKL012410800A030E	Given plant conditions, predict changes in the following: Starting/stopping of fans
COR0010802001160D	Predict the consequences a malfunction of the following would have on the Control Room HVAC system: Fire protection

Related References	
2.2.84	Procedure 2.2.84, HVAC Main Control Room and Cable Spreading Room

Related Skills (K/A)	
600000.AA1.05	Ability to operate and / or monitor the following as they apply to PLANT FIRE ON SITE: Plant and control room ventilation systems (3.0/3.1)

QUESTION: 3724

A fire has occurs in the Cable Spreading Room with the plant operating at 100% power. The fired does not spread beyond the Cable Spreading Room.

What impact does the fire have on the Control Room Ventilation System?

a. Fire Dampers **DO NOT** isolate the Control Room. The Supply Fans **DO NOT** trip. The Emergency Bypass Train starts and supplies the Control Room with outside air.

b. Fire Dampers **DO NOT** isolate the Control Room. The Supply Fans trip. The Emergency Bypass Train does **NOT** start.

c. Fire Dampers isolate the Control Room. The Supply Fans trip. The Emergency Bypass Train does **NOT** start.

d. Fire Dampers isolate the Control Room. The Supply Fans **DO NOT** trip. The Emergency Bypass Train starts and supplies the Control Room with outside air.

ANSWER: 3724

c. Fire Dampers isolate the Control Room. The Supply Fans trip. The Emergency Bypass Train does **NOT** start.

<u>Answer source:</u> COR001-08-01 p. 63
COR001-08-01 figure 23
2.2.84 p. 25, Attachment 4

Distractors:

a. The control room is isolated. The supply fans trip. The emergency bypass train does not start.
b. The control room is isolated..
d. Control room, not Cable Spreading room. The supply fans do trip. The Bypass Train does not start.

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

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
79	14041	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0011602 Off Gas

Related Objectives	
COR0011602001090D	Explain the following Off Gas system related concepts: Charcoal absorption of fission product gases
COR0011602001010O	State the purpose of the following items related to Off Gas system: Charcoal Absorber Beds

Related References	
2.2.58	Procedure 2.2.58, AOG System

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

QUESTION: 14041

Why are the Off-gas system charcoal beds maintained at sub-freezing temperatures?

The low temperatures . . .

a. freeze any remaining moisture in the Off-gas stream to prevent it's intrusion onto the charcoal.

b. increase the adsorption coefficients of Krypton and Xenon increasing selective adsorption and retention.

c. reduces the relative humidity of the Off-gas stream in the charcoal beds increasing adsorption of Krypton and Xenon.

d. increases density of the gases in the Off-gas stream, thereby reducing the volume and increasing the stream contact time with the charcoal bed.

ANSWER: 14041

b. increase the adsorption coefficients of Krypton and Xenon increasing selective adsorption and retention.

A lower than ambient operating temperature of 0°F is selected as the adsorption coefficients (K) of krypton and xenon increase with a decrease in temperature.

Answer source: COR001-16-01, p. 36, section 4

Distractors:

a. Freezing water in or on the charcoal bed would not improve its ability to adsorb Xe and Kr. In addition the relative humidity of the off gas stream should be sufficiently below the temperature of the charcoal beds.

c. The relative humidity of the off-gas stream is controlled by components upstream of the charcoal beds.

d. Density changes (and therefore volume) alone would not significantly affect the flow rate through the charcaol beds.

2.1.28 Knowledge of the purpose and function of major system components and controls. as it applies to 271000 Offgas System

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
80	3973	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0011602 Off Gas

Related Objectives	
COR0011602001080G	Describe the Off Gas system design feature(s) and/or interlock(s) that provide for the following: Automatic system isolation

Related References	
2.3_B-3	PANEL B - ANNUNCIATOR B-3

Related Skills (K/A)	
256000.K4.10	Knowledge of REACTOR CONDENSATE SYSTEM design feature (s) and/or interlocks which provide for the following: (CFR: 41.7) Non-condensable gas removal (2.7/2.7)

QUESTION: 3973

The plant was operating at power when an explosion occurred in the Off-Gas system. The SJAE suction valves automatically closed. The Off-Gas system sustained no damage during the transient and the ERP monitors spiked but immediately returned to normal. Hold up pipe temperature **AND** pressure also returned to normal values.

Which of the following describes the process that reopens the SJAE suction valves?

The Off-Gas High Pressure AND Temperature signals ____________ , and the SJAE suction valves __________________.

a.	reset automatically	are manually reopened
b.	reset automatically	automatically reopen
c.	must be reset manually	are manually reopened
d.	must be reset manually	automatically reopen.

ANSWER: 3973

c.	must be reset manually	are manually reopened

The isolation valves are re-opened manually after the isolation signal is clear and the logic is reset using either the SJAE Suction Isol Reset pushbutton on Control Room Panel B or Local Panel 1R-1E.

Answer source: COR001-16-01, p. 20, section 6.b
COR001-16-01, p. 56, Main Control Room Controls Table

Distractors:

a. The isolation signals do not reset automatically.
b. The isolation signals do not reset automatically and the SJAE suction valves do not open automatically.
d. The SJAE suction valves do not reopen automatically.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
81	3085	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0020102 AVERAGE POWER RANGE MONITOR SKL0124201 AVERAGE POWER RANGE MONITOR SYSTEM

Related Objectives	
COR0020102001080C	Describe the APRM design feature(s) and/or interlock(s) that provide for the following: Alarm seal-in
COR0020102001110E	Given an Average Power Range Monitor System control manipulation, predict the changes in the following parameters: Lights and alarms
SKL012420100A030F	Given plant conditions, predict changes in the following APRM system component/parameters: Lights/alarms.
SKL012420100B030F	Manually operate the APRM system to control the following: Lights/alarms.

Related References	
6.1APRM.303	APRM System Channel Functional Test (Mode Switch In Run/Div 1)

Related Skills (K/A)	
215005.A2.01	Ability to (a) predict the impacts of the following on the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM; and (b) based on those predictions, use procedures to correct, control, or...: (CFR: 41.5/45.6) Power supply degraded (2.7/3.1)

QUESTION: 3085

The plant was operating at rated power when the power supply fuses for "B" APRM blew. The fuses have been replaced and the 1/2 scram has been reset.

Given that annunciator 9-5-1/B-8 **APRM UPSCALE** clears, what actions would the operator need to perform (if any) to clear the alarms on panel 9-14 **AND** clear the alarms on panel 9-5?

The alarms lights on panel 9-14 _________________ and the alarm lights on panel 9-5 benchboard _________________.

a.	clear automatically	clear automatically
b.	clear automatically	must be reset from 9-5
c.	must be reset from 9-14	clear automatically
d.	must be reset from 9-14	must be reset from 9-5

ANSWER: 3085

c.	must be reset from 9-14	clear automatically

Answer source: 6.1APRM.303, p. 5, step 4.19
COR002-01-02, p. 20, Control Room Controls table
COR002-01-02, p. 14, section a)
COR002-01-02, p. 14, section 2.

Distractors:

a. The alarms lights on panel 9-14 must be reset from 9-14.
b. The alarms lights on panel 9-14 must be reset from 9-14. The alarms lights on panel 9-5 clear automatically.
d. The alarms lights on panel 9-5 clear automatically.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
82	18311	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0020202 OPS CONDENSATE AND FEED

Related Objectives	
COR0020202001120C	Given plant conditions, determine if: Minimum Flow Valves should have repositioned

Related References	
2.2.28.1	Procedure 2.2.28.1, Feedwater System Operation
2.2.28	Procedure 2.2.28, Feedwater System Startup And Shutdown

Related Skills (K/A)	
259001.K4.03	Knowledge of REACTOR FEEDWATER SYSTEM design feature (s) and/or interlocks which provide for the following: (CFR: 41.7) RFP minimum flow (2.7/2.7)

QUESTION: 18311

A plant startup is in progress, and RFP 1A is the second RFP to be started. As RFP is being placed in service, RFP 1A flow quickly rises from less than 1000 gpm to approximately 3000 gpm, but then, after approximately 1 minute, RFP 1A experiences excessive vibration and flow is lowered, stabilizing at approximately 100 gpm.

What describes the expected RFP 1A minimum flow valve response?

The RFP minimum flow valve _____________ as flow rises; then, the minimum flow valve _____________ as flow lowers.

a.	automatically closes	automatically reopens
b.	automatically closes	must be manually reopened
c.	must be manually closed	automatically reopens
d.	must be manually closed	must be manually reopened

ANSWER: 18311

c.	must be manually closed	automatically reopens

Each Reactor Feed pump (RFP) is equipped with an air operated minimum flow valve which will automatically open if pump flow decreases to approximately 2000 gpm. These valves have no automatic closing feature and must be closed, using the minimum flow c/s on Panel A, when pump discharge flow is greater than 2000 gpm.

Answer source: COR002-02-02, p. 30, section 5

Distractors:

a. The minimum flow valve doesn't automatically close as flow increases.
b. The minimum flow valve does not close automatically as flow increases and the valve automatically reopens if flow decreases.
d. The minimum flow valve automatically reopens on decreasing flow.

Source: *Direct from bank*

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
83	14053	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0020202 OPS CONDENSATE AND FEED

Related Objectives	
COR0020202001080A	Predict the consequences a malfunction of the following would have on the Condensate and Feedwater system: Plant Air

Related References	
5.2AIR	Loss of Instrument Air

Related Skills (K/A)	
300000.K3.02	Knowledge of the effect that a loss or malfunction of the (INSTRUMENT AIR SYSTEM) will have on the following: (CFR: 41.7 / 45.6) Systems having pneumatic valves and controls (3.3/3.4)

QUESTION: 14053

Given the following conditions:

- ➢ The plant is operating at 100% power
- ➢ The air supply line to COND BOOSTER P B MIN FLOW, MC-AOV-FCV10 is severed
- ➢ **NO** operator action is taken

How will the Feed System respond to these conditions over the next one (1) minute?

"B" Condensate Booster Pump Minimum Flow Valve (MC-AOV-FCV10) fails . . .

a. closed. The speed of **BOTH** RFPs rise.

b. closed. The speed of **BOTH** RFPs remains unchanged.

c. open. The speed of **BOTH** RFPs rise.

d. open. The speed of **BOTH** remains unchanged.

ANSWER: 14053

c. open. The speed of **BOTH** RFPs rise.

MC-AOV-FCV10 fails open on loss of air (see attachment 1 of 5.2AIR). When the valve opens, feed flow will lower causing a FF-SF mismatch. RFP speed will rise until FF=SF and level is at 35" or the Reactor scrams on low level.

Answer source: COR002-02-02, p. 81, section 4)

Distractors:

a. MC-AOV-FCV10 fails open on loss of air.
b. MC-AOV-FCV10 fails open on loss of air.
d. When the valve opens, feed flow will lower causing a FF-SF mismatch. RFP speed will rise until FF=SF and level is at 35".

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
84	3302	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0020302 CONTAINMENT

Related Objectives	
COR0020302001130D	Describe the PCIS design features and/or interlocks that provide for the following: Bypassing of selected isolations
COR0020302001130E	Describe the PCIS design features and/or interlocks that provide for the following: Operator action to defeat/reset isolations
COR0020302001170A	Predict the consequences of the following items on Primary containment: LOCA
COR0020302001210C	Given plant conditions, determine if the following should have occurred: Drywell cooling fan trip.

Related References	
2.2.40	Procedure 2.2.40, HVAC Drywell Cooling

Related Skills (K/A)	
223001.A3.03	Ability to monitor automatic operations of the PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES including: (CFR: 41.7 / 45.7) System indicating light and alarms (3.4/3.3)

QUESTION: 3302

A small break LOCA has occurred with the following conditions:

➢ Reactor water level	+45" (stable NR)
➢ Reactor pressure	560 psig (lowering at 10 psig per minute)
➢ Drywell pressure	3.1 psig (rising at 0.1 psig per minute)
➢ Drywell temperature	195°F (rising at 2°F per minute)
➢ "C" Drywell FCU control switch	RUN

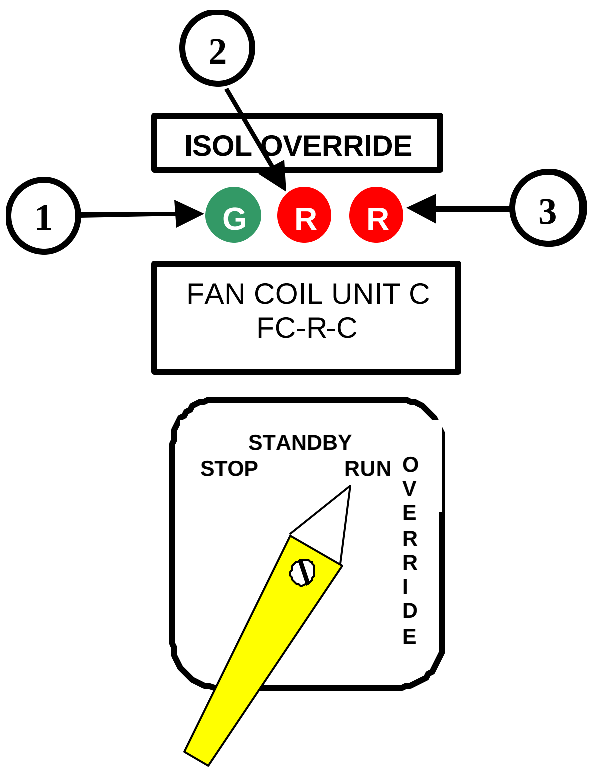
Which lamp(s) is/are illuminated (1, 2 or 3) for "C" Drywell Fan Coil Unit (FCU) and what actions are required for "C" Drywell FCU to operate 10 minutes from now?

a. Lamp #1 is illuminated. Place FCU control switch in OVERRIDE.

b. Lamp #1 is illuminated. Install jumpers to bypass the Core Spray injection valve low pressure permissive signal.

c. Lamp #2 is illuminated. Place FCU control switch in OVERRIDE.

d. Lamp #2 and #3 are illuminated. Install jumpers to bypass the high drywell pressure signal and place FCU control switch in OVERRIDE.



ANSWER: 3302

a. Lamp #1 is illuminated. Place FCU control switch in OVERRIDE.

<u>Answer source:</u> COR002-03-02, p. 26

Distractors:

b. The core spray low pressure permissive signal does not affect FCU logic, just the injection valves.
c. The FCU is not running.
d. The FCU is not in override and is not running. No need to bypass the high drywell pressure signal.

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

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
85	14038	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

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

Related Objectives	
COR0020402001120C	Given a specific CRDH system malfunction, determine the effect on any of the following: Control rod drive mechanisms (CRDMs)
COR0020402001120D	Given a specific CRDH system malfunction, determine the effect on any of the following: Reactor water cleanup pumps
COR0020502001090A	Predict the consequences a malfunction of the following would have on the CRDMs: Loss of CRDH Pumps
COR0020402001120A	Given a specific CRDH system malfunction, determine the effect on any of the following: Recirculation pumps

Related References	
2.2.8	Procedure 2.2.8, Control Rod Drive System
2.4CRD	CRD TROUBLE

Related Skills (K/A)	
201003.K6.01	Knowledge of the effect that a loss or malfunction of the following will have on the CONTROL ROD AND DRIVE MECHANISM: (CFR: 41.7 / 45.7) Control rod drive hydraulic system (3.3/3.3)

QUESTION: 14038

A plant startup is in progress with reactor pressure at 926 psig and reactor power at 8%. The "A" CRD pump trips and cannot be restarted. All attempts by the crew to start "B" CRD pump are also unsuccessful.

If operation were to continue with these conditions, which of the following would result?

a. reduced CRDM seal life

b. inability to scram control rods

c. reactor recirculation (RR) pump seal failure

d. exceed the acceptable pressure temperature range for RWCU pump operation

ANSWER: 14038

a. reduced CRDM seal life.

The loss of both CRD pumps results in a loss of cooling to the CRDMs Operation for long periods at high temperatures will reduce the life of the CRDM seals.

Answer source: COR002-04-02, p.31, section b

Distractors:

b. With reactor pressure above 900 psi normal scram times will be met even if accumulator becomes reduced by the loss of CRDH pumps.
c. Long term the seal life of the RR pump may be shortened but the seal should remain sufficiently cooled by REC to the jacket around the seal.
d. Loss of CRDH would not impact reactor pressure or RWCU inlet temperature.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
86	14496	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0020702 OPS DC ELECTRICAL DISTRIBUTION

Related Objectives	
COR0020702001080D	Given a specific DC Electrical Distribution system malfunction, determine the effect on any of the following: Battery chargers

Related References	
3058	DC One Line Diagram
2.2.24.1	250 VDC ELECTRICAL SYSTEM (DIV 1)

Related Skills (K/A)	
295004.AA1.01	Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER: (CFR: 41.7 / 45.6) D.C. electrical distribution systems (3.3/3.4)

QUESTION: 14496

The plant is at 100% power with RCIC in full-flow test mode when annunciator C-1/C-1, 250 VDC BATT CHARGER 1A TROUBLE alarms.

- ➢ CRT alarm message indicates:

  (3708) 250 VDC BATTERY CHARGER 1A DC VOLTAGE HIGH (in and reset)
  (3707) 250 VDC BATTERY CHARGER 1A AC VOLTAGE FAILURE (in and reset)
  (3709) 250 VDC BATTERY CHARGER 1A DC VOLTAGE LOW

The following 250 VDC indications are observed:

- ➢ 250 VDC Bus 1A indicates 250 volts and stable
- ➢ 250 VDC Battery 1A indicates 75 amps out and stable
- ➢ 250 VDC Charger 1A indicates 0 amps

What is the status of 250 VDC electrical distribution?

a. The AC input breaker on the 250V charger has tripped open automatically.

b. The DC output breaker on the 250V charger has tripped open automatically.

c. The 150 amp fuse on the feeder to the 250 VDC RCIC Starter Rack has blown.

d. The 300 amp fuse on the feeder from 250 VDC BATT CHARGER 1A has blown.

ANSWER: 14496

a. The AC input breaker on the 250V charger has tripped open automatically. DC output over voltage causes the AC input breaker on a 250V CHARGER to trip.

Answer source: COR002-07-02, p. 15, section 2

Distractors:

b. The DC output breaker does NOT automatically trip open.

c. If the 150 amp fuse on the feeder to the 250 VDC RCIC Starter Rack had blown, Annunciator C-1/A-1, 250 VDC SWGR BUS 1A BLOWN FUSE would have alarmed and CRT alarm message would indicate (3703) RCIC Starter Rack normal feeder; also, battery 1A would not indicate being loaded (75 amps and stable) as the RCIC gland seal vacuum pump and condensate pump are currently the only energized loads on 250 VDC Bus 1A.

d. If the 300 amp fuse on the feeder from 250 VDC BATT Charger 1A had blown, Annunciator C-1/A-1, 250 VDC SWGR BUS 1A BLOWN FUSE would have alarmed and CRT alarm message would indicate (3702) Feeder from Battery Charger A.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
87	14034	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0020802 DIESEL GENERATORS

Related Objectives	
COR0020802001110F	Predict the consequences a malfunction of the following would have on the Diesel Generators: DC Power

Related References	
COR0020802	Diesel Generators

Related Skills (K/A)	
264000.K1.02	Knowledge of the physical connections and/or cause- effect relationships between EMERGENCY GENERATORS (DIESEL/JET) and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) D.C. electrical distribution (3.3/3.4)

QUESTION: 14034

The plant was operating at 100% power when a loss of off-site power occurred. Both diesel generators automatically started and loaded their respective busses. An electrical fault results in the loss of 125 VDC panel DG1.

How is DG1 affected and why?

DG1 . . .

a. stops due to loss of power to the fuel oil booster pump.

b. stops due to loss of power to the the electronic trip solenoid.

c. continues to run but the day tank cannot be refilled due to loss of power to transfer pumps.

d. continues to run but the fuel oil booster pump is unavailable should the engine driven pump fail to develop sufficient fuel oil pressure.

ANSWER: 14034

b. stops due to loss of power to the the electronic trip solenoid.

Since both the electronic trip valve (20SD) energizes on the auto start to position the fuel control cylinder to the "FUEL ON" position, a loss of 125 VDC to this valve causes the Diesel Generator to trip.

Answer source: COR002-08-02, p. 26

Distractors:

a. Although a loss of power to the fuel oil booster pump has occurred this is not the reason the DG stops. The engine driven pump would be sufficient to supply the fuel oil needs of the DG.
c. The DG does not continue to run.
d. Although the fuel oil booster pump is in fact unavailable to backup the engine driven fuel oil pump, the DG does not continue to run due to the loss of power to the electronic trip valve.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
88	3286	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0021502 NUCLEAR BOILER INSTRUMENTATION

Related Objectives	
COR0021502001020F	Describe the interrelationships between NBI and the following: Main Turbine/Feedwater
COR0021502001060E	Given a specific NBI malfunction, determine effect on any of the following: Main turbine and feedwater
COR0021502001040A	Briefly describe the following concepts as they apply to NBI: Vessel level measurement
COR0021502001050A	Predict the consequences of the following on the NBI: Detector equalizing valve leaks

Related References	
4.6.1	Procedure 4.6.1, Reactor Vessel Water Level Indication

Related Skills (K/A)	
216000.K1.16	Knowledge of the physical connections and/or cause- effect relationships between NUCLEAR BOILER INSTRUMENTATION and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) Main turbine (3.0/3.1)

QUESTION: 3286

The plant is operating at 100% power with NBI-LT-52C level transmitter (Narrow Range Reactor Water level instrument) failed upscale. NBI-LT-52B level transmitter is selected for level control.

Prior to isolating the NBI-LT-52C level transmitter from service for maintenance, the equalizing valve for NBI-LT-52C is inadvertently opened by I&C.

Assume **NO** operator actions are taken.

Which one of the following describes **ONLY** the effects **DIRECTLY** produced by these conditions?

a. A high reactor water level alarm is received.

b. A full scram is received on a low RPV water level signal.

c. The RFPs and the Main Turbine will trip on a high RPV water level signal.

d. A ½ scram is received on RPS trip system "A" due to a low RPV water level signal.

ANSWER: 3286

c. The RFPs and the Main Turbine will trip on a high RPV water level signal.

Opening LT-52C equalization valve will drain the reference leg and will result in "A" channel indicating upscale as well, giving a high level trip to the RFPT and the MT with the subsequent full scram.

Answer source: COR002-15-02, p. 44, section 6
COR002-15-02, p. 35, Interlocks and Trips

Distractors:

b. A high water level alarm will not occur as "B" is selected.
c. The scram will be due to the turbine trip.
d. A full scram is received on low level.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
89	1238	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0022002 OPS REACTOR MANUAL CONTROL SYSTEM

Related Objectives	
COR0022002001150E	Given plant conditions related to RMCS and/or RPIS, determine if any of the following should occur: Control rod drift alarm

Related References	
2.1.5	Procedure 2.1.5, Reactor Scram

Related Skills (K/A)	
214000.A2.02	Ability to (a) predict the impacts of the following on the ROD POSITION INFORMATION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those...: (CFR: 41.5 / 45 Reactor SCRAM (3.6/3.7)

QUESTION: 1238

The plant was operating at rated power when a manual reactor scram was inserted. The following conditions exist:

- All control rods have fully inserted
- The reactor mode switch has been placed in SHUTDOWN
- The scram has not been reset

What are the **MINIMUM** actions necessary to reset the control rod drift indications on the full core display (vertical section of panel 9-5)?
(NOTE: The choices are arranged in MIMIMUM to MAXIMUM order.)

a. Momentarily rotate ROD DRIFT ALARM TEST switch to RESET.

b. Reset the reactor scram. After control rods have settled at position 00, momentarily rotate ROD DRIFT ALARM TEST switch to RESET.

c. Reset the reactor scram. Select each control rod with a drift alarm. After control rod has settled at position 00, momentarily rotate ROD DRIFT ALARM TEST switch to RESET. Repeat for each control rod with a drift alarm.

d. Select each control rod with a drift alarm. Momentarily place ROD MOVEMENT CONTROL switch to OUT NOTCH. After control rod has settled at position 00, momentarily rotate ROD DRIFT ALARM TEST switch to RESET. Repeat for each control rod with a drift alarm.

ANSWER: 1238

b. Reset the reactor scram. After control rods have settled at position 00, momentarily rotate ROD DRIFT ALARM TEST switch to RESET.

Answer source: 2.1.5 p. 4, step 5.10

Distractors:

a. The scram must be reset.
c. The control rods do not need to be selected.
d. The scram must be reset and do not need to notch each rod.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
90	2816	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0022302 RESIDUAL HEAT REMOVAL

Related Objectives	
COR0022302001030N	Describe RHR System design feature(s) and/or interlocks which provide for the following: Prevention of leakage to the environment through system heat exchanger
COR0022302001040G	Describe the interrelationship between the RHR system and the following: RHR Service Water
COR0022302001080R	Predict the consequences a malfunction of the following will have on the RHR system: RHR Service Water

Related References	
2.3_9-3-1	Panel 9-3 - Annunciator 9-3-1

Related Skills (K/A)	
205000.K6.08	Knowledge of the effect that a loss or malfunction of the following will have on the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE): (CFR: 41.7 / 45.7) RHR service water: Plant-Specific (3.5/3.7)

QUESTION: 2816

The plant is in the process of cooling down. The plant shutdown was required due to fuel failure. The following conditions exist:

- ➢ "A" RHR pump is in Shutdown Cooling
- ➢ RPV water level is + 50" (NR)
- ➢ RPV pressure is 20 psig use Main Turbine Bypass Valves
- ➢ It is determined that tubes in the "A" RHR Heat Exchanger have completely failed

What is the consequence of continuing to operate "A" RHR pump in Shutdown Cooling?

a. RPV pressure will rise.

b. RPV water level will rise.

c. Radioactive water will be released to the environment.

d. Radioactive materials will be deposited in Reactor Building sumps.

ANSWER: 2816

b. RPV water level will rise.

Answer source: COR002-23-02, p. 22, section 4

Distractors:

a. Service water flow into the RPV will lower RPV pressure. The RPV will never go solid with the MSIVs open (BPVs in use for pressure control requires MSIVs be open).
c. Service water pressure is higher than RHR pressure.
d. Service water pressure is higher than RHR pressure, there is no release path to the Reactor Building.

Source: ***Modified***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
91	19100	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
COR0022302	RESIDUAL HEAT REMOVAL
SKL0124223	OPS RESIDUAL HEAT REMOVAL

Related Objectives	
SKL012422300A0200	Explain the Residual Heat Removal system limitations and precautions as stated in the SOP 2.2.69, SOP 2.2.69.1, SOP 2.2.69.2 and SOP 2.2.69.3.
COR0022302001050B	Briefly describe the following concepts as they apply to the RHR system: Valve operation

Related References	
2.2.69.3	Procedure 2.2.69.3, RHR Suppression Pool Cooling And Containment Spray
2.0.1	Procedure 2.0.1, Plant Operations Policy

Related Skills (K/A)	
203000.A4.02	Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) System valves (4.1*/4.1*)

QUESTION: 19100

When operating RHR-MO-66B, B HX BYPASS VLV, what precaution applies and what is the bases for this precaution?

a. Do **NOT** hold the control switch in OPEN after the **red** indicating light turns **ON** to prevent tripping the valve breaker.

b. Do **NOT** hold the control switch in CLOSE after the **green** indicating light turns **ON** to prevent hammering the valve.

c. Hold the control switch in CLOSE for 5 seconds after the **red** indicating light turns **OFF** to ensure the valve closure is terminated by torque switch.

d. Hold the control switch in OPEN for 3 seconds after the **green** indicating light turns **OFF** to ensure the valve is properly backseated.

ANSWER: 19100

c. Hold the control switch in CLOSE for 5 seconds after the **red** indicating light turns **OFF** to ensure the valve closure is terminated by torque switch.

RHR-MO-66B, B HX BYPASS VLV is a throttle valve and throttle valves are held in closed an additional 5 seconds to ensure they are closed by their torque switch.

Answer source: 2.0.1 p. 6, step 9.1

Distractors:

a. This is based on the precaution from the procedure on seal-in limitorque operators.
b. This is based on the precaution from the procedure on seal-in limitorque operators.
d. The valve is held for 5 seconds (3 seconds is time delay for reversing directions) to prevent terminating the closure with the close limit switch.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
92	5401	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0022402 ROD BLOCK MONITOR

Related Objectives
COR0022402001100A State the electrical power supplies to the following: RBM channels.

Related References	
4.1.5	Procedure 4.1.5, Rod Block Monitor System

Related Skills (K/A)	
215002.K6.01	Knowledge of the effect that a loss or malfunction of the following will have on the ROD BLOCK MONITOR SYSTEM: (CFR: 41.7 / 45.7) RPS: BWR-3,4,5 (3.0/3.2)

QUESTION: 5401

Given the following conditions:

- ➢ The plant is operating at 88% power
- ➢ RBM A was bypassed for repairs six (6) hours ago
- ➢ Rod 18-19 is selected
- ➢ RPS Panel PP1B power to the RBM System is lost

What effect does this have on the RBM system?

a. RBM A is deenergized and has initiated a rod block.

b. RBM A is deenergized and **NO** rod block is present.

c. RBM B is deenergized and has initiated a rod block.

d. RBM B is deenergized and **NO** rod block is present.

ANSWER: 5401

c. RBM B is deenergized and has initiated a rod block.

The breaker trip deenergizes RBM B. When RBM B deenergizes it generates a rod block.

Answer source: COR002-24-02 p. 32, section B.1

Distractors:

a. RBM A remains energized and is currently bypassed and no rod block from RBM A is present.
b. RBM A remains energized.
d. When RBM B deenergizes, it generates a rod block.

Source: Direct from bank

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
93	5070	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0022902 STANDBY LIQUID CONTROL

Related Objectives	
COR0022902001100G	Predict the consequences a malfunction of the following would have on the SLC system: Tank Heaters

Related References	
2.2.74	Procedure 2.2.74, Standby Liquid Control System

Related Skills (K/A)	
211000.K5.07	Knowledge of the operational implications of the following concepts as they apply to STANDBY LIQUID CONTROL SYSTEM: (CFR: 41.5 / 45.3) Tank heater operation (2.7/2.9)

QUESTION: 5070

Loss of the SLC Storage tank heaters can result in . . .

a. increased concentration of tank SLC solution.

b. SLC pump suction piping temperature decreases.

c. SLC solution temperature decreasing below saturation.

d. indicated tank level decreasing to less than actual level.

ANSWER: 5070

c. SLC solution temperature decreasing below saturation.

The loss of the SLC Storage Tank heater would result in a decreasing temperature of the SLC solution. This reduction in temperature could eventually result in SLC solution decresing to below saturation for the concentration in the tank.

Answer source: COR002-29-02 p. 19, section 3

Distractors:

a. The concentration of tank SLC solution should remain unchanged until the temperature goes below saturation, once it is below saturation temperature the sodium pentaborate could come out of solution resulting in a decreased concentration of the remaining liquid solution.
b. SLC pump suction piping temperature should remain relatively constant as the heat trace on the pump suction piping is unaffected.
d. Indicated tank level would not experience any appreciable effects. Even though the temperature of the solution would decrease, the mass (and therefore the weight of the solution) above the bubbler would remain constant. The decrease in temperature would decrease actual level as the density of the fluid increased.

Source: ***Direct from Bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
94	16441	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0070510	CNS Tech. Spec. 3.9, Refueling Operations
INT0070501	OPS Introduction to Technical Specifications

Related Objectives	
INT0070501001030A	From memory, define the following terms: Core Alteration
INT00705100010100	Given a set of plant conditions, recognize non-compliance with a Section 3.9 LCO.

Related References	
1.1	Definitions
3.9	Refueling Operations

Related Skills (K/A)	
2.2.27	Knowledge of the refueling process. (CFR: 43.7 / 45.13) (2.6/3.5)

QUESTION: 16441

Refueling operations are in progress with the reactor vessel head removed and a partial load of fuel is in the vessel. Shutdown margin check has been performed.

What action is a CORE ALTERATION?

a. Install a control rod blade into an empty cell.

b. Drive a Source Range Monitor detector to full in.

c. Perform a friction test on a control rod in a loaded cell.

d. Insert the LPRM Instrument Handling Tool below the top guide.

ANSWER: 16441

c. Performing a friction test on a control rod in a loaded cell.

Core alteration includes movement of any reactivity controlling component with the exceptions specified: SRM movement is an exception, control rod movement if there is no fuel in the associated core cell is an exception. The LPRM instrument handling tool is not a reactivity control component.

Answer source: Technical Specifications definitions, p. 1.1-2

Distractors:

a. Control rod movement provided there are no fuel assemblies in the associated core cell is not considered to be a CORE ALTERATION.
b. Movement of a SRM is not considered to be a CORE ALTERATION.
d. The LPRM instrument handling tool is not a reactivity control component.

Source: ***Direct from Bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
95	14003	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0070504	CNS Tech. Spec. 3.3, Instrumentation

Related Objectives	
INT00705040010300	Given a set of plant conditions that constitutes non-compliance with a Section 3.3 LCO, determine the ACTIONS that are required.

Related References	
3.3.1.1	Reactor protection system (RPS) instrumentation
3.3.1.1-1	Table - Functions 1a and 1b

Related Skills (K/A)	
295020.AA2.03	Ability to determine and/or interpret the following as they apply to INADVERTENT CONTAINMENT ISOLATION: (CFR: 41.10 / 43.5 / 45.13) Reactor power. (3.7/3.7)

QUESTION: 14003

The plant has been operating at rated power for 2 months following a refueling outage. An LPRM calibration has just been completed. I&C asked for the TIP detector to be run through channel 10 again. As the TIP being withdrawn (currently between the bottom of the RPV and the bottom of the core), the shear valve is inadvertently fired and the TIP cable breaks. All attempts to subsequently remove the broken part of the TIP cable have failed. The TIP ball valve has been sucessfully closed.

What is the effect on continued power operation and why?

Power operation may continue . . .

a. for the next 115 days (when including any allowed extensions) due to the need to declare APRMs inoperable.

b. until the next refueling outage. 3D MONICORE can compensate for LPRM detector aging without running TIP traces.

c. for only the next hour as all ARPMs must be immediately declared inoperable due to the inability to perform a required surveillance.

d. until core exposure has increased by 1250 MWD/T (when including any allowed extensions) due to the need to declare APRMs inoperable.

ANSWER: 14003

d. until core exposure has increased by 1250 MWD/T (when including any allowed extensions) due to the need to declare APRMs inoperable.

SR 3.3.1.1.8 will not be met at 1250 MWD/T, requiring all APRMs to be declared inoperable.

Answer source: SR 3.3.1.1.8, SR 3.0.2

Distractors:

a. Based on the other 92 day surveillance plus 25% extension (unaffected).
b. Shutdown is required at 1250 MWD/T. 3D MONICORE cannot substitue for LPRM detector surveillances.
c. The surveillance is "met" even if it cannot be "performed" for the next 1250 MWD/T.

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

<u>Provide to Candidate:</u> T.S. 3.0 section and bases, T.S. LCO 3.3.1.1 and bases.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
96	14026	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0070504	CNS Tech. Spec. 3.3, Instrumentation

Related Objectives	
INT00705040010300	Given a set of plant conditions that constitutes non-compliance with a Section 3.3 LCO, determine the ACTIONS that are required.
INT00705040010100	Given a set of plant conditions, recognize non-compliance with a Section 3.3 Requirement.

Related References	
3.3.3.1	Post accident monitoring (PAM) instrumentation

Related Skills (K/A)	
216000.K3.25	Knowledge of the effect that a loss or malfunction of the NUCLEAR BOILER Instrumentation will have on following: (CFR: 41.7 / 45.4) Vessel pressure monitoring (3.9/4.1)

QUESTION: 14026

The plant is operating at rated power with the following conditions:

- NBI-PR-85A (Wide Range Reactor Pressure) (Post Accident Monitor) becomes inoperable at 1100 on 3/28
- NBI-LI-85A (Wide Range RPV water level) becomes inoperable at 0900 on 3/29
- NBI-PR-85B (Wide Range Reactor Pressure) (Post Accident Monitor) becomes inoperable at 1500 on 4/1

IF conditions do not change, what is the LATEST time that the plant is ***allowed*** to enter **MODE 3** by Technical Specifications?

a. 1500 on 4/8

b. 0300 on 4/9

c. 1500 on 5/1

d. 0300 on 5/2

ANSWER: 14026

b. 0300 on 4/9

Enter 3.3.3.1.A and 3.3.1.C at 1500 on 4/1. Enter 3.3.3.1.D at 1500 on 4/8. Enter 3.3.3.1.E at 1500 on 4/8. Be in MODE 3 by 0300 on 4/9.

Answer source: LCO 3.3.3.1

Distractors:

a. Not adding the 12 hours.
c. Not adding 12 hours and not entering condition C and misreading A.
d. not entering condition C and misreading A.

Source: ***New***

Provide to Candidate: T.S. 3.0 section and bases, T.S. LCO 3.3.3.1 and bases.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
97	19123	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0080617	OPS FLOWCHART 5A - SECONDARY CONTAINMENT AND RADIOACTIVITY RELEASE CONTROL

Related Objectives	
INT00806170010100	List the entry conditions to Flowchart 5A (including the radioactivity release path) and briefly explain each.

Related References	
PSTG	Plant Specific Technical Guideline

Related Skills (K/A)	
295035.EK1.01	Knowledge of the operational implications of the following concepts as they apply to SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE: (CFR: 41.8 to 41.10) Secondary containment integrity. (3.9/4.2*)

QUESTION: 19123

Why is entry into EOPs required if the Reactor Building dP cannot be maintained negative?

This reactor building (RB) dP is an indication that . . .

a. an uncontrolled, unmonitored release of radioactivity to the environment could exist.

b. the continued operability of equipment needed to carry out EOP actions may be compromised.

c. radioactivity is being released to the environment when the ventilation system should have automatically isolated.

d. an indication that water from a primary system (or from a primary to secondary system leak) may be discharging into the secondary containment.

ANSWER: 19123

a. an uncontrolled, unmonitored release of radioactivity to the environment could exist.

Answer source: INT008-06-17, p. 6, section C.1

Distractors:

b. This is the basis for the high temperature entry.
c. This is the basis for the high Rx bldg exhaust radiation level.
d. This is the basis for the entry on radiation above Max Normal Operating Level.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
98	16466	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0320104	CNS Administrative Procedures General Operating Procedures (Startup and Shutdown) Procedures (Formal Classroom/Pre-OJT Training)

Related Objectives	
INT032010400A0100	Discuss Precautions and Limitations outlined in General Operating Procedure 2.1.1, Startup Procedure.

Related References	
2.1.1	Procedure 2.1.1, Startup Procedure

Related Skills (K/A)	
2.2.1	Ability to perform pre-startup procedures for the facility / including operating those controls associated with plant equipment that could affect reactivity. (CFR: 45.1) (3.7/3.6)

QUESTION: 16466

During a reactor startup and heatup, reactor period was infinity after withdrawing a control rod. The following conditions are present with **<u>NO</u>** control rod movement for the last two (2) minutes:

- ➢ The reactor is on range 5 of the IRMs (rising)
- ➢ Reactor period is +120 seconds (shortening)
- ➢ Reactor coolant temperature is 180°F (rising)

Which one of the following represents the MINIMUM actions *<u>required</u>*?
(NOTE: The choices are arranged in MIMIMUM to MAXIMUM order.)

a. If reactor period shortens to 50 seconds, insert the last withdrawn control rod until period is longer than 50 seconds.

b. Range IRMs as necessary to keep them on scale until the Point of Adding Heat (POAH) is reached.

c. Insert control rods in reverse order to make the reactor subcritical.

d. Bypass the RWM and insert emergency power reduction rods (10.13 Att. 7) to position 00.

ANSWER: 16466

c. Insert control rods in reverse order to make the reactor subcritical.

Step 2.20 of procedure 2.1.1 states that conservative action ***<u>is required</u>*** whenever an unexpected situation arises with respect to reactivity, criticality, power level, or any other anomalous behavior of reactor core. This conservative action should include rod insertion to reduce power or a reactor scram without hesitation whenever such unanticipated or anomalous behavior is encountered. In this case indicated power is below the POAH yet temperature is rising and even with this negative reactivity feedback power is rising and period is getting shorter. All of which are significant indications of a significant anomoly.

<u>Answer source:</u> 2.1.1 p. 4, step 2.20

Distractors:

a. While the administrative limit for period is 50 seconds, the reactor is currently exhibiting anomolous behavior.
b. This action is not conservative, this action would allow the anomolous reactor behaviour to continue.
d. At this point in the startup operation would be below the 80% rod line and the emergency power reduction control rods are not available.

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

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
99	13406	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
INT0320124 CNS Abnormal Procedure (RO) Reactor Recirculation

Related Objectives	
INT032012400F0F00	Given plant condition(s), determine from memory the appropriate Abnormal/Emergency Procedure(s) to be utilized to mitigate the event(s).
INT032012400G0G00	Given plant condition(s), determine from memory all immediate operator actions required to mitigate the event(s).
INT032012400H0H00	Given plant condition(s), determine from memory if a manual reactor scram or an emergency shutdown from power is required due to the event(s).

Related References	
2.4RR	REACTOR RECIRCULATION ABNORMALS

Related Skills (K/A)	
202001.A2.04	Ability to (a) predict the impacts of the following on the RECIRCULATION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of...: (CFR: 41.5 / 45.6) Multiple recirculation pump trip (3.7/3.8)

QUESTION: 13406

Given the following conditions:

- ➢ Reactor power is 36%
- ➢ Recirculation pump B is idle
- ➢ Preparations to start "B" recirculation pump are in progress
- ➢ Bus 1C is supplied via the 1CS breaker
- ➢ The Start-up Transformer loses power

What action is required?

a. Manually scram the reactor.

b. Enter procedure 2.1.4, Normal Shutdown.

c. Enter procedure 2.1.4.1, Rapid Shutdown.

d. Close "B" Recirculation pump discharge valve, RR-MOV-MO53B.

ANSWER: 13406

a. Manually scram the reactor

Answer source: 2.4RR, p. 1, step 3.1

Distractors:

b. A scram is required.
c. A scram is required.
d. The "A" Recirc discharge valve must be shut, but not the "B".

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
100	12222	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
INT0320115 CHEMISTRY PROCEDURES

Related Objectives
INT0320115E0E0200 Discuss the compliance and use requirements associated with RWP's.

Related References	
9.ALARA.4	Procedure 9.ALARA.4, Radiation Work Permits

Related Skills (K/A)	
2.3.2	Knowledge of facility ALARA program. (CFR: 41.12 / 43.4. 45.9 / 45.10) (2.5/2.9)

QUESTION: 12222

A worker was replacing pump motor coupling in the Radwaste Basement. The work was being performed under an SWP due to radiation levels of 120 mrem in the area of the pump. Shortly after commencing work on the pump coupling in the Radwaste Basement, the worker was immediately needed to assess an emergent problem with a reactor feed pump oil pump.

What are the MINIMUM actions needed in order to allow the worker to proceed to the reactor feed pump?
(NOTE: The choices are arranged in MIMIMUM to MAXIMUM order.)

a. Log off the SWP.

b. Log onto the appropriate RCAWP.

c. Log off the SWP and log onto the appropriate RCAWP.

d. Log onto the appropriate RCAWP and log off of the SWP.

ANSWER: 12222

b. Log onto the appropriate RCAWP.

Logging onto the correct RCAWP automatically logs the worker off of the SWP.

Answer source: 9.ALARA.4, p. 9, step 7.7.2.2

Distractors:

a. The individual would need to log onto the appropriate RCAWP.
c. Because it is not necessary to perform the step of logging off the SWP.
d. As it is not necessary to log off the SWP.

Source: *Direct from Bank*

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

Applicant Information

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

Instructions

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

Applicant Certification

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

____

Applicant's Signature

Results

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

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
1	1111	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0010102 AC Electrical Distribution

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

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

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

QUESTION: 1111

Given the following conditions:

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

Which breaker(s) will trip?

a. **ONLY** EG1

b. **ONLY** 1AF

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

d. **BOTH** EG1 and 1FA

ANSWER: 1111

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

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

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

Distractors:

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

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
2	14036	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0010102 AC Electrical Distribution

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

Related References	
2.2.63	Procedure 2.2.63, PMIS Uninterruptible Power Supply System

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

QUESTION: 14036

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

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

PMIS would be . . .

a. deenergized.

b. powered directly from MDP-1.

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

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

ANSWER: 14036

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

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

Answer source: 2.2.63, p. 10, step 1.2.1

Distractors:

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

Source: *New*

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
3	1099	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0010102 AC Electrical Distribution

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

Related References	
2.2.18	4160V Auxiliary Power Distribution System

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

QUESTION: 1099

Given the following conditions:

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

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

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

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

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

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

ANSWER: 1099

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

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

Answer source: 2.2.20, p. 29, step 2.7.1.5

Distractors:

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

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
4	14035	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0011302 OPS MAIN GENERATOR AND AUXILIARIES

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

Related References	
2.2.14	22 KV Electrical System

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

QUESTION: 14035

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

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

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

MVARs become . . .

a. less negative and field amperage increases.

b. less negative and field amperage decreases.

c. more negative and field amperage increases.

d. more negative and field amperage decreases.

ANSWER: 14035

d. more negative and field amperage decreases.

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

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

Distractors:

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

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
5	14670	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0011402 OPS MAIN TURBINE

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

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

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

QUESTION: 14670

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

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

a. Reheat stop valves.

b. Extraction steam dump valves.

c. Extraction Steam Non-Return valves.

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

ANSWER: 14670

b. Extraction steam dump valves.

Answer source: 2.2.29, p. 19, step 2.1

Distractors:

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

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
6	19124	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0022802 OPS STANDBY GAS TREATMENT COR0011802 OPS Radiation Monitoring

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

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

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

QUESTION: 19124

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

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

NO group isolations or automatic initiations occur.

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

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

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

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

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

ANSWER: 19124

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

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

Answer source: 4.7.5, pp. 5 & 6, steps 1.2.3, 1.2.4, 1.2.5, & 1.3.1.1

Distractors:

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

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
7	5084	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0012002 OPS Reactor Water Cleanup

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

Related References	
2.1.22	Procedure 2.1.22, Recovering From A Group Isolation

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

QUESTION: 5084

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

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

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

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

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

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

ANSWER: 5084

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

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

Answer source: 2.1.22, p. 10, step 6.4

Distractors:

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

Source: ***Modified from 5084***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
8	14043	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0012402 OPS TURBINE EQUIPMENT COOLING SYSTEM

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

Related References	
2.2.76	Procedure 2.2.76, Turbine Equipment Cooling Water System

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

QUESTION: 14043

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

The normal supply is from the . . .

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

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

c. Turbine Equipment Cooling (TEC) system and the backup supply is from the Reactor Equipment Cooling (REC) system.

d. Reactor Equipment Cooling (REC) system and the backup supply is from the Turbine Equipment Cooling (TEC) system.

ANSWER: 14043

a. Turbine Equipment Cooling (TEC) system and the backup supply is from the Service Water (SW) system.

TEC supplies cooling to the Control Room Air Conditioner and can be supplied from SW by manually positioning local valves.

Answer source: 2.2.76, p. 33, step 1.2.2.7.

Distractors:

b. REC is not capable of providing the normal supply the Control Room AC unit.
c. REC is not capable of supplying the backup cooling to the Contol Room AC unit.
d. REC is not capable of providing the normal supply the Control Room AC unit and TEC is the normal supply.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
9	14045	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0013001 HWC Gas Generation System

Related Objectives	
COR0013001001040D	Identify the reason/function of the following systems interface and general physical location of the interface with the HWC system: Offgas System

Related References	
2.2.98	Hydrogen/Oxygen Generation System

Related Skills (K/A)	
272000.K5.01	Knowledge of the operational implications of the following concepts as they apply to RADIATION MONITORING SYSTEM: (CFR: 41.7 / 45.4) Hydrogen injection operation's effect on process radiation indications: Plant-Specific (3.2/3.5)

QUESTION: 14045

The plant is operating at 100% power with the hydrogen injection in service when OWC INJECTION SYS SHUTDOWN, A-3/F-4 alarms. The Control Room operator places the OWC INJECTION SYS ENABLE SWITCH to SHUTDOWN and verifies the that the green (Shutdown) light is on.

How does this affect ERP radioactive release rate and Main Steam Line (MSL) radiation level?

ERP release rate . . .

a. increases and MSL radiation levels increase.

b. decreases and MSL radiation level decrease.

c. is unchanged and MSL radiation level decrease.

d. is unchanged and MSL radiation level is unchanged.

ANSWER: 14045

c. is unchanged and MSL radiation level decrease.

The indications, annunciator and operator action indicate a loss of hydrogen injection. The loss of the hydrogen injection results in a shift of the ratio of N-16 as ammonia or ammoniun to nitrate or nitrite anion forms. This results in less carryover of N-16 out the main steam lines and a reduction in MSL radiation levels. Since N-16 has a short half life this change in carryover does not effect the release rate out the ERP.

<u>Answer source:</u> COR012-03-01, p. 4

"What is going to happen if Cooper starts adding hydrogen to the reactor water? According to the previous paragraph, if the nitrogen reacts with hydrogen, ammonia if formed. With a lot more hydrogen in the Reactor to combine with, the nitrogen will combine with it. Consequently there will be a lot more nitrogen-16 going over to the turbine and dose rates will be much higher."

Distractors:

a. ERP release rate does not increase.
b. ERP release rate does not decrease.
d. MSL radiation levels decrease.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
10	2931	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0020302 CONTAINMENT

Related Objectives	
COR0020302001050F	Describe the interrelationship between the Primary Containment system and the following: Plant Air
COR0020302001120F	Describe the Containment design features and/or interlocks that provide for the following: Reactor building to Torus D/P

Related References	
3.6.1.7	Reactor building-to-suppression chamber vacuum breakers
COR0020302	Containment

Related Skills (K/A)	
295019.AK2.09	Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR and the following: (CFR: 41.7 / 45.8) Containment. (3.3/3.3)

QUESTION: 2931

Drywell sprays are in service to support EOP actions when a complete loss of instrument air (including nitrogen) occurs. Due to a logic failure, the drywell spray valves (RHR-MO-26A and RHR-MO-31A) have been opened manually using the local handwheels.

Is the torus protected from exceeding design negative pressure under these conditions and why/why not?

a. Yes, all reactor building-to-torus vacuum breakers are motor-operated.

b. Yes, the reactor building-to-torus vacuum breakers fail in such a manner as to prevent an excessive negative pressure in the torus.

c. No, the reactor building-to-torus vacuum breakers fail closed on a loss of air.

d. No, the reactor building-to-torus vacuum breakers are not designed to facilitate this amount of flow.

ANSWER: 2931

b. Yes, the reactor building-to-torus vacuum breakers fail in such a manner as to prevent an excessive negative pressure in the torus.

Answer source: COR002-03-02, p. 20

Distractors:

a. One of the vacuum breakers is pneumatically operated.
c. The MOV doesn't fail anywhere on loss of air. The AOV fails open.
d. The vacuum breakers are sized to facilitate this flow.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
11	10081	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0020302 CONTAINMENT

Related Objectives	
COR0020302001080D	State the electrical power supplies to the following: N2 solenoid valve
COR0020302001240B	Predict the consequences of a malfunction of the following on PCIS: DC electrical.
COR0020302001230C	Predict the consequences of a malfunction of the following on the Primary containment: Containment atmospheric control/nitrogen make-up.

Related References	
2.2.60	Procedure 2.2.60, Primary Containment Cooling And Nitrogen Inerting System
2.3_9-3-1	Panel 9-3 - Annunciator 9-3-1
2.2.59	Procedure 2.2.59, Plant Air System

Related Skills (K/A)	
2.1.23	Ability to perform specific system and integrated plant procedures during different modes of plant operation. (CFR: 45: 45.2 / 45.13) (3.9/4.0)

QUESTION: 10081

The plant is at 100% power with the following conditions:

- ➢ Annuciator 9-3-1/G-2, NITROGEN SOLENOID DE-ENERGIZED is in alarm
- ➢ 125 VDC Panel AA2 is de-energized

Which of the following would be the ***quickest*** action that will restore pressure to the drywell pneumatic header?
(NOTE: The choices are listed from QUICKEST to LONGEST order.)

a. Open the cross-connect valve (IA-SOV-SPV21) from instrument air to the drywell pneumatic header using a switch on panel 9-3.

b. Open the Reactor building drywell supply air valve (IA-V-571) above the Southeast Hydraulic Control Units.

c. Open RR-SPV-740 AND RR-SPV-741 SUPPLY SHUTOFF (IA-1672) near RWCU precoat pump.

d. Hook up the nitrogen bottles that are stored in a rack near the header.

ANSWER: 10081

a. Open the cross-connect valve (IA-SOV-SPV21) from instrument air to the drywell pneumatic header using a switch on panel 9-3.

Answer source: 2.3_9-3-1, p. 74,
2.2.59 p. 25, step 1.2.11

Distractors:

b. This would take a person some time to manually open the valve (faster than bottles).
c. These valves are already open and would not restore drywell pneumatics.
d. This would take at least one person, some tools and time.

2.1.23 Ability to perform specific system and integrated plant procedures during different modes of plant operation. (CFR: 45: 45.2 / 45.13) as it applies to: 223002 PCIS/Nuclear Steam Supply Shutoff

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
12	5155	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0020402 CONTROL ROD DRIVE HYDRAULICS

Related Objectives	
COR0020402001140A	State the electrical power supply to the following CRDH components: CRDH pumps motors.

Related References	
5.3EMPWR	EMERGENCY POWER
2.2.8	Procedure 2.2.8, Control Rod Drive System
2.2.8A	Procedure 2.2.8A, Control Rod Drive Hydraulic System Valve Checklist

Related Skills (K/A)	
201001.K6.05	Knowledge of the effect that a loss or malfunction of the following will have on the CONTROL ROD DRIVE HYDRAULIC System: (CFR: 41.7 / 45.7) A.C. power (3.3/3.3)

QUESTION: 5155

Given the following conditions:

- ➢ The plant is operating at 65% power
- ➢ The "B" Control Rod Drive (CRD) Pump is running
- ➢ 480 VAC Critical Switchgear Bus 1G trips

What is the status of "B" CRD pump and Drive Water DP?

a. The "B" Pump is running.
Drive Water DP is unaffected.

b. The "B" Pump is stopped.
Drive Water DP will rapidly lower to zero.

c. The "B" Pump is stopped.
Drive Water DP will decay away over the next several minutes.

d. The "B" Pump is running.
The Drive Header Pressure Control Valve has lost power.

ANSWER: 5155

b. The "B" Pump is stopped.
Drive Water DP will rapidly lower to zero.

480 VAC Bus 1G provides power to CRD Pump "B" so Pump "B" is stopped. With no pump flow, drive header pressure will rapidly lower to Reactor pressure due to flow to the cooling header and Ref Leg Fill. Further reduction will be more gradual due to some check valve leakage, but Drive Water DP will quickly lower, as Drive pressure, to zero.

Answer source: 2.2.8A, p. 10

Distractors:

a. The "B" pump is powered by 480 VAC Bus 1G and will trip.

b. With no pump flow, drive header pressure will rapidly lower to Reactor pressure due to flow to the cooling header and Ref Leg Fill. Further reduction will be more gradual due to some check valve leakage, but Drive Water DP will quickly lower, as Drive pressure, to zero.

d. The "B" pump is powered by 480 VAC Bus 1G and will trip.

Source: *Direct from bank*

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
13	14040	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0020602 CORE SPRAY

Related Objectives	
COR0020602001080H	Given a Core Spray component manipulation, predict and explain the changes in the following: System lineup
COR0020602001120A	Given plant conditions, determine if any of the following Core Spray Actions should occur: System initiation.
COR0020602001120D	Given plant conditions, determine if any of the following Core Spray Actions should occur: Valve reposition.
COR0020602001050E	Describe the Core Spray system design features and/or interlocks that provide for the following: Pump minimum flow

Related References	
2.2.9	Procedure 2.2.9, Core Spray System

Related Skills (K/A)	
209001.A3.03	Ability to monitor automatic operations of the LOW PRESSURE CORE SPRAY SYSTEM including: (CFR: 41.7 / 45.7) System pressure (3.5/3.5)

QUESTION: 14040

The plant was operating at power with the A CS subsystem in full flow test. Core Spray system flow is pump flow is 5000 gpm. An accident occurs that results in increasing drywell pressure and lowering reactor water level and lowering reactor pressure. The following plant conditions exist:

➢ Drywell pressure	11 psig (rising)
➢ Reactor water level	-21" (wide range, lowering)
➢ Reactor pressure	375 psig (lowering)

What is the pressure response of the A Core Spray system ***at this time***?

Core spray system pressure . . .

a. remains the same.

b. increases to pump shut-off head.

c. decreases to just above reactor pressure.

d. increases to just below pump shut-off head.

ANSWER: 14040

d. increases to just below pump shut-off head.

An initiation signal is present for the Core Spray System. The CS test valve would receive a close signal resulting in a significant reduction in flow and since reactor pressure remains above the shut-off head for the pumps flow would be reduced to the point that the minimum flow valve would open. Core Spray system pressure would then be just below pump shut-off head.

Answer source: 2.2.9 p. 20

Distractors:

a. Core Spray pressure would not remain the same because system flow rate would become significantly reduced when the initiation signal occurred and the test line isolated.
b. Core Spray system pressure would be below shut-off head because of the flow that exists through the minimum flow valve.
c. Core Spray pressure increases.

<u>Source:</u> ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
14	14050	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0020702 OPS DC ELECTRICAL DISTRIBUTION

Related Objectives	
COR0020702001090A	Describe the DC Electrical Distribution System design feature(s) and/or interlock(s) that provide for the following: Manual/automatic transfers of control
COR0020702001090B	Describe the DC Electrical Distribution System design feature(s) and/or interlock(s) that provide for the following: Breaker interlocks, permissives, bypasses and crossties

Related References	
2.2.25.2	125 VDC ELECTRICAL SYSTEM (DIV 2)

Related Skills (K/A)	
263000.K4.02	Knowledge of D.C. ELECTRICAL DISTRIBUTION design feature (s) and/or interlocks which provide for the following: (CFR: 41.7) Breaker interlocks, permissives, bypasses and cross ties: Plant-Specific (3.1/3.5)

QUESTION: 14050

The plant is operating at rated power when the normal power supply the 125VDC HPCI Starter Rack is lost.

What interlocks exist for transfer of the HPCI Starter Rack to its alternate supply?

Inadvertent transfer of the 125VDC HPCI Starter Rack from its normal to its alternate supply is prevented by transfer switch design . . .

a. **<u>only</u>**.

b. **AND** the alternate supply breaker is locked open **<u>only</u>**.

c. **AND** a mechanical interlock prevents closing both supply breakers simultaneously **<u>only</u>**.

d. **AND** the alternate supply breaker is locked open **AND** a mechanical interlock prevents closing both supply breakers simultaneously.

ANSWER: 14050

b. **AND** the alternate supply breaker is locked open **<u>only</u>**.

<u>Answer source:</u> 2.2.25.2 pp. 35 & 36, sections 38.3, 38.4, 38.5 & 38.6

Distractors:

a. The alternate supply is locked open.
c. The alternate supply is locked open and both switches are closed at the same time during a transfer.
d. Both switches are closed at the same time during a transfer.

<u>Source:</u> *New*

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
15	19090	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0020702 OPS DC ELECTRICAL DISTRIBUTION COR0021202 INTERMEDIATE RANGE MONITOR

Related Objectives	
COR0020702001080L	Given a specific DC Electrical Distribution system malfunction, determine the effect on any of the following: SRMs
COR0020702001080R	Given a specific DC Electrical Distribution system malfunction, determine the effect on any of the following: Radiation Monitoring systems
COR0021202001070B	Predict the consequences of a loss or malfunction of the following would have on the IRM system: 24/48 VDC
COR0020702001080J	Given a specific DC Electrical Distribution system malfunction, determine the effect on any of the following: Reactor Protection system

Related References	
2.2.22	Procedure 2.2.22, Vital Instrument Power System
2.2.26	Procedure 2.2.26, 24 VDC Electrical System
2.1.22	Procedure 2.1.22, Recovering From A Group Isolation

Related Skills (K/A)	
215003.A3.03	Ability to monitor automatic operations of the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM including: (CFR: 41.7 / 45.7) RPS status (3.7/3.6)

QUESTION: 19090

The station is in MODE 2 withdrawing control rods in an approach to criticality during a startup. The following equipment simultaneously trips:
(NOTE: Other equipment also trips but is not required to assess conditions.)

- IRM "A", "C", "E" and "G"
- SRM "A" and "C"
- Off-Gas Radiation monitor "A"
- Reactor Building Vent Radiation monitors "A" and "C"
- Control rods remain at their pre-transient position
- **NO** group isolations have occurred

What occurred and what actions (if any) are required?

a. A loss of RPSPP "A" has occurred. Manually initiate a Reactor scram and a Group 6 isolation.

b. A loss of RPSPP "A" has occurred. **NO** operator actions are required to compensate for logic actuation failure.

c. A loss of 24 VDC "A" has occurred. Manually initiate a Reactor scram and a Group 6 isolation.

d. A loss of 24 VDC "A" has occurred. **NO** operator actions are required to compensate for logic actuation failure.

ANSWER: 19090

d. A loss of 24 VDC "A" has occurred. **NO** operator actions are required to compensate for logic actuation failure.

The loss of 24 vdc will cause all these instruments to become inoperative. No scram or group isolation will occur due to this single power loss.

<u>Answer source:</u> 2.2.22, p. 9 (RPS loss distractors),
2.2.26, step 2.2.1

Distractors:

a. RPS power loss would not cause the loss of IRMs/SRMs. No ATWS or group isolation failure has occurred.
b. RPS power loss would not cause the loss of IRMs/SRMs.
c. No ATWS or group isolation failure has occurred.

<u>Source:</u> ***Direct From Bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
16	1507	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0020802 DIESEL GENERATORS COR0010102 AC Electrical Distribution

Related Objectives	
COR0020802001090E	Describe the Diesel Generator design feature(s) and/or interlock(s) that provide for the following: Load Shedding and Sequencing
COR0010102001130B	Predict the consequences of the following events on the AC Electrical Distribution System: Loss of coolant accident
COR0010102001130C	Predict the consequences of the following events on the AC Electrical Distribution System: Loss of off-site power

Related References	
3.3.5.1	Emergency Core Cooling System (ECCS) Instrumentation
COR0020802	Diesel Generators

Related Skills (K/A)	
264000.K5.06	Knowledge of the operational implications of the following concepts as they apply to EMERGENCY GENERATORS (DIESEL/JET): (CFR: 41.5 / 45.3) Load sequencing (3.4/3.5)

QUESTION: 1507

The plant was operating at rated power when a loss of all off-site power occurred coincident with a large recirculation suction line break.

What will be the sequential loading of emergency buses?
(T = DG Output Breaker Closure)

a. T+0 the Core Spray pump starst;
   T+5 seconds the first RHR Pump starts;
   T+10 seconds the second RHR pump and SGT start.

b. T+0 the first RHR pump starts;
   T+5 seconds the Core Spray pump and SGT start;
   T+10 seconds the second RHR pump starts.

c. T+0 the first RHR pump and SGT starts;
   T+5 seconds the Core Spray pump starts;
   T+10 seconds the second RHR pump starts.

d. T+0 the first RHR pump and SGT starts;
   T+5 seconds the second RHR pump starts;
   T+10 seconds the Core Spray pump starts.

ANSWER: 1507

d. T+0 the first RHR pump and SGT starts;
   T+5 seconds the second RHR pump starts;
   T+10 seconds the Core Spray pump starts.

Answer source: COR002-08-02, p. 65, Table 1

Distractors:

a. RHR pump starts first and second, CS starts last. SGT starts at T=0.
b. RHR pump starts first and second, CS starts last. SGT starts at T=0.
c. RHR pump starts first and second, CS starts last.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
17	14032	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0021102 OPS High Pressure Coolant Injection (HPCI)

Related Objectives	
COR0021102001080N	Describe the HPCI design features and/or interlocks that provide for following: Pump minimum flow
COR0021102001100H	Predict the consequences of the following on the HPCI system: Low ECST level
COR0021102001080M	Describe the HPCI design features and/or interlocks that provide for following: Protection against draining the CST to the torus

Related References	
2.2.33	Procedure 2.2.33, High Pressure Coolant Injection System

Related Skills (K/A)	
206000.A4.07	Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) Condensate storage tank level: BWR-2,3,4 (3.5/3.5)

QUESTION: 14032

The plant was operating at power with the HPCI system in full flow test per 6.HPCI.103 when annunciator 9-3-2/A-4, HPCI SUCTION TRANSFER alarms. ECST level is 22".

What is the status/alignment of HPCI several minutes later?
(Assume the operator takes no action.)

HPCI suction valves are aligned to the suppression pool, the Minimum Flow Valve (MO-25) is _________ and the Pump Test Return Line Isolation Valves (MO-21 & 24) are _________.

a. open closed

b. closed closed

c. open open

d. closed open

ANSWER: 14032

a. open closed

The low ECST level has initiated a swap of the of the HPCI suction valves. With the swap over to suction from the suppression pool the pump test return isolation valves automatically close. Now the HPCI system is without a discharge path the minimum flow valve opens due to low flow.

Answer source: 2.2.33 p. 15, steps 2.1.1.8 (MO-25), 2.1.1.9 (MO-21) & 2.1.1.10 (MO-24)

Distractors:

b. The minimum flow valve is open.
c. The Pump Test Return Line Isolation Valves, (MO-21 & 24) are closed.
d. The minimum flow valve is open and the Pump Test Return Line Isolation Valves, (MO-21 & 24) are closed.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
18	19051	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
COR0021102	OPS High Pressure Coolant Injection (HPCI)
SKL0124211	HIGH PRESSURE COOLANT INJECTION

Related Objectives	
COR0021102001120A	Given plant conditions, determine if the following HPCI actions should occur: System initiation
SKL012421100A030J	Given plant conditions, predict changes in the following HPCI system components/parameters: Turbine speed
SKL012421100B0600	Comply with all related HPCI system limits and precautions.
COR0021102001100V	Predict the consequences of the following on the HPCI system: High reactor water level
SKL012421100A0200	Explain the HPCI system limitations and precautions as stated in the SOP 2.2.33 and SOP 2.2.33.1.

Related References	
791E271	HPCI System Elementary Diagram
2.2.33	Procedure 2.2.33, High Pressure Coolant Injection System

Related Skills (K/A)	
2.1.32	Ability to explain and apply system limits and precautions. (CFR: 41.10 / 43.2 / 45.12) (3.4/3.8)

QUESTION: 19051

The plant was operating at power when a loss of off-site power occurred. The reactor scrammed and HPCI started on low reactor water level. Reactor water level quickly recovered and the HPCI turbine tripped on high RPV water level. The following plant conditions were present:

➢ Reactor water level	45" (NR) (lowering slowly)
➢ Reactor pressure	850 psig (rising slowly)
➢ Drywell pressure	2.2 psig (rising slowly)

What is/are the **MINIMUM** action(s) required to restart HPCI ***at this time***?
(NOTE: The choices are arranged in MIMIMUM to MAXIMUM order.)

a. Momentarily depress the Reactor Hi Water Level Signal Reset pushbutton **ONLY**.

b. Momentarily depress the Initiation Signal Reset pushbutton **AND** place the HPCI-MO-14 control switch to open.

c. Momentarily depress the Reactor Hi Water Level Signal Reset pushbutton **AND** place the HPCI-MO-14 control switch to open.

d. Momentarily depress the Reactor Hi Water Level Signal Reset pushbutton **AND** momentarily depress the Initiation Signal Reset pushbutton **AND** place the HPCI-MO-14 control switch to open.

ANSWER: 19051

a. Momentarily depress the Reactor Hi Water Level Signal Reset pushbutton **ONLY**.

Per 2.2.33.1:

CAUTION - If HPCI initiation signal cannot be reset, HPCI System will automatically start when RX HI WTR LEVEL SIGNAL RESET pushbutton is depressed and vessel level is ≤ +54".

During the transient drywell pressure has risen to greater than the initiation setpoint for HPCI. Since an automatic initiation signal is present, if the operator depresses the Reactor Hi Water Level Signal Reset pushbutton the system will reinitiate.

Answer source: 2.2.33.1, p. 8
2.2.33, p. 20, step 2.2.6

Distractors
b, c, d - only the high level trip reset need be depressed.

2.1.32 Ability to explain and apply system limits and precautions. (CFR: 41.10 / 43.2 / 45.12) as it applies to: 295008 High Reactor Water Level / 2

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
19	16513	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0020902 Digital Electro-Hydraulic Control

Related Objectives	
COR0020902001040B	Describe how the DEH control system operates to control the following: Reactor pressure
COR0020902001070B	Given a specific DEH Control system malfunction, determine the effect on any of the following: Reactor pressure

Related References	
2.2.77.1	Procedure 2.2.77.1, DEH Control System

Related Skills (K/A)	
241000.K4.01	Knowledge of REACTOR/TURBINE PRESSURE REGULATING SYSTEM design feature (s) and/or interlocks which provide for the following: (CFR: 41.7) Reactor pressure control (3.8/3.8)

QUESTION: 16513

The plant is operating at 100% power when the in-service DEH pressure controller fails such that controller output INCREASES slowly.

Assuming NO operator action is taken, what is the plant response?

a. The reactor scram due to high reactor pressure.

b. The MSIVs isolate due to low reactor pressure.

c. Turbine throttle pressure will be controlled about 4 psig LOWER than before the failure.

d. Turbine throttle pressure will be controlled about 4 psig HIGHER than before the failure.

ANSWER: 16513

b. The MSIVs isolate due to low reactor pressure.

As the controller output increases the turbine governor valves would open in response to the controller output. The opening of the valves would reduce reactor pressure and result in a MSIV closure due to low MSL pressure with the mode switch in RUN.

Answer source: COR002-09-02, p. 49, section "c."

Distractors:

a: Reactor pressure will lower as controller output signals the TCVs to OPEN.
c: The backup pressure regulator is set for a pressure 4 psi higher.
d: Reactor pressure will lower as controller output signals the TCVs to OPEN.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
20	1058	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0021502 NUCLEAR BOILER INSTRUMENTATION

Related Objectives	
COR0021502001040H	Briefly describe the following concepts as they apply to NBI: Recirculation flow effects on level indicators

Related References	
COR0021502	Nuclear Boiler Instrumentation

Related Skills (K/A)	
295001.AA1.07	Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: (CFR: 41.7 / 45.6) Nuclear boiler instrumentation system. (3.1/3.2)

QUESTION: 1058

The plant is operating at 100% power when both Reactor Recirculation pumps slowly run back to minimum speed. No operator action is taken.

What is the difference between ACTUAL and INDICATED Wide Range Reactor Water Level ***prior*** to the power reduction AND what is the expected change in that difference ***during*** the power reduction?

Prior to the power reduction, *actual* downcomer level is ___________ than *indicated* downcomer level AND the difference will get ___________ during the power reduction.

a. lower larger

b. higher larger

c. lower smaller

d. higher smaller

ANSWER: 1058

d. higher smaller

Due to the velocity affects of flow in the annulus, the variable leg will sense a lower pressure than is exerted by the height of water alone. This is seen as a lower indicated level. At higher recirc flows, higher velocities cause a greater difference between Wide Range indicated and actual levels. The difference between indicated and actual levels can range from 4-18"

Answer source: COR002-15-02, p. 42 & 43, section 2.c, d, f, g, & h.

Distractors:

a. Actual level is higher than indicated level. The difference will get smaller as power is reduced.
b. The difference will get smaller as power is reduced.
c. Actual level is higher than indicated level.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
21	5425	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0021602 OPS NUCLEAR PRESSURE RELIEF

Related Objectives	
COR0021602001050J	Describe the Nuclear Pressure Relief system design features and/or interlocks that provide for the following: Safety/Relief operating signals
COR0021602001030J	Describe the interrelationships between the Nuclear Pressure Relief system and the following: RPS (low-low set initiation)

Related References	
2.2.1	Automatic Depressurization System

Related Skills (K/A)	
239002.K6.03	Knowledge of the effect that a loss or malfunction of the following will have on the RELIEF/SAFETY VALVES: (CFR: 41.7 / 45.7) A.C. power: Plant-Specific (2.7*/2.9*)

QUESTION: 5425

Given the following:

- ➢ The plant is operating at 100% power at Beginning of Life (BOL)
- ➢ A loss of off-site power occurs
- ➢ All control rods fully inserted
- ➢ Pressure rises to 1090 psig, **THEN** lowers to 875 psig
- ➢ 20 minutes later, pressure is cycling between 1015 **AND** 875 psig

What is the status of Low Low Set (LLS) and why?
(NOTE: **NO** operator action is taken.)

a. LLS is controlling pressure. LLS logic has no AC powered inputs or components.

b. LLS is controlling pressure. With RPS power unavailable, the LLS logic can ***arm*** irrespective of reactor pressure.

c. LLS is **NOT** controlling pressure. With RPS power unavailable, the SRVs must operate on mechanical relief setpoint to control pressure.

d. LLS is **NOT** controlling pressure. A fault must exist in the LLS logic as all conditions are present for LLS to automatically control pressure.

ANSWER: 5425

b. LLS is controlling pressure. With RPS power unavailable, the LLS logic can ***arm*** irrespective of reactor pressure.

Answer source: COR002-16-02, p. 24 & p. 24, p. 43 section E,
COR002-16-02 Figure 6

Distractors:

a. LLS logic is armed by RPS high pressure signal.
c. LLS logic will arm on high pressure with no RPS power available.
d. LLS is controlling pressure.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
22	5608	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0021602 OPS NUCLEAR PRESSURE RELIEF

Related Objectives	
COR0021602001020A	State the electrical power supply to the following NPR components: ADS logic
COR0021602001080F	Predict the consequences a malfunction of the following would have on the NPR system: D.C. power

Related References	
791E253	Automatic Blowdown System
2.2.1	Automatic Depressurization System

Related Skills (K/A)	
218000.K2.01	Knowledge of electrical power supplies to the following: (CFR: 41.7) ADS logic (3.1*/3.3*)

QUESTION: 5608
An accident has occurred, resulting in the following conditions:

➢ Reactor pressure	720 psig (lowering)
➢ RPV water level	-120" (WR stable)
➢ Drywell pressure	6.2 psig (rising)
➢ 125 VDC panel AA2	De-energized

If present conditions continue, how will ADS respond?

ADS valves will . . .

a. ***fail to open*** due to loss of logic power.

b. ***fail to open*** due to RPV water level conditions not met.

c. be opened by the B logic circuit powered from its ***normal*** power source.

d. be opened by both logic circuits powered from their ***alternate*** power sources.

ANSWER: 5608

c. be opened by the B logic circuit powered from its ***normal*** power source.

Answer source: COR002-16-02, p. 21, & p. 22, section 3, p. 41
COR002-16-02 Figures 4 & 5

Distractors:

a. ADS will initiate powered from BB2.
b. ADS will initiate.
d. ADS "A" has no alternate source and is de-energized.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
23	14679	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0021702 PLANT MANAGEMENT INFORMATION SYSTEM

Related Objectives	
COR0021702001070A	Given a specific PMIS malfunction, determine the effect on any of the following: On Demand print out.

Related References	
2.6.3	Procedure 2.6.3, COMPUTER SYSTEMS OPERATION AND OUTAGE RECOVERY
2.4COMP	Computer Malfunction

Related Skills (K/A)	
2.1.19	Ability to use plant computer to obtain and evaluate parametric information on system or component status. (CFR: 45.12) (3.0/3.0)

QUESTION: 14679

A plant shutdown is in progress with reactor power at 30% of rated and both PMIS computers in service when a loss of the primary PMIS computer occurs.

What is the impact of these conditions on plant operation?

Official Cases are . . .

a. available and, if the shutdown were to continue, RWM would be available.

b. available and, if the shutdown were to continue, RWM would be **UN**available.

c. **UN**available and, if the shutdown were to continue, RWM would be available.

d. **UN**available and, if the shutdown were to continue, RWM would be **UN**available.

ANSWER: 14679

a. available and, if the shutdown were to continue, RWM would be available.

When both PMIS computers are unavailable, monitoring functions (Computer Edits) 3D Monicore Official Cases, process parameter alarm monitoring and many other important functions are lost; however, in this instance, with the backup computer available, a Loss of primary computer will result in automatic fail-over to the backup computer, so there is no immediate impact on plant operation, all computer functions are available.

Answer source: COR002-17-02, p. 9, section B.1

Distractors:

b. The backup computer is available so the RWM remains available on the subsequent shutdown.
c. The backup computer is available so computer on-demand printouts remain available.
d. The backup computer is available so computer on-demand printouts remain available as does the RWM on the susequent shutdown.

Source: ***Direct from Bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
24	14451	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0021802 OPS Reactor Core Isolation Cooling (RCIC)

Related Objectives	
COR0021802001120D	Given plant conditions, determine if the following RCIC actions should occur: Minimum flow valve position change
COR0021802001120A	Given plant conditions, determine if the following RCIC actions should occur: RCIC system initiation
COR0021802001100O	Predict the consequences of the following on the RCIC system: RCIC Turbine control system failure
COR0021802001120E	Given plant conditions, determine if the following RCIC actions should occur: RCIC turbine trip

Related References	
2.2.67	Procedure 2.2.67, Reactor Core Isolation Cooling System

Related Skills (K/A)	
217000.K5.02	Knowledge of the operational implications of the following concepts as they apply to REACTOR CORE ISOLATION COOLING SYSTEM (RCIC): (CFR: 41.5 / 45.3) Flow indication (3.1/3.1)

QUESTION: 14451

The Reactor Core Isolation Cooling (RCIC) flow transmitter has failed low such that it senses 0 gpm irrespective of actual RCIC flow.

What is the expected RCIC system response upon receipt of a valid initiation signal?

The RCIC turbine will start and . . .

a. run normally.

b. trip on overspeed.

c. run continuously at minimum speed.

d. run continuously at approximately 4500 rpm.

ANSWER: 14451

d. run continuously at approximately 4500 rpm.

Loss of flow signal input to the flow controller results in a maximum speed demand signal. Since the output of the control box is limited to 50 milliamps, the turbine speed will top out at approximately 4500 rpm.

<u>Answer source:</u> COR002-18-02, p. 56, section 2

Distractors:

a. RCIC will not run normally.
b. The ramp generator is still functional on startup. RCIC RPM will not exceed 4500 when controller output is at 100%. Overspeed occurs at 5625 RPM.
c. RCIC will not run at mimimum speed.

<u>Source:</u> *Direct from bank*

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
25	14027	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
ACD0070306 Heat Transfer & Heat Exchangers (GP) ACD0070307 Thermal Hydraulics (GP)

Related Objectives	
ACD00703020010800	Apply saturated and superheated steam tables in solving liquid-vapor problems.
ACD00703060010500	Solve heat flux and heat transfer rate problems.
ACD0060507001310C	Explain the relationship between decay heat generation and: time since reactor shutdown

Related References	
ACD0060507	Reactor Operational Physics

Related Skills (K/A)	
295007.AK1.02	Knowledge of the operational implications of the following concepts as they apply to HIGH REACTOR PRESSURE: (CFR: 41.8 to 41.10) Decay heat generation (3.1/3.4)

QUESTION: 14027

The plant has been shutdown for several days following a long operating cycle. Shutdown Cooling has been placed in service and a cooldown established. The following conditions are present:

- Reactor pressure is 15 psig
- Reactor vessel inventory is 350,000 lbm
- Decay heat rate is 0.6% of rated thermal power
- Ambient heat loss is 7.5 MWt
- Reactor coolant specific heat capacity is 1.08 BTU/lbm°F

A station blackout then occurs.

What reactor pressure will exist **two** (2) hours following this loss of all AC power?
(Assume that reactor inventory and ambient losses remain constant for the entire two hours.)

a. 83 psig

b. 90 psig

c. 165 psig

d. 180 psig

ANSWER: 14027

c. 165 psig

The thermal power of the reactor is 14.3 Mwt (.006*2381=14.3Mwt). The thermal power that is absorbed in the coolant is 14.3Mwt - 7.5 Mwt (ambient loss) =6.8Mwt. 6.8 Mwt is converted to BTU/hr by multiplying by 3.41E6. This yields 2.31E7 BTU/hr. Since the question asks for the conditions 2 hours after the loss of AC power this heat rate continues for 2 hours so 2.31E7BTU/hr is multiplied by 2 hrs to yield the total BTUs absorbed by the coolant or 4.62E7 BTUs. Now the known values can be substituted into the following equation to solve for the final temperature. $Q=Mc_p\Delta T$ The final temperature is calculated to be 373°F. Now steam tables are used to find the saturation pressure for that temperature (180 psia). The value is then converted to psig and the final answer of 162 psig. As the RPV is an enclosed volume under these conditions, addition of energy to the mass of water will result in a temperature change and very little phase change. The amount of energy lost to the latent heat of vaporization is insignificant when compared to the total enthalpy of the water in the RPV.

Answer source: Steam Tables, Generic Fundamentals

Distractors:

a. Would only be obtained if the candidate failed to account for two hours after the loss of AC power.
b. Would be obtained if the candidate failed to account two hours since the loss of AC power and failed to convert that answer to psig.
d. Would be only be obtained if the candidate failed to convert the final answer to psig.

Source: ***New***

Provide to Candidate: Calculator, Steam Tables, GFES Formula sheet.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
26	2521	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0021902 REACTOR EQUIPMENT COOLING

Related Objectives	
COR0021902001060K	Given a specific REC malfunction, determine the efect on any of the following: Fuel Pool Cooling system

Related References	
2.2.65	Procedure 2.2.65, Reactor Equipment Cooling Water System
5.2REC	LOSS OF REC

Related Skills (K/A)	
295018.AK2.01	Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER and the following: (CFR: 41.7 / 45.8) System loads (3.3/3.4)

QUESTION: 2521

Closure of which of the following REC valves could lead to an increase in the fuel pool water temperature and in increase in airborne contamination?

a. Drywell Supply Isolation (REC-MO-702).

b. Non-Critical Header Supply (REC-MO-700).

c. Augmented Radwaste Supply (REC-MO-1329).

d. Critical Loop Return Crossover Valve (REC-MO-694).

ANSWER: 2521

b. Non-Critical Header Supply (REC-MO-700).

Answer source: 2.2.65, pp. 12 & 13, section 1.2

Distractors:

a. FPC is supplied by the non-critical header.
c. FPC is supplied by the non-critical header.
d. FPC is supplied by the non-critical header.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
27	14039	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0021902 REACTOR EQUIPMENT COOLING

Related Objectives	
COR0021902001110A	Given plant conditions, determine if any of the following should occur: Non-Critical loop isolation
COR0021902001110C	Given plant conditions, determine if any of the following should occur: Any REC valve automatic reposition

Related References	
2.2.65	Procedure 2.2.65, Reactor Equipment Cooling Water System

Related Skills (K/A)	
400000.K6.05	Knowledge of the effect that a loss or malfunction of the following will have on the CCWS: (CFR: 41.7 / 45.7) Motors (2.8/2.9)

QUESTION: 14039

The plant was at power with Reactor Equipment Cooling (REC) pumps A, B, and D running and REC pump C tagged out for maintenance. The B REC pump then tripped due to an electrical fault in the motor. REC pressure lowered to 40 psig before increasing and stabilizing two (2) minutes later at 53 psig.

Which of the following loads **CAN** be supplied with REC?

a. "A" Drywell Fan Coil Unit

b. "A" Station Air Compressor

c. "A" Control Rod Drive pump

d. Northwest Quad Fan Coil Unit.

ANSWER: 14039

d. Northwest Quad Fan Coil Unit.

With an isolation signal present REC-MO-702MV can be reopened, however, the REC-MO-712 and 713 will auto close on the low pressure and cannot be overriden. This will isolate REC to the non-critical loops/components. The fan coil unit is the only load listed supplied from the critical loop.

<u>Answer source:</u> 2.2.65, p. 14, section 2.5
2.2.65, p. 15, sections 2.9 & 2.10

Distractors:

a. The drywell fancoil will remain isolated because REC pressure remains below the isolation setpoint.
b. REC flow to the air compressor will remain isolated because REC pressure remains below the isolation setpoint.
c. REC flow to the CRD pump will remain isolated because REC pressure remains below the isolation setpoint.

<u>Source:</u> Modified *Original Question 5279*

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
28	14024	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0022002 OPS REACTOR MANUAL CONTROL SYSTEM

Related Objectives	
COR0022002001150E	Given plant conditions related to RMCS and/or RPIS, determine if any of the following should occur: Control rod drift alarm
COR0022002001010I	State the purpose of the following items related to the Reactor Manual Control System and/or the Rod Position Information System: Rod Drift Alarm Test Switch

Related References	
6.CRD.303	CONTROL ROD WITHDRAWAL/OPERABILITY TEST MODE 3, 4, AND 5

Related Skills (K/A)	
201002.A4.03	Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) Rod drift test switch (2.8/2.8)

QUESTION: 14024

What represents the **MINIMUM** action(s) required to generate a rod drift alarm?
(NOTE: The choices are arranged in MIMIMUM to MAXIMUM order.)

a. Rod drift test switch momentarily held to test position.

b. Rod drift test switch held in the test position while control rod is inserted one notch.

c. Rod drift test switch held in the test position while a control rod is inserted and then withdrawn one notch.

d. Control rod inserted one notch and the rod drift test switch is momentarily taken to test while the amber rod settle light is energized.

ANSWER: 14024

b. Rod drift test switch held in the test position while control rod is inserted one notch.

Any Rod movement that leaves an even reed switch or picks up an odd reed switch with the Rod Drift Alarm Test switch in test generates a Rod Drift Alarm.

Answer source: COR002-20-02, p. 18, section 4

Distractors:

a. Just Placing the Rod Drift Alarm Test switch to TEST does not generate a rod drift alarm.

c. While this would generate a rod drift alarm, the rod need only be either inserted OR withdrawn, so these actions do not represent the minimum required by the question.

d. This action may not generate a rod drift alarm, if the even reed switch for the control rod is made up before the Rod Drift Alarm Test switch is taken to TEST, no alarm would be generated.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
29	1208	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0022102 REACTOR PROTECTION SYSTEM

Related Objectives	
COR0022102001100K	Describe the interrelationship between the RPS and the following: Primary Containment
COR0022102001050A	Briefly describe the following concepts as they apply to RPS: Logic arrangements

Related References	
2.1.5	Procedure 2.1.5, Reactor Scram
2.2.22	Procedure 2.2.22, Vital Instrument Power System
2.1.22	Procedure 2.1.22, Recovering From A Group Isolation

Related Skills (K/A)	
212000.A2.09	Ability to (a) predict the impacts of the following on the REACTOR PROTECTION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the conseq...: (CFR: 41.5 / 45.6) High containment/drywell pressure (4.1*/4.3*)

QUESTION: 1208

The plant is operating at 10% power and rated pressure during a plant startup with the following conditions:

- The "A" Reactor Protection System (RPS) MG has tripped due to a motor fault
- A loss of Drywell (DW) cooling has caused a slow but continuous rise in DW temperature and pressure
- Primary Containment parameters do not improve

What drywell pressure switch configuration will result in a full reactor scram and how is overfill of the RPV prevented?

A reactor scram will occur . . .

a. if <u>any single</u> RPS system "B" DW pressure switch opens. CRD-MO-20, DRIVE PRESSURE CONT VALVE must be closed.

b. only when <u>both</u> RPS system "B" DW pressure switches open. CRD-MO-20, DRIVE PRESSURE CONT VALVE must be closed.

c. if <u>any single</u> RPS system "B" DW pressure switch opens. CRD-V-29, CHARGING WATER HEADER ROOT VALVE must be closed.

d. only when <u>both</u> RPS system "B" DW pressure switches open. CRD-V-29, CHARGING WATER HEADER ROOT VALVE must be closed.

ANSWER: 1208

c. if <u>any single</u> RPS system "B" DW pressure switch opens. CRD-V-29, CHARGING WATER HEADER ROOT VALVE must be closed.

<u>Answer source:</u> COR002-21-02 p.12, section 3
2.1.5 p. 8, section 1.3

Distractors:

a. closing the drive pressure control valve will not prevent overfill.
b. Does not need to be both switches, closing the drive pressure control valve will not prevent overfill.
d. Does not need to be both switches.

Source: ***Modified***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
30	19096	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
SKL0124222 REACTOR RECIRCULATION SYSTEM COR0022202 REACTOR RECIRCULATION

Related Objectives	
COR0022202001130F	Given plant conditions, determine if any of the following should occur: Recirculation MG set scoop tube lock.
COR0022201001060A	Given plant and/or reactor recirculation system conditions, apply the design features and/or interlocks that provide for the following: MG Set Scoop Tube Lockout
SKL012422200A030I	Given plant conditions, predict changes in the following Reactor Recirculation System components/parameters: RR pump speed

Related References	
2.2.68	Procedure 2.2.68, Reactor Recirculation System Operations

Related Skills (K/A)	
202002.A4.01	Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) MG sets (3.3/3.1)

QUESTION: 19096

The plant is operating at 30% when the reactor operator is directed to raise power using Recirculation flow. As the controller output is raised, a momentary (1.5 seconds) loss of signal occurs from the "A" Recirculation Flow Controller. The operator continues to raise the controller output for several more seconds.

How will the "A" Recirculation MG Set be affected by this momentary loss and operator action?

a. The pump will automatically run back to ~ 22% speed.

b. A scoop tube lockup will prevent any further speed change.

c. After a 1.5 second pause, recirculation pump speed will rise for several seconds.

d. Speed will initially rise, then lower rapidly for 1.5 seconds, then rise again for several seconds.

ANSWER: 19096

b. A scoop tube lockup will prevent any further speed change.

Answer source: COR002-22-02, p. 35, section "d"

Distractors:

a. There is no runback.
b. Scoop tube lockup prevents any speed changes.
d. Scoop tube lockup prevents any speed changes.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
31	1744	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0022302 RESIDUAL HEAT REMOVAL

Related Objectives	
COR0022302001030P	Describe RHR System design feature(s) and/or interlocks which provide for the following: Spray flow cooling
COR0022302001170C	Given plant conditions, determine actions necessary to place RHR in the following flowpaths: Drywell Spray

Related References	
2.2.69.3	Procedure 2.2.69.3, RHR Suppression Pool Cooling And Containment Spray
2.2.69	Procedure 2.2.69, Residual Heat Removal System

Related Skills (K/A)	
2.4.48	Ability to interpret control room indications to verify the status and operation of system / and understand how operator actions and directives affect plant and system conditions. (CFR: 43.5 / 45.12) (3.5/3.8)

QUESTION: 1744

Following a LOCA, the following conditions are present :

- Reactor pressure 700 psig (lowering slowly)
- RPV water level - 100 in (**wide range**, stable)
- Drywell press 11.0 psig (rising slowly)

What are the **MINIMUM** actions that are required in order to initiate Drywell Sprays?
(NOTE: The choices are arranged in MIMIMUM to MAXIMUM order.)

a. Place Drywell Inbd (MO-31A/B) and Outbd (MO-26A/B) Spray Valve control switches in OPEN.

b. Place Containment Cooling Valve Control Permissive switches in MANUAL, then place Drywell Inbd (MO-31A/B) and Outbd (MO-26A/B) Spray Valve control switches in OPEN.

c. Place Containment Cooling 2/3 Core Valve Control Permissive switches in OVERRIDE, place the Containment Cooling Valve Control Permissive switches in MANUAL, then place Drywell Inbd (MO-31A/B) and Outbd (MO-26A/B) Spray Valve control switches in OPEN.

d. Depress Containment Spray Initiation Signal Reset pushbuttons, place Containment Cooling 2/3 Core Valve Control Permissive switches in OVERRIDE, place the Containment Cooling Valve Control Permissive switches in MANUAL, then place Drywell Inbd (MO-31A/B) and Outbd (MO-26A/B) Spray Valve control switches in OPEN.

ANSWER: 1744

b. Place Containment Cooling Valve Control Permissive switches in MANUAL, then place Drywell Inbd (MO-31A/B) and Outbd (MO-26A/B) Spray Valve control switches in OPEN.

Answer source: 2.2.69, pp. 35 - 36, section 2.2.10

Distractors:

a. The permissive switch must be placed in MANUAL.
c. No need to place 2/3 core height in override.
d. No need to place 2/3 core height in override or reset logic.

2.4.48 Ability to interpret control room indications to verify the status and operation of system / and understand how operator actions and directives affect plant and system conditions. (CFR: 43.5 / 45.12) as it applies to: 295024 High Drywell Pressure

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
32	4029	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0022302 RESIDUAL HEAT REMOVAL

Related Objectives	
COR0022302001020A	State the electrical power supplies to the following: RHR pump motors
COR0022302001080A	Predict the consequences a malfunction of the following will have on the RHR system: A.C. electrical power (including RPS)

Related References	
2.2.69	Procedure 2.2.69, Residual Heat Removal System

Related Skills (K/A)	
230000.K2.02	Knowledge of electrical power supplies to the following: (CFR: 41.7) Pumps (2.8*/2.9*)

QUESTION: 4029

The plant was at power with the breaker for RHR Pump 1C tagged out for maintenance when the following occur:

- ➢ A reactor coolant leak in the drywell results in a drywell pressure of 8 psig (slowly rising)
- ➢ A loss of 4160 VAC Switchgear Critical Bus 1F occurs

What RHR pumps/loops remain available and what operations be accomplished <u>from the Control Room</u>?

RHR . . .

a. Loop A with **one** pump is available for LPCI injection. Torus sprays **CANNOT** be established in either loop.

b. Loop B with **one** pump is available for LPCI injection. Torus sprays are available from RHR loop B **ONLY**.

c. Loop B with **both** pumps is available for LPCI injection. Torus sprays **CANNOT** be established in either loop.

d. Loop A with **one** pump **AND** RHR Loop B with **one** pump are available for LPCI injection. Torus sprays are available from RHR loop A **ONLY**.

ANSWER: 4029

b. Loop B with **one** pump is available for LPCI injection. Torus sprays are available from RHR loop B **ONLY**.

Torus sprays are available from RHR loop B **ONLY**. The loss of power to 1F results in the loss of power to RHR pumps A and B. With C pump already out of service, the only remaining pump is RHR pump D, a B loop pump. Since the B loop injection valve is DC powered LPCI remains available from B loop. Power remains available to the containment cooling valves for the B loop so torus spray is available.

<u>**Answer source:**</u> COR002-23-02, pp. 18 & 19, Section 4
COR002-23-02, p. 67, Table 1

Distractors:

a. Neither RHR loop A pumps are available. The C pump is OOS for maintenance and with 1F bus deenergized no power is available to the A pump.
c. Only RHR pump B still has power availble so only one pump is available. Torus sprays are available from the B loop.
d. No Loop A pumps remain available and torus sprays are not available from RHR loop A.

<u>Source</u>: ***Modified original question 4029.***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
33	14028	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0022302 RESIDUAL HEAT REMOVAL

Related Objectives	
COR0022302001080N	Predict the consequences a malfunction of the following will have on the RHR system: Suction flow path
COR0022302001080K	Predict the consequences a malfunction of the following will have on the RHR system: Reactor water level

Related References	
2.2.69	Procedure 2.2.69, Residual Heat Removal System
2.4SDC	RHR Loss of Shutdown Cooling

Related Skills (K/A)	
203000.A2.02	Ability to (a) predict the impacts of the following on the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of...: (CFR: 41.5 / 45.6) Pump trips (3.5/3.5)

QUESTION: 14028

"A" loop of RHR is in shutdown cooling at 150°F with "C" RHR pump running.

- ➢ RPV water level drops unexpectedly
- ➢ RPV water level continues to lower to -120 inches Wide Range

3 minutes later, what is the status of "A" and "C" LPCI pumps and what are the MIMIMUM actions are necessary to inject with **BOTH** pumps?
(NOTE: The choices are listed from MIMIMUM to MAXIMUM.)

a. "A" LPCI pump is running, "C" LPCI pump is idle. Take the control switch for the "C" LPCI pump momentarily to STOP and then to START.

b. Both pumps are idle. Align pump suction paths to the Torus. Take the control switches for the non-running pumps momentarily to STOP and then to START.

c. "A" LPCI pump is running, "C" LPCI pump is idle. Align pump suction paths to the Torus. Take the control switch for the "C" LPCI pump momentarily to STOP and then to START. Press SDC ISOL RESET VLV 25A button.

d. Both pumps are idle. Align pump suction paths to the Torus. Take the control switches for the non-running pumps momentarily to STOP and then to START. Press SDC ISOL RESET VLV 25A button.

ANSWER: 14028

d. Both pumps are idle. Align pump suction paths to the Torus. Take the control switches for the non-running pumps momentarily to STOP and then to START. Press SDC ISOL RESET VLV 25A button.

Answer source: 2.4SDC p. 13, Attachment 3

Distractors:

a. Both pumps trip and lock out on anti-pump. The SDC isolation must be reset. The suction path must be realigned.
b. The SDC isolation must be reset.
c. Both pumps trip and lock out on anti-pump.

Source: *New*

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
34	2127	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0022602 OPS ROD WORTH MINIMIZER

Related Objectives	
COR0022602001010A	State the purpose of the following items related to the Rod Worth Minimizer: Rod Worth Minimizer System
COR00226020010300	State the design bases for the RWM system as described in the associated student text.
COR0022602001050M	Briefly describe the following concepts as they apply to the RWM: Minimize clad damage during control rod drop accident (CRDA)

Related References	
4.2	Procedure 4.2, Rod Worth Minimizer

Related Skills (K/A)	
2.1.28	Knowledge of the purpose and function of major system components and controls. (3.2/3.3)

QUESTION: 2127

Why is the RWM required below 10% power but **not** above 10%?

At higher power,

a. fewer rod movements occur which reduces the chances of an error.

b. the Rod Block Monitor prevents fuel damage in the event of a rod drop accident.

c. the effects of a rod drop accident are less due to increased voiding causing lower rod worths.

d. the effects of a rod drop accident are less due to increased moderator temperature causing lower rod worths.

ANSWER: 2127

c. the effects of a rod drop accident are less due to increased voiding causing lower rod worths.

Answer source: COR00-2-26-02, p. 9, Section "e"

Distractors:

a. voids reduce rod worths.
b. RBM mitigates rod withdrawal error.
d. Moderator temperature rise increases rod worths.

Source: ***Direct from Bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
35	14033	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0022802 OPS STANDBY GAS TREATMENT

Related Objectives	
COR0022802001100K	Predict the consequences of the following on the Standby Gas Treatment system: Z sump failures

Related References	
2.2.73	Procedure 2.2.73, Standby Gas Treatment System

Related Skills (K/A)	
261000.A1.01	Ability to predict and/or monitor changes in parameters associated with operating the STANDBY GAS TREATMENT SYSTEM controls including: (CFR: 41.5 / 45.5) System flow (2.9/3.1)

QUESTION: 14033

The plant was operating at power when an LOCA occurred. The reactor building ventilation isolated and SGT automatically started. Additional plant failures occurred that resulted in the failure of Z-sump pumps and a subsequent high level in the Z-sump.

What is the potential effect on SGT?

a. SGT flow decrease

b. Inlet HEPA filter moisture damage

c. Moisture impingment on the SGT fans

d. Increased iodine carryover at the SGT train outlet

ANSWER: 14033

a. SGT flow decrease.

The discharge lines have drain lines that are connected to the Z sumps located at the base of the ERP. These SGT discharge lines can become blocked by excessive water level in the Z sump. If water collects in the 10" underground lines, SGT discharge flow may be restricted. Reduced SGT system flow effects the operability of the SGT systems.

Answer source: COR002-28-02, p. 23 & 24, section 3

Distractors:

b. High Z-sump level would not increase the moisture at the inlet of the SGT train.
c. A high level in the Z-sump would not impact the SGT fan.
d. A high level in the Z-sump would not result in increased iodine at the outlet of the train. The reduced flow through the train may even slightly reduce iodine at the outlet.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
36	14025	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0022902 STANDBY LIQUID CONTROL

Related Objectives	
COR0022902001120A	Briefly describe the relationships that exist between the SLC system and the following: Core Spray line leak detection

Related References	
COR0022902	SLC
2.2.9	Procedure 2.2.9, Core Spray System

Related Skills (K/A)	
211000.K1.09	Knowledge of the physical connections and/or cause-effect relationships between STANDBY LIQUID CONTROL SYSTEM and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) Core spray system: Plant-Specific (3.2*/3.4*)

QUESTION: 14025

Where do the Core Spray Line Break Detection differential pressure switches (dPIS-43A/B) connect?

The high pressure side of the pressure switch is connected to the SLC sparger to sense pressure . . .

a. below the core plate and the low pressure side of the switch senses pressure upstream of the Core Spray injection check valve.

b. below the core plate and the low pressure side of the switch senses pressure downstream of the drywell Core Spray manual isolation valve.

c. above the core plate in the core bypass region and the low pressure side of the switch senses pressure upstream of the Core Spray injection check valve.

d. above the core plate in the core bypass region and the low pressure side of the switch senses pressure downstream of the drywell Core Spray manual isolation valve.

ANSWER: 14025

d. above the core plate in the core bypass region and the low pressure side of the switch senses pressure downstream of the drywell Core Spray manual isolation valve.

Downstream of the manual isolation valve (14A/B) each Core Spray system has an instrument line that is connected to the low pressure side of the Core Spray Line Break Detection differential pressure switch. The high pressure side of the dPIS is connected to the Standby Liquid Control "outer" pipe, which detects the pressure in the bypass region above the Core Plate.

Answer source: COR002-29-02 Figure 7

Distractors:

a. The high pressure side senses above the core plate NOT below the core plate and the low pressure side of the switch senses pressure downstream of the drywell Core Spray manual isolation valve NOT upstream of the injection check valve.
b. The high pressure side senses above the core plate NOT below the core plate.
c. The low pressure side of the switch senses pressure downstream of the drywell Core Spray manual isolation valve NOT upstream of the injection check valve.

<u>Source:</u> ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
37	5348	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0023002 SOURCE RANGE MONITOR SUBSYSTEM

Related Objectives	
COR0023002001060F	Describe the SRM system design features and/or interlocks that provide for the following: IRM/SRM interlock

Related References	
4.1.1	Procedure 4.1.1, Source Range Monitoring System

Related Skills (K/A)	
215004.K3.02	Knowledge of the effect that a loss or malfunction of the SOURCE RANGE MONITOR (SRM) SYSTEM will have on following: (CFR: 41.7 / 45.4) Reactor manual control: Plant-Specific (3.4/3.4)

QUESTION: 5348

A Reactor Startup is in progress with the following conditions:

- ➢ Power is rising with a stable, positive period
- ➢ SRM detectors are withdrawn except for SRM "A" which fails to withdraw
- ➢ The SRM UPSCALE OR INOPERATIVE alarm has been received
- ➢ The SRM is **NOT** bypassed

As power continues to rise, what is the **FIRST** point that rods will be able to be withdrawn?

a. Associated IRMs are on Range 3 or higher.

b. Associated IRMs are on Range 8 or higher.

c. Associated IRMs are on Range 9 or higher.

d. The Mode switch is placed to RUN.

ANSWER: 5348

b. Associated IRMs are on Range 8 or higher.

The SRM Upscale or Inop Rod Block is bypassed when all associated IRM's are selected to range 8 or above.

Answer source: 4.1.1 p. 5, step 1.2.2

Distractors:

a. Range 3 bypasses the detector withdrawal permissive interlock of 100 cps.
c. Range 9 has no bypass functions.
d. While RUN bypasses all SRM Interlocks/Trips, Range 8 will be achieved first.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
38	19084	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0023102 OPS TRAVERSING IN-CORE PROBE

Related Objectives	
COR0023102001110A	Describe the TIP system design features and/or interlocks that provide for the following: Primary containment isolation
COR0023102001130C	Given a TIP system control manipulation, predict and explain the changes in the following parameters: Valve status
COR0023102001140H	Predict the consequences of the following on the TIP system: High primary containment pressure
COR0023102001160B	Given plant conditions, determine if any of the following TIP actions should occur: Ball valve closure

Related References	
4.1.4	Procedure 4.1.4, Traversing In-Core Probe System

Related Skills (K/A)	
215001.K6.04	Knowledge of the effect that a loss or malfunction of the following will have on the TRAVERSING IN-CORE PROBE: (CFR: 41.7 / 45.7) Primary containment isolation system: Mark-I&II (Not- BWR1) (3.1/3.4)

QUESTION: 19084

The plant is at 100% power with TIP traces in progress. Only "A" TIP machine is being used at this time. Currently, TIP "A" has reached the Core Bottom Limit and is moving at slow speed to the Core Top Limit. The IN-CORE light is ON.

A reactor scram due to low RPV water level occurs. One (1) minute later, an operator observes:

- ➢ TIP valve indication on Containment Isolation display (Panel 9-3) is RED
- ➢ **IN-SHIELD** light for TIP "A" is ON at Panel 9-13
- ➢ Drywell pressure is normal

What action is ***required***?

a. Fire TIP "A" shear valve.

b. Close TIP "A" ball valve.

c. Manually retract TIP "A" to fire the shear valve.

d. Manually retract TIP "A" to close the ball valve.

ANSWER: 19084

b. Close TIP "A" ball valve.

If red light (Panel 9-3) stays on, at least one TIP ball valve has not closed. After automatic withdrawal of the TIP on the PCIS group 2 isolation signal, the ball valve failed to automatically close.   This failed automatic action requires immediate operator action to manually perform the ball valve closure. The procedure directs the operator to attempt to manually retract TIP. Since the TIP is already retracted (IN-SHIELD light is on), this action is not necessary. If ball valve cannot be closed and there are indications of a reactor coolant leak in drywell (as evidenced by the high drywell pressure) then fire appropriate shear valve by operating appropriate keylock switch.

Answer source: 4.1.4 p. 2, step 2.10
4.1.4 p. 8, step 6.3

Distractors:

a. There is no indications of a LOCA and no attempt has yet been made to close the ball valve.
c. The TIP has already retracted and retracting the TIP does not fire the shear valve.
d. The TIP has already retracted.

Source: Direct from Bank

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
39	14004	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0023202 OPS REACTOR VESSEL LEVEL CONTROL

Related Objectives	
COR0023202001070D	Given a RVLC system control manipulation, predict and explain the changes in the following parameters: Controller Indications
COR0023202001060C	Predict the consequences of the following on the RVLC system: Control Signal Failure/Track and Hold

Related References	
2.4RXLVL	RPV WATER LEVEL CONTROL TROUBLE
2.2.28.1	Procedure 2.2.28.1, Feedwater System Operation

Related Skills (K/A)	
259002.A1.04	Ability to predict and/or monitor changes in parameters associated with operating the REACTOR WATER LEVEL CONTROL SYSTEM controls including: (CFR: 41.5 / 45.5) Reactor water level control controller indications (3.6/3.6)

QUESTION: 14004

Given the following conditions:

- ➢ Reactor power is 90%
- ➢ Reactor water level is 35"
- ➢ RVLC is in 3 element
- ➢ The Master Controller is in BAL the tape setpoint is 35"
- ➢ The selected level instrument **INSTANTANEOUSLY** fails downscale

What is the current configuration of RFC-CS-RFPTA (RFPT A M/A station)?

The RFPT M/A station shifts to __________ mode with controller output ____________ output prior to the event.

a. MDEM the same as

b. MDVP the same as

c. MDEM higher than

d. MDVP higher than

ANSWER: 14004

a. MDEM, the same as

Answer source: New Lovejoy training material (no electronic version). New electronic versions of 2.2.28.1 and 2.4RXLVL not yet available without exam compromise.

Distractors:

b. The controller shifts to MDEM mode.
c. The output won't change.
d. The controller shifts to MDEM mode and the output won't change.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
40	14014	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0023402 Alternate Shutdown (LO)

Related Objectives	
COR00234020010700	State the design bases for the ASD system as described in the associated Student Text.
COR00234020010100	State the purpose of the Alternate Shutdown system.

Related References	
5.1ASD	Shutdown From Outside The Control Room

Related Skills (K/A)	
295016.AK3.03	Knowledge of the reasons for the following responses as they apply to CONTROL ROOM ABANDONMENT: (CFR: 41.5 / 45.6) Disabling control room controls. (3.5/3.7*)

QUESTION: 14014

When the control room is evacuated, why are the ASD panel isolation switches placed in the ISOLATE Position?

a. To prevent spurious equipment operation.

b. To ensure automatic operation of ECCS remains available.

c. To isolate circuits to meet divisional physical separation criteria.

d. To prevent overloading the associated DG during a design basis LOCA.

ANSWER: 14014

a. To prevent spurious equipment operation.

Answer source: COR002-34-02 p. 11, section 4

Distractors:

b. Automatic operation of ECCS does NOT remain available.
c. Operation of these switches has nothing to do with divisional separation.
d. Operation of these switches has nothing to do with diesel loading.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
41	8970	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0070501	OPS Introduction to Technical Specifications

Related Objectives	
INT00705010010800	From memory, state each CNS Safety Limit and discuss the basis for each of the Safety Limits.

Related References	
2.1.1.1	With the reactor steam dome pressure < 785 psig or core flow < 10% rated core flow: THERMAL POWER shall be < 25% RTP.

Related Skills (K/A)	
290002.K5.07	Knowledge of the operational implications of the following concepts as they apply to REACTOR VESSEL INTERNALS: (CFR: 41.5 / 45.3) Safety limits (3.9/4.4)

QUESTION: 8970

The plant was operating at 100% power when a DEH failure results in a reactor pressure reduction. Reactor pressure decreases to 700 psig and reactor power decreases to 65%. The Group 1 isolation fails to actuate. The operating crew scrams the reactor and manually closes the Main Steam Isolation Valves.

What is a potential consequence of this event?

a. Increased likelihood of thermal hydraulic instabilities.

b. The linear heat generation rate limit for some fuel is exceeded.

c. The average planar linear heat generation rate limit is exceeded.

d. The potential is created for radioactive release in excess of 10CFR100 limits.

ANSWER: 8970

d. The potential is created for radioactive release in excess of 10CFR100 limits.

The scenario given represents the violation of the fuel integrity safety limit. Reactor power is greater than 25% with reactor pressure less than 785 psig. Exceeding a safety limit may cause fuel damage and create the potential for radioactive releases in excess of 10CFR100.

Answer source: Safety Limit Violation Bases p. B 2.0-5

Distractors:

a. The likelihood of thermal hydraulic instabilities is actually reduced by the decrease in core inlet subcooling casued by the rapid pressure reduction.
b. This reduction in power is global resulting in a power reduction for each core bundle. This would increase the margin to LHGR limit.
c. With the global reduction in power average linear heat generation rate would decrease and put operation farther away from the limit.

Source: *Modified from 8970*

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
42	3995	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0070502	CNS Tech. Spec. 3.1, Reactivity Control Systems

Related Objectives	
INT0070502001060A	From memory, for MODES 1 and 2, state the actions required in less than one hour for: two or more control rod scram accumulators inoperable with the reactor steam dome pressure less than or equal to 940 psig (LCO 3.1.5 B.1).

Related References	
3.1.5	Control rod scram accumulators

Related Skills (K/A)	
295022.AK2.03	Knowledge of the interrelations between LOSS OF CRD PUMPS and the following: (CFR: 41.7 / 45.8) Accumulator pressures. (3.4/3.4)

QUESTION: 3995

Given the following conditions:

- ➢ A Reactor startup **AND** heatup is in progress
- ➢ Reactor power is 3%
- ➢ Reactor Steam Dome Pressure is 835 psig
- ➢ Control Rod Drive Hydraulic Pump 1A trips and will not restart
- ➢ Control Rod Drive Hydraulic Pump 1B will not start

What action(s) are required by Technical Specifications for these conditions?

When the accumulator pressure is < 935 psig for . . .

a. **ANY** control rod, immediately declare the associated Control Rod inoperable.

b. **ANY** control rod, immediately place the Reactor Mode Switch in SHUTDOWN.

c. one **withdrawn** control rod, immediately place the Reactor Mode Switch in SHUTDOWN.

d. one **withdrawn** control rod, restore Charging Header pressure to greater than 940 psig within 20 minutes.

ANSWER: 3995

c. one **withdrawn** control rod, immediately place the Reactor Mode Switch in SHUTDOWN.

Tech Spec 3.1.5 Condition C applies. With RPV Pressure < 900 psig, withdrawn rods with inoperable accumulators may fail to scram under low pressure conditions and must be immediately inserted. Since the rod cannot be inserted without drive pressure, Condition D applies and the Reactor must be scrammed.

Answer source: Actions per LCO 3.1.5 for reactor pressure < 900 psig (RA C.1 & D.1).

Distractors:

a. This is the action for a slow control rod with an inoperable accumulator.
b. The control rod must be withdrawn to require the scram.
d. This is the action if Reactor Pressure is > 900 psig.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
43	14048	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0070504	CNS Tech. Spec. 3.3, Instrumentation

Related Objectives	
INT00705040010200	Discuss the applicable Safety Analysis in the Bases associated with each Section 3.3 Specification.

Related References	
3.3.2.2	Feedwater and main turbine high water level trip instrumentation

Related Skills (K/A)	
295014.AK3.01	Knowledge of the reasons for the following responses as they apply to INADVERTENT REACTIVITY ADDITION: (CFR: 41.5 / 45.6) Reactor SCRAM. (4.1*/4.1)

QUESTION: 14048

Why does Technical Specifications require the main turbine to trip on high reactor water level?

a. To indirectly prevent damage to the Moisture Separators by low enthalpy fluid.

b. To prevent ECCS equipment damage from missiles created by main turbine failure.

c. To ensure flow induced vibration of the main steam lines remains within analytical limits.

d. To indirectly provide a reactor scram to mitigate the reduction in MCPR during a feedwater controller maximum demand failure.

ANSWER: 14048

d. To indirectly provide a reactor scram to mitigate the reduction in MCPR during a feedwater controller maximum demand failure.

Per 3.3.2.2 bases "The feedwater and main turbine high water level trip instrumentation is assumed to be capable of providing a turbine trip in the design basis transient analysis for a feedwater controller failure, maximum demand event. The Level 8 trip indirectly initiates a reactor scram from the main turbine trip (above 30% RTP) and trips the feedwater pumps, thereby terminating the event. The reactor scram mitigates the reduction in MCPR."

Answer source: Tech Spec Bases p. 3.3-55, Safety Analysis

Distractors:

a. This is not the basis.
b. The missile damage potential is not assumed by the accident analysis.
c. There is no flow induced vibration related to the main turbine trip, but is related to Recirculation flow mismatch.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
44	14042	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0070505	CNS Tech. Spec. 3.4, Reactor Coolant System (RCS)

Related Objectives	
INT00705050010100	Given a set of plant conditions, recognize non-compliance with a Chapter 3.4 LCO.

Related References	
6.LOG.601	Daily Surveillance Log (Tech Specs)

Related Skills (K/A)	
268000.K3.04	Knowledge of the effect that a loss or malfunction of the RADWASTE will have on following: (CFR: 41.5 / 45.3) Drain sumps (2.7/2.8)

QUESTION: 14042

The plant was operating at power when the Sump F Totalizer (RW-FQ-527) failed.

What is required?

Place/Leave . . .

a. **either** F-1 or F-2 sump pumps in AUTO and repair the totalizer within 4 hours or be in mode 3 within 12 hours and mode 4 within 36 hours.

b. **both** F-1 and F-2 sump pumps in AUTO and repair the totalizer within 12 hours or be in mode 3 within 12 hours and mode 4 within 36 hours.

c. **either** F-1 or F-2 sump pump switches in PULL-TO-LOCK and leave the other pump in AUTO. Record each time the pump in AUTO pumps.

d. **both** F-1 and F-2 sump pump switches to PULL-TO-LOCK. At 8 hour intervals and also when Sump F high alarm is received, pump the sump using one pump and record seconds of operation.

ANSWER: 14042

d. F-1 and F-2 sump pump switches to PULL-TO-LOCK.

At 8 hour intervals and also when Sump F high alarm is received, pump the sump using one pump and record seconds of operation. When the Sump F Totalizer failed, 6.LOG.601 requires that the total gallons be calculated per Sump F Totalizer table. This table directs that both sump pump switches be placed in Pull-To-Lock and on 8 hour intervals and also when Sump F high alarm is received, pump sump using one pump and time seconds of operation.

Answer source: 6.LOG.601 pp. 9 (Note a) & 10 (DETERMINATION OF TOTAL GALLONS WITH FAILED SUMP F TOTALIZER)

Distractors:

a. The pumps are not placed/left in AUTO and 30 days are allowed to repair the totalizer.
b. The pumps are not placed/left in AUTO and 30 days are allowed to repair the totalizer.
c. Neither pump is left in AUTO.

Source: ***New***

Provide to Candidate: T.S. 3.0 section and bases, T.S. LCO 3.4.4 and bases, T.S. LCO 3.4.5 and bases.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
45	14051	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0070507	CNS Tech. Spec. 3.6, Containment Systems

Related Objectives	
INT00705070010300	Given a set of plant conditions that constitutes non-compliance with a Chapter 3.6 LCO, determine the ACTIONS that are required.
INT00705070010100	Given a set of plant condtions, recognize non-compliance with a Chapter 3.6 LCO.

Related References	
3.6.1.5	Drywell air temperature
3.6.1.4	Drywell pressure

Related Skills (K/A)	
295010.AK1.03	Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL PRESSURE: (CFR: 41.8 to 41.10) Temperature increases. (3.2/3.4)

QUESTION: 14051

The plant was operating at power when drywell pressure and temperature began rising. The following parameters existed at 0700 on 5/20:

- ➢ Drywell pressure 0.8 psig (rising)
- ➢ Drywell temperature 142°F (rising)

The crew then reduced reactor power using Reactor Recirculation flow and began venting the drywell and the torus. At 0800 on 5/20, the crew discovered the following primary containment parameters:

- ➢ Drywell pressure 0.45 psig (stable)
- ➢ Drywell temperature 151°F (rising)

IF conditions do not improve, what is the LATEST time that the plant is ***allowed*** to enter **MODE 3** by Technical Specifications?

Be in MODE 3 by . . .

a. 1600 on 5/20.

b. 0300 on 5/21.

c. 0400 on 5/21.

d. 0300 on 5/22.

ANSWER: 14051

c. 0400 on 5/21.

Enter 3.6.1.4 at 0700 (RA 3.6.1.A). Exit 3.6.1.4 at 0800. Enter 3.6.1.5 at 0800, RA A.1. Enter RA B.1 (B.2) at 1600 on 5/20. Be in MODE 3 by 0400 on 5/21.

Answer source: Tech Spec LCO 3.6.1.4 p. 3.6-16

Distractors:

a. Time when RA 3.6.1.5.B.1 must be entered, not completed.
b. Time to MODE 3 if use wrong start time (0700).
d. Time to MODE 4 if use wrong start time (0700).

Source: ***New***

Provide to Candidate: T.S. 3.0 section and bases, T.S. LCO 3.6.1.4 and bases, 3.6.1.5 and bases.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
46	5247	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0080605	OPS FLOWCHART 1A - RPV CONTROL/RPV PRESSURE

Related Objectives	
INT0080605001010A	List the entry conditions of Flowchart 1A: Describe the importance of each in an emergency situation.

Related References	
EOP 1A, RPV CON	RPV Control

Related Skills (K/A)	
2.4.2	Knowledge of system set points / interlocks and automatic actions associated with EOP entry conditions. (CFR: 41.7 / 45.7 / 45.8) (3.9/4.1)

QUESTION: 5247

The plant is at 35% power when Turbine vibration required the operator to insert a manual scram and trip the turbine. The following conditions exist one (1) minute after the operator has depressed both manual reactor scram pushbuttons:

- ➢ Turbine Bypass Valves are 75% open **(stable)**
- ➢ Main Steam Isolation Valves (MSIVs) are open
- ➢ RPV Narrow Range level is 25" **AND steady**
- ➢ RPV pressure is 940 psig **AND steady**

Which procedure(s) must be executed?

2.1.5 "Reactor Scram" . . .

a. **ONLY**.

b. **AND** EOP-1A "RPV Control" **ONLY**.

c. **AND** EOP-6A "Reactor Pressure/Power (Failure to Scram)" **AND** EOP-7A "Reactor Level (Failure to Scram)" **ONLY**.

d. **AND** EOP-1A "RPV Control" **AND** EOP-6A "Reactor Pressure/Power (Failure to Scram)" **AND** EOP-7A "Reactor Level (Failure to Scram)".

ANSWER: 5247

d. **AND** EOP-1A "RPV Control" **AND** EOP-6A "Reactor Pressure/Power (Failure to Scram)" **AND** EOP-7A "Reactor Level (Failure to Scram)".

Bypass valves at 75% open is approximately 19% power. 19% power after a scram is an entry condition to 1A. EOP 1A directs 6A and 7A to be entered.

<u>**Answer source:**</u> EOP-1A, 2.1.5 p. 1

Distractors:

a. EOP 1A must be entered.
b. Entry into 6A & 7A is required.
c. Entry into 1A is required.

Source: ***Direct from Bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
47	13407	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0080605	OPS FLOWCHART 1A - RPV CONTROL/RPV PRESSURE

Related Objectives	
INT00806050010900	Identify any EOP support procedures addressed in Flowchart 1A and apply any associated special operating instructions or cautions.

Related References	
5.8.2	Procedure 5.8.2, Alternate Emergency Depressurization Systems (Table 2)

Related Skills (K/A)	
2.3.10	Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure. (CFR: 43.4 / 45.10) (2.9/3.3)

QUESTION: 13407

While responding to a LOCA, the RCIC overspeed trip must be reset to allow it to be used as an injection system.

- ➢ The TSC is NOT operational.
- ➢ Several Reactor Building ARMs that an operator must pass within 10 feet of to get to the area are alarming and indicate upscale.
- ➢ Both high range Drywell radiation monitors read 1.5E4 R/hr.

In addition to standard RP practices, what additional (if any) <u>MINIMUM</u> requirement must be met to dispatch an operator to perform this task?
(NOTE: The choices are arranged in MIMIMUM to MAXIMUM order.)

a. No additional requirements must be met.

b. A survey instrument that monitors radiation dose rates must be taken.

c. A RP Technician must accompany the operator.

d. The TSC must be declared operational.

ANSWER: 13407

d. The TSC must be declared operational.

If DRYWELL RAD MONITOR RMA-RM-40A or DRYWELL RAD MONITOR RMA-RM-40B (Panel 9-02) is reading $\geq$ 1E4 rem/hour, entry into Secondary Containment is prohibited until TSC is operational and personnel can be dispatched per Procedure 5.7.15. The operator cannot be dispatched until the TSC is operational because both drywell radiation monitors are above 1E4 rem/hr.

<u>Answer source:</u> 5.8.2 p. 12, section 5.1

Distractors:

a. The TSC must also be operational to dispatch the operator.
b. The TSC must also be operational to dispatch the operator.
c. The TSC must also be operational to dispatch the operator.

<u>Source:</u> *Direct from Bank*

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
48	14000	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
COR0020302	CONTAINMENT
INT0080605	OPS FLOWCHART 1A - RPV CONTROL/RPV PRESSURE

Related Objectives	
COR0020302001210B	Given plant conditions, determine if the following should have occurred: Any of the PCIS group isolations.
INT0080605001010A	List the entry conditions of Flowchart 1A: Describe the importance of each in an emergency situation.

Related References	
2.1.22	Procedure 2.1.22, Recovering From A Group Isolation

Related Skills (K/A)	
2.4.2	Knowledge of system set points / interlocks and automatic actions associated with EOP entry conditions. (CFR: 41.7 / 45.7 / 45.8) (3.9/4.1)

QUESTION: 14000

RPV water level lowers to 20 inches below the value that is an entry condition for EOP-1A, "RPV Control". No other EOP entry conditions are satisfied.

What group isolations have automatically occurred?

Group 2 . . .

a. **only**.

b. and Group 3 **only**.

c. and Group 3 and Group 6 **only**.

d. and Group 1 and Group 3 and Group 6.

ANSWER: 14000

c. and Group 3 and Group 6 **only**.

Answer source: 2.1.22 p. 7, section 5.3
2.1.22 p. 10, section 6.3
2.1.22 p. 17, section 9.6

Distractors:

a. Group 3 and Group 6 also occur.
b. group 6 also occurs.
d. Group 1 does not occur.

2.4.2 Knowledge of system set points / interlocks and automatic actions associated with EOP entry conditions. (CFR: 41.7 / 45.7 / 45.8) as it applies to: 295009 Low Reactor Water Level

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
49	17991	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0080605	OPS FLOWCHART 1A - RPV CONTROL/RPV PRESSURE

Related Objectives	
INT00806050010600	Identify the characteristics of a "cycling SRV" and the reasons the cycling must be stopped.
INT00806050011100	Given an EOP flowchart 1A, RPV CONTROL step, state the reason for the actions contained in the step.

Related References	
INT0080605	Flowchart 1A - RPV Control/RPV Pressure
PSTG	Plant Specific Technical Guideline

Related Skills (K/A)	
295025.EA1.02	Ability to operate and/or monitor the following as they apply to HIGH REACTOR PRESSURE: (CFR: 41.7 / 45.6) Reactor/turbine pressure regulating system (3.8/3.8)

QUESTION: 17991

When executing EOP flowchart 1A-RPV CONTROL, the operator is directed to open SRVs to reduce RPV pressure to 940 psig if SRVs are cycling.

What is basis for selecting the value of 940 psig?

940 psig is . . .

a. the lowest pressure that will result in fully open Main Turbine bypass valves.

b. low enough to provide a sufficient margin below the high pressure scram setpoint.

c. high enough to provide a sufficient margin above the shutoff head of CS and RHR pumps.

d. low enough to prevent the actuation of SRVs on mechanical setpoints and prevents arming of LLS.

ANSWER: 17991

a. the lowest pressure that will result in fully open Main Turbine bypass valves.

RPV pressure reduction with SRVs is continued until RPV pressure reaches the pressure at which steam flow through the main turbine BPVs is at 100% of BPV capacity. If the MSIVs are open, reducing RPV pressure to below this value results in partial closure of the BPVs and a corresponding rise in the amount of steam discharged to the suppression pool through the SRVs. If the MSIVs are not open, reducing RPV pressure to the lowest pressure at which all BPVs would be fully open, if controlling pressure, provides an adequate operating margin below the setpoint pressure of the lowest lifting SRV.

Answer source: INT008-06-05 p. 15

Distractors:

b. While this would provide a margin to the scram setpoint, it is not the reason why this action is taken in EOP-1A.

c. While this pressure does provide a substantial margin to the shutoff head of the CS and RHR pumps it is not the reason for this action.

d. While this pressure is below the lowest lift setpoint of the relief valves prevention of LLS arming and actuation is not part of the reason. Verification of Low Low Set operation may ensure this action is performed automatically.

Source*: Direct from Bank*

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
50	19034	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0080618	EOP AND SAG GRAPHS
INT0080605	OPS FLOWCHART 1A - RPV CONTROL/RPV PRESSURE

Related Objectives	
INT00806050011000	Given plant conditions and EOP flowchart 1A, RPV CONTROL, determine required actions.
INT00806180010200	For each graph used in the flowcharts, identify the action(s) required if the parameters associated indicate operation in the restricted or prohibited area.
INT00806050011200	Given plant conditions, assess if RPV water level can be determined or not.

Related References	
5.8	Procedure 5.8, Emergency Operating Procedures (EOPs)
EOP 1A, RPV CON	RPV Control

Related Skills (K/A)	
226001.A1.02	Ability to predict and/or monitor changes in parameters associated with operating the RHR/LPCI: CONTAINMENT SPRAY SYSTEM MODE controls including: (CFR: 41.5 / 45.5) Containment/drywell temperature (3.4/3.5)

QUESTION: 19034

A Loss of Coolant Accident has occurred with the following conditions:

➢	Reactor pressure	270 psig (lowering at 5 psig/minute)
➢	Indicated water level	-120" (Wide range, steady)
➢	Drywell pressure	5.5 psig (rising slowly)
➢	Drywell temperature	350° F (all points) (steady)

What is the status of Wide Range Reactor Level Instrumentation ***now*** and what effect (if any) will drywell sprays have on ***future*** availability?

Wide Range Reactor Level Instrumentation . . .

a. can be used for trending.
Initiation of drywell sprays will help maintain instrument availability.

b. can be used for trending.
Initiation of drywell sprays will have **NO** effect on future instrument availability.

c. **CANNOT** be used for trending.
Initiation of drywell sprays will restore instrument availability.

d. **CANNOT** be used for trending.
Initiation of drywell sprays will have **NO** effect on future instrument availability.

ANSWER: 19034

a. can be used for trending.
Initiation of drywell sprays will help maintain instrument availability.

Indicated WR Level is above the minimum Indicated Level of EOP Graph 15 for 350° F so the instrument can be used for trending purposes. Initiation of DW sprays cools containment and prevents intrusion into the unsafe region of reactor saturation graph (EOP Graph 1), in addition DW sprays and their cooling effect on DW temperature ensure that the instrumet **stays** in the safe region of Graph 15.

Answer source: EOP Graph 1 & EOP Graph 15, effects of drywell spray

Distractors:

b. Drywell sprays will maintain Instrument availability.
c. The instruments are currently available for trending.
d. The instruments are currently available for trending and drywell sprays will maintain Instrument availability.

Source: ***Direct from bank***

Provide to Candidate: EOP Graphs

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
51	14001	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0080606	FLOWCHART 6A - RPV PRESSURE/POWER FAILURE-TO-SCRAM

Related Objectives	
INT00806060011000	Given plant conditions and ESP 5.8.3, ALTERNATE ROD INSERTION METHODS, determine which methods would successfully insert control rods.
INT00806060010600	List the methods of alternate rod insertion.

Related References	
5.8.3	Procedure 5.8.3, Alternate Rod Insertion Methods

Related Skills (K/A)	
295015.AK2.04	Knowledge of the interrelations between INCOMPLETE SCRAM and the following: (CFR: 41.7 / 45.8) RPS (4.0/4.1)

QUESTION: 14001

The plant was operating at 100% power with a half scram on RPS A due to a relay failure and 1B CRD pump tagged out of service. Subsequently, power was lost to RPSPP 1B. The following conditions now exist:

- Many control rods failed to insert
- Reactor power is 5%
- 4160 VAC Bus 1F is de-energized

What method can the crew use to successfully insert the control rods that failed to insert on the scram?

a. Individually scram control rods.

b. Drain the SDV and manually scram.

c. Manually vent CRDM over-piston area.

d. Insert control rods using emergency override switch.

ANSWER: 14001

c. Manually vent CRDM over-piston area.

The multiple RPS failures result in the inability to reset the scram. In addition, having the 1B CRD pump out of service in conjunction with the loss of power to the operating CRD pump precludes the use of RMCS to drive control rods. Venting the over piston area of the CRDM is the only action listed that is consistent with 5.8.3, Alternate Rod Insertion Methods.

Answer source: 5.8.3 flowchart, path "B"

Distractors:

a. The scram valves cannot be closed due to the RPS failures.
b. Draining the SDV and manually scramming would require that the scram valves be closed which is not possible with the RPS failures.
d. RMCS cannot be used to drive control rods because there are no operating CRD pumps.

<u>Source</u>: *New*

<u>Provide to Candidate:</u> 5.8.3 Flowchart

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
52	16472	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0080606	FLOWCHART 6A - RPV PRESSURE/POWER FAILURE-TO-SCRAM

Related Objectives	
INT00806060010600	List the methods of alternate rod insertion.
INT00806060011000	Given plant conditions and ESP 5.8.3, ALTERNATE ROD INSERTION METHODS, determine which methods would successfully insert control rods.
INT00806060010900	Identify any EOP support procedures addressed in Flowchart 6A and apply any associated special operating instructions or cautions.

Related References	
5.8.3	Procedure 5.8.3, Alternate Rod Insertion Methods

Related Skills (K/A)	
295037.EK2.05	Knowledge of the interrelations between SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN and the following: (CFR: 41.7 / 45.8) CRD hydraulic system (4.0/4.1)

QUESTION: 16472

The plant scrammed from full power, but many control rods failed to insert (see following figure). Control rods will be inserted using the Reactor Manual Control System.

Which one of the following rod insertion sequences should be used?
(NOTE: Blank positions on the figure indicate control rod is fully inserted.)

Fully insert rod . . .

a. 26-27, then fully insert 26-39, then 30-27, then 34-27 and continue to insert rods that have high worth.

b. 26-27, then fully insert 22-23, then 22-31 and continue this spiral pattern outward from the center of the core.

c. 30-27, then fully insert 18-27, then 30-35 and continue to criss-cross the core outward from the center of the core.

d. 30-27, then fully insert 22-27, then 22-23 and continue this spiral pattern inserting the most withdrawn control rods first.

	02	06	10	14	18	22	26	30	34	38	42	46	50
51													
47													
43													
39				10			32	36					
35					12	26		36					
31				12		36			10				
27					12	48	46	42	32				
23						48			24				
19					10		34	12					
15													
11													
07													
03													

ANSWER: 16472

b. Fully insert rod 26-27, then fully insert 22-23, then 22-31 and continue this spiral pattern outward from the center of the core.

Under ATWS conditions, control rod insertions should begin in the center of the core, and proceed to every other control rod in an outward spiral pattern.

Answer source: 5.8.3 p. 4, section 4.2

Distractors:

a. Skips control rods, this method has no real pattern to it.
c. Does NOT start with the center rod and does not insert every other rod by geometric location.
d. Does NOT start with the center rod and does not insert every other rod by geometric location.

Source: ***Direct from Bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
53	16485	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0080610	OPS FLOWCHART 7A - RPV LEVEL FAILURE-TO-SCRAM
INT0080612	FLOWCHART 7B - RPV FLOODING FAILURE-TO-SCRAM
INT0080602	OPS FLOWCHART ORGANIZATION AND STRUCTURE

Related Objectives	
INT00806100010900	Given an EOP flowchart 7A, RPV LEVEL (FAILURE TO SCRAM) step, state the reason for the actions contained in the step.
INT00806120010200	Describe the conditions required to assure adequate core cooling while flooding the core during a failure to scram transient.
INT00806020010800	List the three mechanisms used in the EOP flowcharts to assure adequate core cooling.

Related References	
PSTG	Plant Specific Technical Guideline

Related Skills (K/A)	
295015.AK1.04	Knowledge of the operational implications of the following concepts as they apply to INCOMPLETE SCRAM: (CFR: 41.8 to 41.10) Reactor pressure: Plant-Specific. (3.8/3.8)

QUESTION: 16485

Which one of the following conditions, by itself, assures fuel clad temperature does not exceed 1500°F ***during an ATWS***?
(Note: All RPV levels are as INDICATED on the Fuel Zone instruments.)

Reactor Pressure . . .

a. 197 psig, ***indicated*** RPV level -50 inches, Two (2) SRVs open.

b. 351 psig, ***indicated*** RPV level -55 inches, Three (3) SRVs open.

c. 452 psig, ***indicated*** RPV level -66 inches, One (1) SRV open.

d. 790 psig, ***indicated*** RPV level -60 inches, **NO** SRVs open.

ANSWER: 16485

b. 351 psig, ***indicated*** RPV level -55 inches, Three (3) SRVs open.

Actual level in all 4 choices is below -25 inches for adequate core cooling with level/steam cooling. Minimum Steam Cooling Pressure met only in choice "b" (MSCP for 3 SRVs is 280 psig).

Answer source: EOP flowchart 7A steps FS/L-12 & FS/L-17
EOP Flowcharts 7A & 7B, Table 14
INT008-06-06 p.20

Distractors:

a. Below -25" actual, below MSCP.
b. Below -25" actual, below MSCP.
d. Below -25" actual, below MSCP.

Source: ***Modified***

Provide to Candidate: EOP-1A, EOP-2A, EOP-2B, EOP-6A, EOP-7A and EOP Graphs with entry conditions and cautions removed.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
54	8939	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
INT0080610 OPS FLOWCHART 7A - RPV LEVEL FAILURE-TO-SCRAM

Related Objectives	
INT00806100010300	State the basis for intentionally lowering RPV water level and the criteria for the lowered level (LL).
INT00806100010900	Given an EOP flowchart 7A, RPV LEVEL (FAILURE TO SCRAM) step, state the reason for the actions contained in the step.

Related References	
INT0080610	Flowchart 7A RPV Level Failure to Scram
PSTG	Plant Specific Technical Guideline

Related Skills (K/A)	
295031.EA1.12	Ability to operate and/or monitor the following as they apply to REACTOR LOW WATER LEVEL: (CFR: 41.7 / 45.6) Feedwater. (3.9/4.1*)

QUESTION: 8939

Why is RPV level lowered during a failure-to-scram event to 100" (corrected fuel zone)?

To prevent or mitigate the consequences of . . .

a. any bundle exceeding the MCPR safety limit.

b. exceeding the peak cladding temperature of 1500 °F.

c. any large irregular neutron flux oscillations induced by neutronic/thermal-hydraulic instabilities.

d. exceeding the Heat Capacity Temperature Limit and challenging the Primary Containment Pressure Limit.

ANSWER: 8939

c. any large irregular neutron flux oscillations induced by neutronic/thermal-hydraulic instabilities.

<u>Answer source:</u> EOP-7A step FS/L-6

Distractors:

a. Lowering level to -100" FZ suppression any large irregular neutron flux oscillations but, there is on assurance that the MCPR safety limit will not be violated on a failure-to-scram event.
b. The lower limit (-25 in. (FZ)) is the Minimum Steam Cooling RPV Water Level (MSCRWL). Which is the basis for the 1500 ºF PCT.
d. 0 inches on Table 17 & 28 is based on the ability of SPC to remove heat and not challenge Pri Cont.

<u>Source:</u> *Direct from bank*

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
55	14047	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0080613	OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL

Related Objectives	
INT00806130011100	Given plant conditions and EOP Flowchart 3A, PRIMARY CONTAINMENT CONTROL, determine required actions.

Related References	
5.8	Procedure 5.8, Emergency Operating Procedures (EOPs)
EOP 3A, PCCP	EOP 3A Primary Containment Control Procedure

Related Skills (K/A)	
295013.AA2.01	Ability to determine and/or interpret the following as they apply to HIGH SUPPRESSION POOL TEMPERATURE: (CFR: 41.10 / 43.5 / 45.13) Suppression pool temperature (3.8/4.0)

QUESTION: 14047

Given the following conditions:

- ➢ Primary Containment water level 14 feet
- ➢ Reactor pressure 800 to 1000 psig (with SRVs)

Which of the following is the **LOWEST** average suppression pool water temperature which requires Emergency Depressurization?

a. 195°F

b. 205°F

c. 211°F

d. 215°F

ANSWER: 14047

b. 205°F

Answer source: EOP graph 7
EOP flowchart 3A step SP/T-5

Distractors:

a. Too low.
c. Too high.
d. Too high.

Source: ***New***

Provide to Candidate: EOP-1A and EOP-3A with entry conditions and cautions removed, EOP graphs.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
56	5268	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0080613	OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL

Related Objectives	
INT00806130011100	Given plant conditions and EOP Flowchart 3A, PRIMARY CONTAINMENT CONTROL, determine required actions.
INT00806130011000	Identify any EOP support procedures referenced in Flowchart 3A and apply any associated special operating instructions or cautions.

Related References	
EOP 3A, PCCP	EOP 3A Primary Containment Control Procedure

Related Skills (K/A)	
295012.AA2.02	Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE: (CFR: 41.10 / 43.5 / 45.13) Drywell pressure (3.9/4.1)

QUESTION: 5268

Given the following conditions:

- ➢ A small coolant leak has occurred in the Drywell
- ➢ Drywell temperature is 185°F and rising slowly
- ➢ Drywell pressure is 4.0 psig and rising slowly

What action is required to control Drywell conditions?

a. Initiate drywell sprays.

b. Operate all available drywell cooling.

c. Vent primary containment with torus vent line.

d. Vent primary containment with drywell vent line.

ANSWER: 5268

b. Operate all available Drywell Cooling.

This action is specified in the drywell temperature leg of EOP-3A when drywell temperature cannot be maintained below 150°F.

Answer source: EOP flowchart 3A step DW/T-3

Distractors:

a. Drywell sprays are not permitted with torus pressure at the current 2.6 psig.
c. Venting the torus is not allowed with the LOCA signal present and pressure well below PCPL-A.
d. Venting the drywell is not allowed with the LOCA signal present and pressure well below PCPL-A.

Source: ***Modified 5268***

Provide to Candidate: EOP-3A with entry conditions and cautions removed.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
57	5332	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0080613	OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL

Related Objectives	
INT00806130011100	Given plant conditions and EOP Flowchart 3A, PRIMARY CONTAINMENT CONTROL, determine required actions.
INT00806130011200	Given plant conditions and EOP flowchart 3A, PRIMARY CONTAINMENT CONTROL, state the reasons for the actions contained in the steps.
INT0080613001040A	State the basis for primary containment control actions as they apply to the following: Specific setpoints

Related References	
PSTG	Plant Specific Technical Guideline

Related Skills (K/A)	
2.1.20	Ability to execute procedure steps. (CFR: 41.10 / 43.5 / 45.12) (4.3/4.2)

QUESTION: 5332

Given the following conditions:

- A Loss of Coolant Accident has occurred
- Reactor pressure is 590 psig
- Drywell pressure is 6.5 psig **AND** rising
- Drywell temperature is 200° F
- Drywell Spray is in service with suction ***from the torus***
- Torus Temperature is 165° F
- Torus level is 16.8 feet **AND** rising
- Torus spray is in service

What containment spray action is required **AND** what is the bases for the action?

a. Terminate Drywell Spray to stop the water addition to the Torus.

b. Terminate Torus Spray since the Torus Spray Header is submerged.

c. Terminate Torus Spray to raise Torus pressure to drive non-condensable gases into the Drywell.

d. Terminate Drywell Spray because the primary containment vacuum relief system capacity has been exceeded.

ANSWER: 5332

d. Terminate Drywell Spray because the primary containment vacuum relief system capacity has been exceeded.

<u>Answer source:</u> EOP flowchart 3A steps DS-1 & DS-2
INT008-06-13 p. 11

Distractors:

a. Drywell Spray water can be taken from the Torus.
b. The Spray header is submerged at 26.75'.
c. The Vacuum Breakers will not pass sufficient flow when covered.

2.1.20 Ability to execute procedure steps. (CFR: 41.10 / 43.5 / 45.12) as it applies to: 295029 High Suppression Pool Water level / 5

Source: ***Direct from bank***

Provide to Candidate: EOP-3A with cautions and entry conditions removed.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
58	5333	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0080613	OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL

Related Objectives	
INT0080613001040A	State the basis for primary containment control actions as they apply to the following: Specific setpoints
INT00806130011200	Given plant conditions and EOP flowchart 3A, PRIMARY CONTAINMENT CONTROL, state the reasons for the actions contained in the steps.

Related References	
5.8	Procedure 5.8, Emergency Operating Procedures (EOPs)
PSTG	Plant Specific Technical Guideline

Related Skills (K/A)	
500000.EK3.04	Knowledge of the reasons for the following responses as they apply to HIGH PRIMARY CONTAINMENT HYDROGEN CONCENTRATIONS: (CFR: 41.5 / 45.6) Emergency depressurization (3.1/3.9)

QUESTION: 5333

What hydrogen **AND** oxygen concentration values require Emergency Depressurization **AND** what is the bases for the Emergency Depressurization?

Containment hydrogen AND oxygen concentrations require an Emergency Depressurization at ________________ in order to ensure the . . .

a. 5% H_2 **AND** 6% O_2
source of hydrogen production is removed.

b. 6% H_2 **AND** 5% O_2
source of hydrogen production is removed.

c. 5% H_2 **AND** 6% O_2
Reactor is at the lowest energy state possible.

d. 6% H_2 **AND** 5% O_2
Reactor is at the lowest energy state possible.

ANSWER: 5333

d. 6% H_2 **AND** 5% O_2
Reactor is at the lowest energy state possible.

Answer source: EOP flowchart 3A Table 7
INT008-06-13 p. 18

Distractors:

a,b. Combustible limits are 6% hydrogen and 5% oxygen. Depressurization does not remove the hydrogen source.
c. Concentrations are reversed.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
59	14023	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0080613	OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL

Related Objectives	
INT00806130011200	Given plant conditions and EOP flowchart 3A, PRIMARY CONTAINMENT CONTROL, state the reasons for the actions contained in the steps.

Related References	
PSTG	Plant Specific Technical Guideline
EOP 3A, PCCP	EOP 3A Primary Containment Control Procedure

Related Skills (K/A)	
295028.EK3.05	Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL TEMPERATURE: (CFR: 41.5 / 45.6) Reactor SCRAM (3.6/3.7)

QUESTION: 14023

When performing actions in EOP-3A "Primary Containment Control" for high drywell temperature, what is the reason for entering **AND** performing EOP-1A "RPV Control" concurrently without a specific EOP-1A entry condition being met?

Entering EOP-1A "RPV Control" and inserting a manual scram ensures . . .

a. the power produced by the reactor will be within the Primary Containment vent capability.

b. the RPV is at the lowest possible energy state before implementing more severe strategies.

c. the reactor is scrammed and shutdown by control rod insertion before RPV depressurization is initiated.

d. the main source of potential energy addition to the Primary Containment is removed before conditions warrant Emergency Depressurization.

ANSWER: 14023

c. the reactor is scrammed and shutdown by control rod insertion before RPV depressurization is initiated.

Answer source: INT008-06-13 p. 22

Distractors:

a. RCIC exhaust flowrate, factors for basis of PCPL-A.
b. The "lowest energy state" is an Emergency Depressurization basis. There is no more severe strategy than Emergency Depressurization.
d. It is not a "potential" energy addition and scramming the reactor does not remove the source of the energy addition.

Source: ***New***

Provide to Candidate: EOP-3A with cautions and entry conditions removed.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
60	14044	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0080613	OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL

Related Objectives	
INT00806130011200	Given plant conditions and EOP flowchart 3A, PRIMARY CONTAINMENT CONTROL, state the reasons for the actions contained in the steps.
INT00806130011000	Identify any EOP support procedures referenced in Flowchart 3A and apply any associated special operating instructions or cautions.

Related References	
PSTG	Plant Specific Technical Guideline

Related Skills (K/A)	
223001.K6.04	Knowledge of the effect that a loss or malfunction of the following will have on the PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES: (CFR: 41.7 / 45.7) Combustible gas mixing: Plant-Specific (2.8/2.8)

QUESTION: 14044

The plant was operating at rated power for several months when a LOCA with hydrogen generation occurred.

What affect will the inability to initiate drywell sprays have on combustible gas flammability and the primary containment?

The inability to initiate drywell sprays __________ the flammability of combustible gases and _________ the chance of containment damage from hydrogen deflagration.

a.	reduces	reduces
b.	reduces	increases
c.	increases	reduces
d.	increases	increases

ANSWER: 14044

d.	increases	increases

Answer source: INT008-06-13 p. 9

Distractors:

a., b, c Drywell spray reduces the flamability and potential for deflagration. Loss of Drywell sprays has the opposite affect.

Source: ***New***

Provide to Candidate: EOP-3A with cautions and entry conditions removed.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
61	5730	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0080617	OPS FLOWCHART 5A - SECONDARY CONTAINMENT AND RADIOACTIVITY RELEASE CONTROL

Related Objectives	
INT00806170010400	State the basis for the limits of the maximum safe operating values (MSO) as they apply to personnel protection and equipment operability.

Related References	
PSTG	Plant Specific Technical Guideline

Related Skills (K/A)	
2.3.1	Knowledge of 10 CFR: 20 and related facility radiation control requirements. (CFR: 41.12 / 43.4. 45.9 / 45.10) (2.6/3.0)

QUESTION: 5730

What is the basis for the value selected as the EOP Flowchart 5A Maximum Safe Operating (MSO) Radiation level?

It is based on personnel exposures exceeding the . . .

a. CNS TEDE dose limit of 3 Rem ***per quarter*** for a **<u>one hour</u>** stay time.

b. CNS TEDE dose limit of 4 Rem ***per year*** for a **<u>two hour</u>** stay time.

c. 10CFR20 planned special exposure limit of 3 rem for a **<u>one hour</u>** stay time.

d. 10CFR20 planned special exposure limit of 5 rem for a **<u>two hour</u>** stay time.

ANSWER: 5730

b. CNS TEDE dose limit of 4 Rem ***per year*** for a **<u>two hour</u>** stay time.

<u>Answer source:</u> INT008-06-17 pp. 10 & 11

Distractors:

c, d. MSO Radiation levels are not based on the 10CFR20 limits.
a. The quarterly limit no longer exists, but it was previously the basis for MSO radiation level, and herefore remains a plausible distractor.

2.3.1 Knowledge of 10 CFR: 20 and related facility radiation control requirements as it applies to 290001 Secondary Containment

<u>Source:</u> *Modified from 5730*

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
62	14019	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0080617	OPS FLOWCHART 5A - SECONDARY CONTAINMENT AND RADIOACTIVITY RELEASE CONTROL

Related Objectives	
INT00806170010700	Given plant conditions and EOP flowchart 5A, SECONDARY CONTAINMENT CONTROL and RADIOACTIVITY RELEASE CONTROL, state the reasons for the actions contained in the steps.

Related References	
PSTG	Plant Specific Technical Guideline

Related Skills (K/A)	
295038.EK3.04	Knowledge of the reasons for the following responses as they apply to HIGH OFF-SITE RELEASE RATE: (CFR: 41.5 / 45.6) Emergency depressurization (3.6/3.9)

QUESTION: 14019

What is the basis for performing an Emergency Depressurization during the execution of EOP 5A, **RADIOACTIVITY RELEASE CONTROL**?

Performing an Emergency Depressurization ensures the . . .

a. availability of equipment in the turbine building that may be necessary to mitigate the event is not challenged.

b. energy level of the radiation and the atomospheric dispersion factors fall within the bounds of the accident analysis.

c. isotopic mixture of radioactive materials deposited off-site will be within the bounds of the accident analysis.

d. lowest possible driving head and flow of primary systems that are unisolated and discharging outside of containment.

ANSWER: 14019

d. lowest possible driving head and flow of primary systems that are unisolated and discharging outside of containment.

Answer source: INT008-06-17 p. 14

Distractors:

a. availability of turbine building equipment is not an EOP consideration, but sounds like the reactor building ED reason.
b. This is a reason to use the DOSE program for the projections vice the ODAM calculations, but is not the reason for the ED.
c. This is not the basis for the ED.

Source: ***New***

Provide to Candidate: EOP 5A with entry conditions and cautions removed.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
63	14021	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0080617	OPS FLOWCHART 5A - SECONDARY CONTAINMENT AND RADIOACTIVITY RELEASE CONTROL

Related Objectives	
INT00806170010700	Given plant conditions and EOP flowchart 5A, SECONDARY CONTAINMENT CONTROL and RADIOACTIVITY RELEASE CONTROL, state the reasons for the actions contained in the steps.

Related References	
PSTG	Plant Specific Technical Guideline

Related Skills (K/A)	
295036.EK3.02	Knowledge of the reasons for the following responses as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL: (CFR: 41.5 / 45.6) Reactor SCRAM. (2.8/2.8)

QUESTION: 14021

Why is a reactor scram required before exceeding a Maximum Safe Operating Water Level if a primary system is discharging into the Reactor building (EOP-5A step SC-12)?

Scramming the Reactor . . .

a. provides mitigating action such that the condition does not pose an immediate threat to the health and safety of the public.

b. promptly reduces energy to decay heat levels and reduces the likelihood of requiring rapid depressurization of the RPV.

c. promptly reduces to decay heat levels the energy discharged into primary containment and reduces the driving head and flow of primary systems that are unisolated.

d. promptly places the primary system in its lowest possible energy state and reduces the driving head and flow of primary systems that are unisolated and discharging into the secondary containment.

ANSWER: 14021

b. promptly reduces energy to decay heat levels and reduces the likelihood of requiring rapid depressurization of the RPV.

Answer source: INT008-06-17 p. 10, section 10

Distractors:

a. Not a bases for scramming.
c. Primary Containment Control bases.
d. Bases for emergency depressurization.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
64	16483	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0080617	OPS FLOWCHART 5A - SECONDARY CONTAINMENT AND RADIOACTIVITY RELEASE CONTROL

Related Objectives	
INT00806170010600	Given plant conditions and EOP flowchart 5A, SECONDARY CONTAINMENT CONTROL and RADIOACTIVITY RELEASE CONTROL, determine required actions.

Related References	
EOP 5A, SCCP	EOP 5A, Secondary Containment Control
EOP/SAG PSTG	Plant Specific Technical Guideline

Related Skills (K/A)	
2.4.31	Knowledge of annunciators alarms and indications / and use of the response instructions. (CFR: 41.10 / 45.3) (3.3/3.4)

QUESTION: 16483

The plant is at 90% when a RCIC steam line break occurs. Initial event conditions were:

- ➢ RCIC cannot be isolated
- ➢ RCIC temperatures in the NE Quad are 214°F and rising
- ➢ Reactor Building ventilation exhaust radiation levels are 25 mr/hr
- ➢ RCIC Room radiation levels are 300 mr/hr and rising
- ➢ A reactor scram is inserted and all control rods fully insert

Five (5) minutes later, annunciator 9-4-1/E-4 RX BLDG VENT HI HI RAD clears.

What is the required EOP response to this annunciator clearing and what is the bases for the response?

a. Restart Reactor Bldg. HVAC to ensure all radioactive discharges are elevated.

b. Restart Reactor Bldg. HVAC to help return secondary containment parameters to normal.

c. Do **NOT** restart Reactor Bldg. HVAC because EOP 1A requires a group 6 isolation.

d. Do **NOT** restart Reactor Bldg. HVAC until RP ensures normal radiation levels to minimize the spread of contamination.

ANSWER: 16483

b. Restart Reactor Bldg. HVAC to help return secondary containment parameters to normal.

<u>Answer source:</u> INT008-06-17 p. 7, section "b"

Distractors:

a. This is <u>not</u> a bases for restarting Reactor Bldg. HVAC
c. The Isolation would only occur on low level (3) and if so, it should be bypassed.
d. Per 2.1.22 if a Group 6 Isol had occurred, it should not be reset until Chem. and HP have ensured normal rad levels, but EOPs take precedence.

<u>Source:</u> *Direct from bank*

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
65	19068	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0080617	OPS FLOWCHART 5A - SECONDARY CONTAINMENT AND RADIOACTIVITY RELEASE CONTROL

Related Objectives	
INT00806170010700	Given plant conditions and EOP flowchart 5A, SECONDARY CONTAINMENT CONTROL and RADIOACTIVITY RELEASE CONTROL, state the reasons for the actions contained in the steps.

Related References	
EOP/SAG PSTG	Plant Specific Technical Guideline

Related Skills (K/A)	
2.3.11	Ability to control radiation releases. (CFR: 45.9 / 45.10) (2.7/3.2)

QUESTION: 19068

What is the basis for restarting building ventilation in the Turbine Building when executing EOP-5A, RADIOACTIVITY RELEASE CONTROL?

Operation of Turbine Building ventilation . . .

a. maintains equipment availability **AND** assures that radioactivity releases pass through a monitored release point.

b. preserves personnel accessibility **AND** assures that radioactivity releases pass through a monitored release point.

c. maintains equipment availability **AND** assures a minimum amount of radioactivity plates out on turbine building surfaces.

d. preserves personnel accessibility **AND** assures a minimum amount of radioactivity plates out on turbine building surfaces.

ANSWER: 19068

b. preserves personnel accessibility **AND** assures that radioactivity releases pass through a monitored release point.

Continued personnel access to the turbine building, radwaste and augmented radwaste may be essential for responding to emergencies. These structures are not air tight and radioactivity release inside them would not only limit personnel access, but would eventually lead to an unmonitored ground level release. Operation of ventilation in these structures preserves accessibility, and assures that radioactivity is discharged through an elevated, monitored release point.

Answer source: INT008-06-17 p. 13, section B.1

Distractors:

a. The purpose of restarting Turbine Building ventilation is not to preserve equipment availability.
c. The purpose of restarting Turbine Building ventilation is not to preserve equipment availability nor to minimize deposition of radioactivity in the building.
d. The purpose of restarting Turbine Building ventilation is not to minimize deposition of radioactivity in the building.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
66	14046	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
INT0080618 EOP AND SAG GRAPHS

Related Objectives	
INT00806180010100	Using the graphs provided in the EOP and SAG Graphs Flowchart, determine how the shape of each curve or family of curves was determined.

Related References	
PSTG	Plant Specific Technical Guideline
INT0080618	EOP and SAG Graphs and Cautions

Related Skills (K/A)	
295030.EK1.03	Knowledge of the operational implications of the following concepts as they apply to LOW SUPPRESSION POOL WATER LEVEL: (CFR: 41.8 to 41.10) Heat capacity. (3.8/4.1*)

QUESTION: 14046

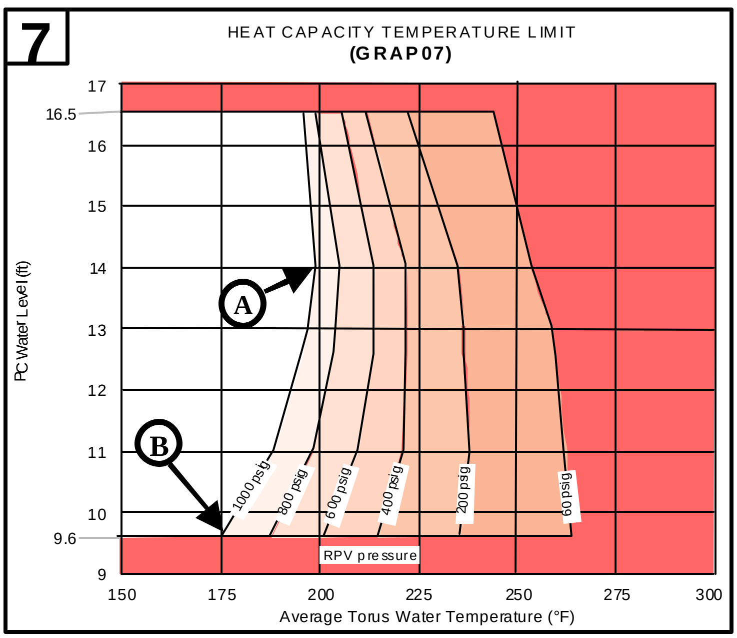
What is the bases for the shape of the Heat Capacity Temperature Limit (GRAPH 07) from point "A" to point "B"?

a. As torus level lowers, there is less backpressure on the SRV tailpipes. With lower SRV backpressure, a lower initial temperature will exceed analyzed blowdown rates during a LOCA.

b. As torus level lowers, there is less backpressure on the downcomers. With lower downcomer backpressure, a lower initial temperature will exceed peak design containment pressure during a LOCA blowdown.

c. Torus airspace volume increase has a greater effect on the limit than torus heat sink mass decrease. As torus airspace volume rises, a lower temperature is needed to heat and pressurize the airspace to PCPL-A.

d. Torus heat sink mass decrease has a greater effect on the limit than torus airspace volume increase. As torus level decreases, there is less mass available to absorb the blowdown energy from the RPV, requiring a more restrictive limit.



ANSWER: 14046

d. Torus heat sink mass decrease has a greater effect on the limit than torus airspace volume increase. As torus level decreases, there is less mass available to absorb the blowdown energy from the RPV, requiring a more restrictive limit.

Answer source: INT008-06-18 pp. 12 & 13

Distractors:

a. HCTL is based on not exceeding PCPL-A and has nothing to do with SRV backpressure.
b. HCTL is based on not exceeding PCPL-A and has nothing to do with downcomer backpressure.
d. Torus airspace volume increase as a lesser effect and acts to make a less restrictive limit rather than a more resrictive limit.

Source: ***New***

Provide to Candidate: EOP Graphs.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
67	5764	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0320101	CNS Administrative Procedures Volume 0, Administrative Procedures (Formal Classroom/Pre-OJT Training)

Related Objectives	
INT032010100G010I	Discuss the following as described in Administrative Procedure 0.26, Surveillance Program: Surveillance test authorization
INT032010100G010J	Discuss the following as described in Administrative Procedure 0.26, Surveillance Program: Surveillance test performance
INT032010100G010N	Discuss the following as described in Administrative Procedure 0.26, Surveillance Program: Surveillance Test Scheduling

Related References	
0.26	Surveillance Program

Related Skills (K/A)	
2.2.12	Knowledge of surveillance procedures. (CFR: 41.10 / 45.13) (3.0/3.4)

QUESTION: 5764

A Standby Gas Treatment operability surveillance test is due to be performed this shift.

Who is responsible for the coordination and authorization of performance of this test?

a. Shift Supervisor

b. Operations Manager

c. Operations Supervisor

d. Surveillance Coordinator

ANSWER: 5764

a. Shift Supervisor

<u>Answer source:</u> 0.26 p. 4, section 5

Distractors:

b. No direct responsibility for 0.26 actions.
c. Only responsible for arranging working load to ensure surveillance completed on time.
d. Only schedules surveillance.

2.2.12 Knowledge of surveillance procedures. (CFR: 41.10 / 45.13) as it applies to: 261000 SGTS

<u>Source:</u> ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
68	12215	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0320102	CNS Administrative Procedures Site Services Procedures (Formal Classroom/Pre-OJT Training)

Related Objectives	
INT032010200C030A	Discuss the requirement associated with the following items: Visitor acces and departure
INT032010200C010B	Discuss the following as described in Administrative Procedure 1.15, Visitor/Tour Station Access: Vital Area access criteria

Related References	
1.15	Visitor/Tour Station Access

Related Skills (K/A)	
2.3.1	Knowledge of 10 CFR: 20 and related facility radiation control requirements. (CFR: 41.12 / 43.4. 45.9 / 45.10) (2.6/3.0)

QUESTION: 12215

Four vendor representatives were provided an escorted tour to perform a walkdown of the RHR system to assist in correcting an emergent problem. TLDs were not issued to the vendors. Following the walkdown, the following dose was received by each of the vendors on their DRDs:

➢ Rick Cox	106 mrem
➢ Ira Fox	98 mrem
➢ Marvin Estes	51 mrem
➢ Robert Smith	101 mrem

Which of the following are required?

An Exposure History Worksheet must be completed for . . .

a. **all** the visitors prior to departure.

b. **only** Rick Cox before departure.

c. **only** Rick Cox and Robert Smith before departure.

d. **only** Rick Cox, Robert Smith and Ira Fox before departure.

ANSWER: 12215

c. **only** Rick Cox and Robert Smith before departure.

If the dose received is > 100 mrem (1.0 mSv) by DRD, or equivalent, and no TLD was issued, Radiological Protection shall be contacted and a CNS-RP-10, Exposure History Worksheet, shall be completed before allowing the visitor to depart. Both these individuals received greater than 100 mrem and since TLDs were not issued, a CNS-RP-10, Exposure History Worksheet, shall be completed before allowing they depart.

Answer source: 1.15 page 4, step 4.6

Distractors:

a. Ira Fox and Marvin Estes received less than 100 mrem and are not required to complete CNS-RP-10, Exposure History Worksheet.
b. Robert Smith is also required to complete CNS-RP-10, Exposure History Worksheet prior to departure.
d. Ira Fox received less than 100 mrem and is not required to complete CNS-RP-10, Exposure History Worksheet.

Source: ***Direct from Bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
69	12209	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0320103	CNS Administrative Procedures Conduct of Operations and General Alarm Procedures (Formal Classroom/Pre-OJT Training)

Related Objectives	
INT032010300E010B	Discuss the following as described in Alarm Procedure 2.3.1, General Alarm Procedure: Alarm acknowledgement

Related References	
2.3.1	Procedure 2.3.1, General Alarm Procedure
OI-07	Operations Management Expectations

Related Skills (K/A)	
2.4.31	Knowledge of annunciators alarms and indications / and use of the response instructions. (CFR: 41.10 / 45.3) (3.3/3.4)

QUESTION: 12209

The plant is operating at 100% power. Surveillance Procedure 6.2PCIS.302, MAIN STEAM LINE HIGH TEMPERATURE CHANNEL FUNCTIONAL TEST 9 (DIV 2) has just commenced.

STEAM TUNNEL HIGH TEMP CHANNEL A (9-5-1/E-1) then alarms (the first annunciation of this alarm for the shift).

What action(s) is/are required?

Acknowledge the annunciator . . .

a. **only**.

b. **AND** announce "expected annunciator".

c. **AND** announce "STEAM TUNNEL HIGH TEMP CHANNEL A" **only**.

d. **AND** announce "STEAM TUNNEL HIGH TEMP CHANNEL A" **AND** pull the associated alarm card.

ANSWER: 12209

d. **AND** announce "STEAM TUNNEL HIGH TEMP CHANNEL A" **AND** pull the associated alarm card.

This annunciator is not associated with the surveillance and therefore the alarm should be acknowledged, announced and the alarm card pulled.

Answer source: 2.3.1 p. 3, steps 4.12, 4.13 & 4.14

Distractors:

a. This annunciator is not expected for this surveillance.
b. This annunciator is not expected for this surveillance.
c. The alarm card must be pulled for this annunciator.

Source: ***Direct from Bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
70	13673	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0320104	CNS Administrative Procedures General Operating Procedures (Startup and Shutdown) Procedures (Formal Classroom/Pre-OJT Training)

Related Objectives	
INT032010400G0300	State from memory the " Mitigating Task Scram Actions" associated with Procedure 2.1.5, Reactor Scram.

Related References	
2.1.5	Procedure 2.1.5, Reactor Scram

Related Skills (K/A)	
295006.AA2.05	Ability to determine and/or interpret the following as they apply to SCRAM: (CFR: 41.10 / 43.5 / 45.13) Whether a reactor SCRAM has occurred (4.6*/4.6*)

QUESTION: 13673

The plant was operating at full power when a reactor scram occurred.

Per 2.1.5, "Reactor Scram," which methods are to be utilized to determine that all control rods have been fully inserted into the core?

a. REFUEL MODE SELECT PERMISSIVE light ON **ONLY**.

b. The REFUEL MODE SELECT PERMISSIVE light is ON **OR** ***ALL*** green FULL-IN lights on the full core display are ON.

c. All green FULL-IN lights on the full core display are ON **AND** ***ALL*** rod positions indicating 00 on the PMIS RPIS display screen.

d. The REFUEL MODE SELECT PERMISSIVE light is ON **OR** ***ALL*** rod positions indicating 00 on the PMIS RPIS display screen.

ANSWER: 13673

b. The REFUEL MODE SELECT PERMISSIVE light is ON **OR** ***ALL*** green FULL-IN lights on the full core display are ON.

Answer source: 2.1.5 p. 5, Attachment 1, step 1.5

Distractors:

a. Utilizing the green full in lights on the full core display is also allowed.
c. The use of the RPIS PMIS display is not directed by procedure 2.1.5.
d. The use of the RPIS PMIS display is not directed by procedure 2.1.5

Source: ***Direct from Bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
71	16796	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
INT0320126 CNS Abnormal Procedures (RO) Cooling Water

Related Objectives	
INT0320126Q0Q0100	Given plant condition(s), and the applicable Abnormal/Emergency Procedure, determine the correct subsequent actions required to mitigate the event(s).

Related References	
2.4SDC	RHR Loss of Shutdown Cooling

Related Skills (K/A)	
295021.AA2.07	Ability to determine and/or interpret the following as they apply to LOSS OF SHUTDOWN COOLING: (CFR: 41.10 / 43.5 / 45.13) Reactor recirculation flow (2.9/3.1)

QUESTION: 16796

With the plant shutdown in Mode 4 and RHR loop "B" operating in Shutdown Cooling, the following conditions exist:

- ➢ Reactor pressure is 0 psig
- ➢ Recirc suction temperature is 170°F
- ➢ Reactor water level is 36" (NR)
- ➢ RHR pumps "A" and "C" have both motors disconnected from their pumps

Subsequently, "B" RHR Loop develops a leak, requiring the Control Room operators to remove RHR loop "B" from service.

What action is required, and for what reason?

a. RPV water level must be raised to > 48" to aid in natural circulation flow.

b. RPV water level must be raised to > 48" in order to maximize Reactor coolant contact with RPV metal for enhanced heat transfer to the Drywell atmosphere.

c. A Reactor Recirculation pump must be started to reduce the possibility of excessive thermal stresses on the CRD stub tubes.

d. A Reactor Recirculation pump must be started in order to reduce the possibility of thermal binding of the RHR-MO-25A/B valves caused by the expected coolant heatup.

ANSWER: 16796

a. RPV water level must be raised to > 48" to aid in natural circulation flow.

Answer source: 2.4SDC p. 10, Attachment 2, step 1.1

Distractors:

b. Water level is raised to enhance natural ciruclation flow, not heat transfer.
c. A Recirculation pump is started, but not to prevent thermal stresses on the stub tubes. Starting a recirc pump causes thermal stress on the stub tubes.
d. A Recirculation pump is started, but not to prevent thermal binding of the gate valves. These valves are cycled if closed during a cooldown.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
72	14426	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0320132	CNS Abnormal Procedures (RO) Off Gas/Vaccuum
COR0011402	OPS MAIN TURBINE

Related Objectives	
COR0011402001060D	Given a specific Main Turbine and Auxilary systems malfunction, determine the effect on any of the following: Condenser vacuum
INT0320132L0L0100	Given plant condition(s), and the applicable Abnormal/Emergency Procedure, determine the correct subsequent actions required to mitigate the event(s).
COR0011402001140E	Given a Main Turbine and Auxiliaries component manipulation, predict and explain the changes in the following parameters: Steam seal pressure

Related References	
2.4VAC	Loss of Condenser Vacuum
2.2.75	Procedure 2.2.75, Steam Sealing System

Related Skills (K/A)	
295002.AK2.11	Knowledge of the interrelations between LOSS OF MAIN CONDENSER VACUUM and the following: (CFR: 41.7 / 45.8) Seal steam: Plant-Specific. (2.6/2.7)

QUESTION: 14426

The plant is at 30% power with the following conditions:

- ➢ Main Condenser vacuum is currently 27"Hg and degrading
- ➢ SJAE air flow (AR-FR-47) has shown a large increase over the last several minutes
- ➢ Gland Steam supply pressure indicator (MS-PI-83) indicates 0 psig

What action will correct the problem?

a. Throttle ***Open*** MS-MO-BMV3, Steam Supply Bypass Valve.

b. Throttle ***Closed*** MS-MO-BMV3, Steam Supply Bypass Valve.

c. Throttle ***Open*** MS-MO-BMV4, Steam Unloader Bypass Valve.

d. Throttle ***Closed*** MS-MO-BMV4, Steam Unloader Bypass Valve.

ANSWER: 14426

a. Throttle ***Open*** MS-MO-BMV3, Steam Supply Bypass Valve.

Regardless of the valve alignment or supply, the correct action to take at this power level is to open the steam bypass from main steam to put pressure on the seals.

Answer source: 2.2.75 p. 3, step 4.19.2

Distractors:

b. Taking action to close BMV3 could only make a low pressure situation worse, and BMV3 is probably already closed to begin with.
c. BMV4 would be opened if pressure was too high and steam was leaking from the seals.
d. Throttling closed BMV4 could make a high pressure situation worse, if the steam unloaders did not have enough capacity to maintain pressure.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
73	4214	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0320134	OPS CNS Abnormal Procedures (RO) - Fire

Related Objectives	
INT0320134H0H0100	Given plant condition(s), and the applicable Abnormal/Emergency Procedure, determine the correct subsequent actions required to mitigate the event(s).
INT0320134D0D0100	Given plant condition(s), determine from memory the appropriate Abnormal/Emergency Procedure(s) to be utilized to mitigate the event(s).

Related References	
5.4.FIRE-SD	FIRE INDUCED SHUTDOWN FROM OUTSIDE CONTROL ROOM

Related Skills (K/A)	
2.4.34	Knowledge of RO tasks performed outside the main control room during emergency operations including system geography and system implications. (CFR: 43.5 / 45.13) (3.8/3.6)

QUESTION: 4214

A shutdown from outside the Control Room is in progress per 5.4FIRE-S/D.

How is Service Water flow established/controlled through RHR HX?

a. The Reactor Building operator manually operates SW-MO-89B from the handwheel.

b. The Reactor Building operator operates the open contacts at MCC-Y for SW-MO-89B.

c. The ASD Operator controls flow by the control switch for SW-MO-89B on the RHR panel in the ASD Room.

d. The Reactor Building operator fully opens SW-MO-89B at MCC-Y and the Control Building operator throttles on the SWBP discharge valve.

ANSWER: 4214

b. The Reactor Building operator operates the open contacts at MCC-Y for SW-MO-89B.

5.4FIRE-SD directs the operator to ensure breaker is closed and to rotate silver screw under switch which defeats mechanical interlock and open cubicle door. Then to remove control power fuse to prevent spurious operation, and to open valve by pressing button on LOWER contactor for time indicated for valve on the procedure attachment.

Answer source: 5.4FIRE-S/D p.22, step 1.1.6

Distractors:

a. 5.4.FIRE-SD directs the use of the open and closed contacts at MCC-Y for SW-MO-89B.
c. 5.4.FIRE-SD directs the use of the open and closed contacts at MCC-Y for SW-MO-89B.
d. 5.4.FIRE-SD directs the use of the open and closed contacts at MCC-Y for SW-MO-89B.

Source: ***Direct from Bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
74	5127	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
SKL0080101 WATCHSTANDING PRINCIPLES

Related Objectives	
SKL00801010011500	State what should be done if an out of limits reading is recorded on the logs or if an error is made recording a reading.

Related References	
1.9	Control and Retention of Records

Related Skills (K/A)	
2.1.18	Ability to make accurate / clear and concise logs / records / status boards / and reports. (CFR: 45.12 / 45.13) (2.9/3.0)

QUESTION: 5127

The RO realizes that an incorrect river level was transcribed by the Station Operator into 6.LOG.601, DAILY SURVEILLANCE LOG - MODES 1, 2, AND 3."

How is the the wrong river level corrected?

a. Correct number should be entered **AND** dated.

b. Circle the number, **AND** write in the correct number with an explanation.

c. Circle, initial **AND** date the old number, **AND** write in the correct number.

d. Draw one line through the incorrect old number, initial **AND** date, **AND** write in the correct number.

ANSWER: 5127

d. Draw one line through the incorrect old number initial **AND** date, **AND** write in the correct number.

Procedure 1.9 requires a single line to be drawn through the information to be corrected such that the information crossed out must remain legible. The person making the correction must initial and date the correction. Additional information may be added.

Distractors:

a. Required to draw a single line through correction, intial & date and write in the correct entry.
b. Action for a reading out of limit. Required to draw a single line through correction, intial & date and write in the correct entry.
c. Action for a reading out of limit. Required to draw a single line through correction, intial & date and write in the correct entry.

Source: ***Direct from Bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
75	768	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0012102 Refueling

Related Objectives	
COR0012102001030C	Given a Reactor Refueling and Servicing Equipment manipulation, predict and explain the changes in the following parameters: Core reactivity level

Related References	
10.25	Refueling - Core Unload, Reload, and Shuffle

Related Skills (K/A)	
295023.AK1.02	Knowledge of the operational implications of the following concepts as they apply to REFUELING ACCIDENTS: (CFR: 41.8 to 41.10) Shutdown margin. (3.2/3.6)

QUESTION: 768

A core reload is in progress.

- ➢ SRM "B" count rate is 50 cps
- ➢ Fuel Pool water temperature is 95°F.
- ➢ A fuel bundle is being lowered into the core and is just passing through the top guide.
- ➢ SRM "B" count rate rises to 500 cps.

Per procedure 10.25 (Refueling - Core Unload, Reload and Shuffle), what action is required to be performed?

a. Immediately terminate fuel loading.

b. Continue to insert the bundle normally. If the SRM reaches 5 "doubles," terminate fuel loading.

c. Insert the bundle half way into the core, then stop and monitor the count rate. Remove the fuel bundle from the core if the SRM reaches 5 "doubles."

d. Slowly lower the bundle into the core by moving in six (6) inch increments, stopping to monitor SRM count rates at each increment. Terminate fuel loading if the SRM reaches 5 "doubles."

ANSWER: 768

a. Immediately terminate fuel loading.

Answer source: 10.25 p. 12, step 8.1.18

Distractors:

b. Procedure 10.25 requires fuel handling be terminated if unexpected rise on SRMs is noted. The maximum count rate expected during a fuel shuffle is 100 cps.
c. Procedure 10.25 requires fuel handling be terminated if unexpected rise on SRMs is noted. The maximum count rate expected during a fuel shuffle is 100 cps.
d. Procedure 10.25 requires fuel handling be terminated if unexpected rise on SRMs is noted. The maximum count rate expected during a fuel shuffle is 100 cps.

55.43 section(s): (7)

SRO Justification: SRO licensed personnel supervise refueling activities and direct the actions of the refuel bridge operator.

Source: ***Modified***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
76	5101	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
COR0012102	Refueling
INT0320117	CNS Administrative Procedures Volume Ten, Nuclear Performance Procedures (Formal Classroom/Pre-OJT Training)

Related Objectives	
COR0012102001030C	Given a Reactor Refueling and Servicing Equipment manipulation, predict and explain the changes in the following parameters: Core reactivity level

Related References	
10.23	New Fuel Inspection, Channeling, and Control Blade Inspection
10.25	Refueling - Core Unload, Reload, and Shuffle
2.2.31	Procedure 2.2.31, Fuel Handling - Refueling Platform

Related Skills (K/A)	
2.2.27	Knowledge of the refueling process. (CFR: 43.7 / 45.13) (2.6/3.5)

QUESTION: 5101

When handling NEW fuel in the Fuel Pool Area, how many bundles are allowed outside normal storage area **OR** shipping container at a time?

a. one (1)

b. two (2)

c. three (3)

d. four (4)

ANSWER: 5101

c. three (3)

Answer source: 10.23 p. 4, step 2.25

Distractors:

a. 3 bundles are allowed.
b. 3 bundles are allowed.
d. 3 bundles are allowed.

55.43 section(s): (7)

SRO Justification: SRO responsible for refueling and the Special Nuclear Material Executor must be an SRO licensed individual.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
77	5468	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0020902 Digital Electro-Hydraulic Control

Related Objectives	
COR0020902001070B	Given a specific DEH Control system malfunction, determine the effect on any of the following: Reactor pressure

Related References	
2.2.77.1	Procedure 2.2.77.1, DEH Control System
3.4.10	Reactor steam dome pressure

Related Skills (K/A)	
295007.AA2.02	Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE: (CFR: 41.10 / 43.5 / 45.13) Reactor power (4.1*/4.1*)

QUESTION: 5468

Reactor power is 97%. After raising pressure set on DEH, the setpoint continues to rise.

What affect (if any) will the rising pressure setpoint have on reactor power?

Reactor power will . . .

a. remain constant. Reactor pressure will remain within the steady-state Technical Specification limit.

b. lower until the max setpoint is reached, then return to 97%. Reactor pressure will exceed the steady-state Technical Specification limit.

c. rise until the max setpoint is reached, then return to 97%. Reactor pressure will remain within the steady-state Technical Specification limit.

d. rise until the reactor scrams on high reactor pressure or high reactor power. Reactor pressure will exceed the steady-state Technical Specification limit.

ANSWER: 5468

d. rise until the reactor scrams on high reactor pressure or reactor power. Reactor pressure will exceed the steady-state Technical Specification limit.

Explanation: DEH Max pressure setpoint is 1200 psig.

Answer source: Tech Spec LCO 3.4.10
2.4DEH, p 5, Attachment 1
COR002-09-02, p. 28, section 2.b

Distractors:

a. Reactor pressure will rise. The Tech Spec limit of 1020 psig will be exceeded.
b. Reactor pressure will rise.
c. The reactor will scram. The Tech Spec limit of 1020 psig will be exceeded.

55.43 section(s): (2)

SRO Justification: SRO assess plant conditions during an abnormal event and determine if the plant is responding per design. This knowledge would be used ot select the appropriate procedure to mitigate the event, as required. Knowledge of the Tech Spec limit for Steam Dome pressure and range of DEH controls, reactor power response from memory.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
78	14052	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0020302 CONTAINMENT

Related Objectives	
COR0020302001160D	Given a Containment/PCIS component manipulation, predict and explain the changes in the following: Drywell to suppression chamber D/P
COR0020302001140F	Briefly describe the following concepts as they apply to the Primary containment: Drywell to Torus Differential Pressure.

Related References	
2.2.60	Procedure 2.2.60, Primary Containment Cooling And Nitrogen Inerting System

Related Skills (K/A)	
2.1.7	Ability to evaluate plant performance and make operational judgments based on operating characteristics / reactor behavior / and instrument interpretation. (CFR: 43.5 / 45.12 / 45.13) (3.7/4.4)

QUESTION: 14052

The plant was operating at power when drywell pressure and temperature began rising. The crew vented ***only*** the torus using the Standby Gas Treatment system. 30 minutes later, torus pressure is 0.28 psig and drywell pressure is 0.3 psig.

Is this the expected containment response and why/why not?

a. Yes. The torus air space and drywell atmosphere are connected via the inerting piping, bypassing the downcomers.

b. Yes. The torus air space and drywell atmosphere are connected via the vent piping, irrespective of which air space SGT is aligned to.

c. No. A differential pressure of approximately 1.3 psid is required to overcome backpressure on the vent header downcomers if the pressure suppression function is intact.

d. No. A differential pressure of approximately 1.3 psid is required to establish flow between the the torus air space and drywell atmosphere if the primary containment is intact.

ANSWER: 14052

a. Yes. The torus air space and drywell atmosphere are connected via the inerting piping, bypassing the downcomers.

Answer source: 2.2.60 valve lineup
COR002-03-02 Figure 9a

Distractors:

b. SGT does not cross-connect the air spaces.
c. This is the expected containment response.
d. This is the expected containment response.

2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics / reactor behavior / and instrument interpretation.(CFR: 43.5 / 45.12 / 45.13) as it applies to: 295010 High Drywell Pressure / 5

55.43 section(s): (5)

<u>SRO Justification:</u> SRO persons assess plant conditions and determine compliance with Technical Specification and action required for non-compliance. Technical Specification assessment of containment integrity.

<u>Source:</u> ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
79	32	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0020302 CONTAINMENT

Related Objectives	
COR0020302001160B	Given a Containment/PCIS component manipulation, predict and explain the changes in the following: Drywell pressure

Related References	
5.2REC	LOSS OF REC
2.4PC	PRIMARY CONTAINMENT CONTROL

Related Skills (K/A)	
295020.AK3.03	Knowledge of the reasons for the following responses as they apply to INADVERTENT CONTAINMENT ISOLATION: (CFR: 41.5 / 45.6) Drywell/containment temperature response (3.2/3.2)

QUESTION: 32

The plant has been operating at rated power for the last 6 months when the MSIVs suddenly close.

If all systems respond as designed and no operator action is taken, how does Drywell pressure respond over the next 2 hours and why?

Drywell pressure will . . .

a. lower slowly as the drywell heat load is significantly reduced, lowering drywell average temperature.

b. remain relatively constant as the heat load in the drywell remains relatively constant at a given reactor pressure.

c. rise slowly as the heat added to the torus migrates to the drywell, raising average drywell temperature.

d. rise rapidly due to loss of drywell cooling and heat addition to the torus, raising average drywell temperature.

ANSWER: 32

a. lower slowly as the drywell heat load is significantly reduced, lowering drywell average temperature.

With one recirculation pump tripped (one is powered from the Normal Transformer that trips when the Main Generator trips), and the other recirculation pump at minimum speed (< 20% feed flow limits recirc speed to 22%), the heat load on drywell cooling is significantly decreased. With RWCU isolated, a very large heat load is removed from the REC system. Drywell pressure was stable with the previous drywell heat load. The combination of these effects results in lower drywell temperature and pressure after the scram. Torus water temperature does not directly affect drywell airspace temperature.

Answer source: 2.4PC p. 3, step 5.4
5.2REC p. 2, step 4.3.3.2
5.2REC p. 5, step 5.8.2

Distractors:

b. RWCU isolates, one reactor recirc pump trips and the other reactor recirc pump goes to minimum speed, significantly reducing the heat load inside the primary containment.
c. RWCU isolates, one reactor recirc pump trips and the other reactor recirc pump goes to minimum speed, significantly reducing the heat load inside the primary containment.
d. RWCU isolates, one reactor recirc pump trips and the other reactor recirc pump goes to minimum speed, significantly reducing the heat load inside the primary containment. Drywell cooling is not lost on a normal group 1 isolation and scram.

USAR XIV-5-6

55.43 section(s): (5)

SRO Justification: SRO personnel assess plant response to transients to ensure the plant response is as designed, and based on that assessment, direct plant response to the event.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
80	36	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0020302 CONTAINMENT

Related Objectives	
COR0020302001120J	Describe the Containment design features and/or interlocks that provide for the following: Secondary containment over pressure protection

Related References	
V.XII.2.3	
XII 2.3.5.2	USAR and Appendix C (loepxviii1)

Related Skills (K/A)	
295035.EK3.01	Knowledge of the reasons for the following responses as they apply to SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE: (CFR: 41.5 / 45.6) Blow-out panel operation: Plant-Specific. (2.8/3.1)

QUESTION: 36

What is the reason for designing the Reactor Building metal siding to blow off if excessive Reactor Building to atmosphere differential pressure exists?

a. To limit the stresses on the Reactor Building during a tornado.

b. To ensure steam leaks in the steam tunnel do not migrate into the Secondary Containment quads, causing safety-related equipment to become non-functional.

c. To allow rapid restoration of Secondary Containment integrity after a large steam leak to limit releases to small fractions of 10 CFR 100 limits.

d. To ensure that any steam leakage into the Secondary Containment does not migrate to the Control Building, ensuring radioactive dose to the operators remains within analyzed limits.

ANSWER: 36

a. To limit the stresses on the Reactor Building during a tornado.

Answer source: USAR p. XII-2-14, section 2.3.3.2.4
USAR p. C-2-10, section 2.5.4

Distractors:

b. The reason for the Rx Bldg blow out panels is to limit the stresses on the Reactor Building during a tornado.
c. The reason for the Rx Bldg blow out panels is to limit the stresses on the Reactor Building during a tornado.
d. The reason for the Rx Bldg blow out panels is to limit the stresses on the Reactor Building during a tornado.

55.43 section(s): (2)

SRO Justification: SRO knowledge of basic plant design is required to assess plant response to abnormal and emergency plant events and to predict impacts of events to allow direction of appropriate actions.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
81	44	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
SKL0124223	OPS RESIDUAL HEAT REMOVAL
OTH0020414	Systems Precautions and Limitations Examination (SRO)

Related Objectives	
SKL012422300A0200	Explain the Residual Heat Removal system limitations and precautions as stated in the SOP 2.2.69, SOP 2.2.69.1, SOP 2.2.69.2 and SOP 2.2.69.3.
SKL012422300B0600	Comply with all related Residual Heat Removal system limits and precautions.

Related References	
2.2.69.2	Procedure 2.2.69.2, RHR System Shutdown Operations

Related Skills (K/A)	
205000.A2.12	Ability to (a) predict the impacts of the following on the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE); and (b) based on those predictions, use procedures to correct, control, or mitigate...: (CFR 41.5 / 45.6) Inadequate system flow (2.9/3.0)

QUESTION: 44

The plant is operating during a refueling outage with the following conditions:

- ➢ 180 fuel bundles have been removed from the reactor core and placed in the fuel pool
- ➢ "B" RHR loop is in Shutdown Cooling
- ➢ RHR-MO-27B, OUTBD INJECTION VLV is inadvertently closed for several seconds
- ➢ "B" RHR loop flow now indicates 2800 gpm

What actions are required?

a. Raise RPV water level to > +48" (NR) **only**.

b. Raise shutdown cooling flow to > 5000 gpm but ≤ 7000 gpm.

c. Shut "B" RHR loop minimum flow valve **AND** raise shutdown cooling flow to > 7000 gpm but ≤ 8400 gpm.

d. Shut "B" RHR loop minimum flow valve **AND** raise RPV water level to > +48" (NR) **AND** raise shutdown cooling flow to > 5000 gpm but ≤ 8400 gpm.

ANSWER: 44

b. Raise shutdown cooling flow to > 5000 gpm but ≤ 7000 gpm.

Answer source: 2.2.69.2 p. 4, step 2.26.4
2.2.69.2 p. 3, step 2.15

Distractors:

a. Reactor cavity is flooded and no loss of inventory should have occurred. Flow must be raised > 5000 gpm and ≤ 7000 gpm.
c. Flow is limited to 7000 gpm with fuel removed around instrument channels. With 180 bundles removed, at least some instrument dry tubes must be exposed.
d. Flow is limited to 7000 gpm with fuel removed around instrument channels. With 180 bundles removed, at least some instrument dry tubes must be exposed. Reactor cavity is flooded and no loss of inventory should have occurred.

55.43 section(s): (5)

SRO Justification: SRO from-memory knowledge of system precautions and limitations, prediction of integrated system response.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
82	4002	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0070501	OPS Introduction to Technical Specifications
INT0070504	CNS Tech. Spec. 3.3, Instrumentation

Related Objectives	
INT00705010010200	Given plant conditions and a Specification, apply the rules of Section 3.0 to determine appropriate actions.
INT00705040010300	Given a set of plant conditions that constitutes non-compliance with a Section 3.3 LCO, determine the ACTIONS that are required.

Related References	
3.0	"LCO and Surveillance Applicability"
3.3.1.1	Reactor protection system (RPS) instrumentation

Related Skills (K/A)	
2.1.12	Ability to apply technical specifications for a system. (CFR: 43.2 / 43.5 / 45.3) (2.9/4.0)

QUESTION: 4002

The plant is operating at full power with the "A" and "C" APRMs failed downscale due to internal circuitry problems. The Control Room operators have placed a half scram on RPS "A".

What effect will this condition have on the continued ability to operate the plant in Mode 1?

a. Unless one of the failed APRMs is returned to normal service prior to the next scheduled Channel Functional Test of one of the RPS B side APRMS, the plant will have to be shutdown.

b. Continued operation in this condition is not permitted. Tech Specs requires immediate action to initiate insertion of all insertable control rods in control cells containing one or more fuel assemblies.

c. While in this condition, the remaining operable APRMs must be channel checked every 12 hours AND channel functional tests must be performed every 7 days, with the exception that their trip units need not be tripped as long as there are no abnormal responses observed in the performance of either test.

d. Continued operation is permitted indefinitely in this condition as Tech Specs allows resetting the trip inserted due to the failed APRMs in order to perform testing required to demonstrate the operability of the operable APRMs in service that are associated with the B logic of the Reactor Protection System.

ANSWER: 4002

d. Continued operation is permitted indefinitely in this condition as Tech Specs allows bypassing the trip functions of the failed APRMs in order to perform testing required to demonstrate the operability of the operable APRMs in service that are associated with the B logic of the Reactor Protection System.

Answer source: Tech Spec 3.0.5

Distractors:

a. Tech spec 3.0.5 allows equipment removed from service or declared inoperable to comply with ACTIONS to be returned to service to support testing of other equipment.
b. Tech spec 3.0.5 allows equipment removed from service or declared inoperable to comply with ACTIONS to be returned to service to support testing of other equipment. Trip capability is maintained.
c. The channel functional tests are still required for "B" side instruments.

2.1.12 Ability to apply technical specifications for a system.(CFR: 43.2 / 43.5 / 45.3) as it applies to: 215005 APRM / LPRM

55.43 section(s): (2)

SRO Justification: SRO persons assess plant conditions and determine compliance with Technical Specification and action required for non-compliance. Technical Specification REQUIRED ACTION and application of 3.0.5.

Source: ***Direct from Bank***

Provide to Candidate: T.S. 3.0 section and bases, T.S. LCO 3.3.1.1 and bases.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
83	38	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0070501	OPS Introduction to Technical Specifications
INT0070509	CNS Tech. Spec. 3.8, Electrical Power System
INT0070504	CNS Tech. Spec. 3.3, Instrumentation

Related Objectives	
INT00705010010200	Given plant conditions and a Specification, apply the rules of Section 3.0 to determine appropriate actions.
INT00705040010300	Given a set of plant conditions that constitutes non-compliance with a Section 3.3 LCO, determine the ACTIONS that are required.
INT00705090010300	Given a set of plant conditions that constitutes non-compliance with a Section 3.8 LCO, determine the ACTIONS that are required.

Related References	
3.3.3.1	Post accident monitoring (PAM) instrumentation
3.8.1	AC Sources - Operating

Related Skills (K/A)	
216000.K6.01	Knowledge of the effect that a loss or malfunction of the following will have on the NUCLEAR BOILER INSTRUMENTATION: (CFR: 41.7 / 45.7) A.C. electrical distribution (3.1/3.3)

QUESTION: 38

The plant was operating at rated power when the following occurred:

- NBI-LI-85A (Wide Range RPV water level) becomes inoperable at 0900 on 7/06
- An air leak in the starting air system for DG2 occurs at 1200 on 7/12. Air pressure lowers to 100 psig.

IF conditions do not change, which of the following is the **LATEST** that LCO 3.3.3.1 "PAM Instrumentation" allows the plant to enter **MODE 3**?

a. 1600 on 7/12.

b. 1600 on 7/17.

c. 2400 on 7/19.

d. 0400 on 7/20.

ANSWER: 38

d. 0400 on 7/20.

3.8.3.F requires DG2 be declared inoperable immediately.

NBI-LI-85A (Wide Range RPV water level PAM instrument powered from CCP-1A) is inoperable. DG #2 becomes inoperable requiring 4 hours later, the Conditions and Required Actions for both Wide Range RPV water level PAM instruments inoperable (3.3.3.1 Condition "C") must be entered as the inoperable PAM instrument on a division opposite that of the inoperable DG. NBI-LI-85B is powered by CCP which is supported by DG #2. NBI-LI-85A is "an inoperable redundant required feature supported by the other DG".

Enter 3.3.3.1.C at 1600 on 7/12. Enter 3.3.3.1.D at 1600 on 7/19. Enter 3.3.3.1.E at 1600 on 7/19. Be in MODE 3 0400 on 7/20

Answer source: Tech Spec LCO 3.3.3.1 and 3.8.1

Distractors:

a. Time when 3.3.3.1.C is entered.
c. Math error for adding 7 days to 7/12.
d. Time when DG spec (3.8.1) requires MODE 3

55.43 section(s): (2)

SRO Justification: SRO persons assess plant conditions and determine compliance with Technical Specification and action required for non-compliance. SRO personnel implement Technical Specifications.

Source: ***New***

Provide to Candidate: T.S. 3.0 section and bases, T.S. LCO 3.3.3.1 and bases, T.S. LCO 3.8.1 and bases, T.S. LCO 3.8.3 and bases.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
84	48	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0070502	CNS Tech. Spec. 3.1, Reactivity Control Systems

Related Objectives	
INT00705020010100	Given a set of plant conditions, recognize non-compliance with a Section 3.1 LCO.
INT00705020010300	Given a set of plant conditions that constitutes non-compliance with a Section 3.1 LCO, determine the ACTIONS that are required.

Related References	
3.1.3	Control rod operability
3.1.4	Control rod scram times

Related Skills (K/A)	
2.2.21	Knowledge of pre and post maintenance operability requirements.(CFR: 43.2) (2.3/3.5)

QUESTION: 48

During the current refueling outage, the control rod drive mechanism was removed, rebuilt and then reinstalled for control rod 26-27.

How are control rod surveillance requirments and the plant startup coordinated?

A coupling check *must* be performed on the rod . . .

a. ***prior*** to reactor startup. Control rod scram time does not need to be tested until reactor pressure is at least 800 psig and ***must*** be completed ***prior*** to exceeding 40% power.

b. ***prior*** to reactor startup. Control rod scram time ***must*** be tested ***prior*** to startup and tested again when reactor pressure is at least 800 psig (***prior*** to exceeding 40% power).

c. when it is fully withdrawn during the startup, but is **NOT** required ***prior*** to the reactor startup. Control rod scram time does not need to be tested until reactor pressure is at least 800 psig and ***must*** be completed ***prior*** to exceeding 40% power.

d. when it is fully withdrawn during the startup, but is **NOT** required ***prior*** to the reactor startup. Control rod scram time ***must*** be tested ***prior*** to startup and tested again when reactor pressure is at least 800 psig (***prior*** to exceeding 40% power).

ANSWER: 48

b. ***prior*** to reactor startup. Control rod scram time ***must*** be tested ***prior*** to startup and tested again when reactor pressure is at least 800 psig (***prior*** to exceeding 40% power).

SR 3.1.3.5 requires coupling check prior to calling a control rod operable after work that could affect coupling. SR 3.1.4.3 (referenced from SR 3.1.3.4) requires scram testing at reduced pressure prior to calling a control rod operable after work that could affect scram time. SR 3.1.4.1 requires scram testing at or above 800 psig prior to exceeding 40% power.

<u>Answer source:</u> SR 3.1.3.4, SR 3.1.3.5, SR 3.1.4.1 & SR 3.1.4.3

Distractors:

a. Scram times must be checked prior to calling control rod operable.
c. Coupling check is required prior to statup. Scram times must be checked prior to calling control rod operable.
d. Coupling check is required prior to statup.

2.2.21 Knowledge of pre and post maintenance operability requirements.(CFR: 43.2) as it applies to:
201003 Control Rod and Drive Mechanism

55.43 section(s): (2) & (6)

SRO Justification: SRO persons assess plant conditions and determine compliance with Technical Specification and action required for non-compliance. Integration of multiple Technical Specification surveillance requirements.

Source: ***New***

Provide to Candidate: T.S. 3.0 section and bases, T.S. LCO 3.1.3 and bases, 3.1.4 and bases.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
85	114	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0070502	CNS Tech. Spec. 3.1, Reactivity Control Systems

Related Objectives	
INT00705020010300	Given a set of plant conditions that constitutes non-compliance with a Section 3.1 LCO, determine the ACTIONS that are required.

Related References	
3.1.6	Rod pattern control
3.3.2.1	Control rod block instrumentation

Related Skills (K/A)	
2.2.33	Knowledge of control rod programming. (CFR: 43.6) (2.5/2.9)

QUESTION: 114

A plant casualty occurred which required a rapid power reduction to 8% RTP. As a result, the RWM has initiated control rod Insert and Withdrawal blocks. Seventeen (17) control rods are Withdrawal errors and two (2) control rods are Insert errors.

What actions are required per Technical Specifications?

a. Suspend withdrawal of control rods immediately. Place the reactor mode switch in the shutdown position within 1 hour. Actions may continue to insert Withdrawal error control rods.

b. Bypass RWM and insert/withdraw the control rods to the correct position per the Banked Position Withdrawal Sequence (BPWS) OR declare associated control rods inoperable with 8 hours. Movement of control rods shall be verified in accordance with BPWS by a second licensed operator or other qualified member of the technical staff.

c. Bypass RWM and withdraw the Insert error control rods to the correct position per the Banked Position Withdrawal Sequence (BPWS). Declare the Withdrawal error control rods inoperable with 8 hours. Movement of the control rods shall be verified in accordance with BPWS by a second licensed operator or other qualified member of the technical staff.

d. Suspend control rod movement except by scram immediately OR verify at least 12 rods are still withdrawn immediately OR verify that startup with RWM inoperable has not been performed within the last calendar year immediately. Movement of control rods shall be verified in accordance with BPWS by a second licensed operator or other qualified member of the technical staff.

ANSWER: 114

a. Suspend withdrawal of control rods immediately. Place the reactor mode switch in the shutdown position within 1 hour. Actions may continue to insert Withdrawal error control rods.

Answer source: Tech Spec 3.1.6 and 3.3.2.1

Distractors:

b. Must discontinue withdrawal of control rods. RMS must be placed in shutdown within 1 hour.
c. Must discontinue withdrawal of control rods. RMS must be placed in shutdown within 1 hour.
d. Can insert rods. The 12 rod specification is specific to RWM operability and plant startup only.

55.43 section(s): (2)

SRO Justification: SRO persons assess plant conditions and determine compliance with Technical Specification and action required for non-compliance. SRO ability to implement Tech Specs.

Source: ***Direct from bank***

Provide to Candidate: T.S. 3.0 section and bases, T.S. LCO 3.1.6 and bases, 3.3.2.1 and bases.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
86	42	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0070505	CNS Tech. Spec. 3.4, Reactor Coolant System (RCS)

Related Objectives	
INT00705050010200	Discuss the applicable Safety Analysis in the Bases associated with each Chapter 3.4 Specification.

Related References	
3.4.2	Jet pumps

Related Skills (K/A)	
202001.K1.06	Knowledge of the physical connections and/or cause-effect relationships between RECIRCULATION SYSTEM and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) Jet pumps (3.6/3.6)

QUESTION: 42

Why is a plant shutdown required if one or more jet pumps is determined to be inoperable?

a. Excessive vibration of the jet pumps may occur.

b. Increased risk of uncontrolled thermal hydraulic oscillations.

c. The assumed blowdown flow during a LOCA may exceed the analyzed value.

d. The reactor may not have sufficient flow coastdown characteristics to maintain fuel thermal margins during abnormal operational transients.

ANSWER: 42

c. The assumed blowdown flow during a LOCA may exceed the analyzed value.

Answer source: Tech Spec Bases p. B 3.4-10, top paragraph

Distractors:

a. Basis for recirc pump speed mismatch.
b. Basis for power/flow limitations.
d. Basis for loop flow mismatch.

55.43 section(s): (2)

SRO Justification: SRO persons assess plant conditions and determine compliance with Technical Specification and action required for non-compliance. SRO knowledge of Tech Spec basis.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
87	16415	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0320101	CNS Administrative Procedures Volume 0, Administrative Procedures (Formal Classroom/Pre-OJT Training)
INT0070507	CNS Tech. Spec. 3.6, Containment Systems

Related Objectives	
INT032010100G010L	Discuss the following as described in Administrative Procedure 0.26, Surveillance Program: Precautions and Limitations
INT00705070010100	Given a set of plant condtions, recognize non-compliance with a Chapter 3.6 LCO.

Related References	
3.6.1.3	Primary Containment Isolation Valves (PCIVs)
3.6.2.3	Residual Heat Removal (RHR) Suppression Pool Cooling

Related Skills (K/A)	
2.2.21	Knowledge of pre and post maintenance operability requirements.(CFR: 43.2) (2.3/3.5)

QUESTION: 16415

The plant is at power with the 6.1RHR.201, RHR Power Operated Valve Operability Test in progress. When attempting to time open RHR-MO-34A, SUPPR POOL COOLING INBD THROTTLE VLV, the valve remained closed and the breaker tripped at the MCC. The Station Operator reports an acrid smell at the MCC. The valve motor actuator gear set was replaced last outage.

Which one of the following describes . . .

(1) if the breaker can be reset to try to stroke the valve (yes or no)
(2) if the breaker will not reset, are TS 3.6.1.3 and/or TS 3.6.2.3 entered?

a. (1) No (2) Both TS are entered

b. (1) No (2) Only TS 3.6.2.3 is entered

c. (1) Yes (2) Both TS are entered

d. (1) Yes (2) Only TS 3.6.2.3 is entered

ANSWER: 16415

a. (1) No (2) Both TS are entered

Immediately declare the valve inoperable and enter the ACTIONS of TS 3.6.2.3. TS 3.6.1.3 is also entered because the valve is a primary containment isolation valve.

Answer source: Tech Spec 3.6.2.3 and 3.6.1.3

Distractors:

b. TS 3.6.1.3. entry is required because this is a PCIS valve.
c. Breaker cannot be reset and closed.
d. Breaker cannot be reset and closed. TS 3.6.1.3. entry is required because this is a PCIS valve

55.43 section(s): (2)

SRO Justification: SRO persons assess plant conditions and determine compliance with Technical Specification and action required for non-compliance. SRO responsible for sureillance requirements and implementing requirements.

Source: ***Direct from bank***

Provide to Candidate: T.S. 3.0 section and bases, T.S. LCO 3.6.1.3 and bases, 3.6.2.3 and bases, 6.1RHR.201**.**

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
88	24	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0070507	CNS Tech. Spec. 3.6, Containment Systems

Related Objectives	
INT00705070010200	Discuss the applicable Safety Analysis in the Bases associated with each Chapter 3.6 specification.

Related References	
3.6.2.1	Suppression pool average temperature

Related Skills (K/A)	
295013.AK1.04	Knowledge of the operational implications of the following concepts as they apply to HIGH SUPPRESSION POOL TEMPERATURE: (CFR: 41.8 to 41.10) Complete condensation. (2.9/3.2)

QUESTION: 24

The plant is operating at rated power with the following conditions:

- ➢ High Pressure Coolant Injection (HPCI) was in service for the quarterly full-flow test.
- ➢ The crew removed HPCI from service when average torus water temperature reached 105°F.
- ➢ 26 hours after HPCI was secured, average torus water temperature is 98°F (due to elevated Missouri River temperature.)

What is a potential consequence of continuing to operate the reactor at power under these conditions?

a. HPCI will not be available during a LOCA due to elevated lube oil temperature.

b. The capacity of the torus-to-drywell vacuum breakers will be exceeded during a LOCA.

c. The reactor building-to-torus vacuum breakers will operate if drywell sprays are initiated during a LOCA.

d. Steam will not completely condense at the outlet of the drywell to suppression pool downcomers during a LOCA.

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ANSWER: 24

d. Steam will not completely condense at the outlet of the drywell to suppression pool downcomers during a LOCA.

Answer source: Tech Spec bases p. B 3.6-51

Distractors:

a. HPCI is normally aligned to the ECST and lube oil is not affected until 140°F.
b. This would be an issue for initiating drywell spray with evaporative cooling and is positively affected by high torus temp, not negatively affected.
c. This would be an issue for initiating drywell spray with evaporative cooling and is positively affected by high torus temp, not negatively affected.

55.43 section(s): (2)

SRO Justification: SRO persons assess plant conditions and determine compliance with Technical Specification and action required for non-compliance. Technical Specification bases for maximum suppression pool temperature from memory.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
89	40	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
COR0020802	DIESEL GENERATORS
INT0070509	CNS Tech. Spec. 3.8, Electrical Power System

Related Objectives	
COR0020802001090C	Describe the Diesel Generator design feature(s) and/or interlock(s) that provide for the following: Speed Droop Control
INT00705090010100	Given a set of plant condtions, recognize non-compliance with a Section 3.8 LCO.

Related References	
2.2.20	Procedure 2.2.20, Standby AC Power System (Diesel Generator)
2.2.20.1	Procedure 2.2.20.1, Diesel Generator Operations

Related Skills (K/A)	
264000.K5.04	Knowledge of the operational implications of the following concepts as they apply to EMERGENCY GENERATORS (DIESEL/JET): (CFR: 41.5 / 45.3) Governor control (2.4/2.5)

QUESTION: 40

How is DG1 operation affected if the DROOP/PARALLEL Switch left in DROOP?

During a design bases LOCA coincident with off-site power, DG1 will . . .

a. reach rated voltage and frequency within design time limits. Bus voltage and frequency will be maintained as assumed in the accident analysis as bus loads sequence on.

b. **NOT** reach rated voltage and frequency within design time limits. Bus voltage and frequency will be maintained as assumed in the accident analysis as bus loads sequence on.

c. reach rated voltage and frequency within design time limits. Bus voltage and frequency will **NOT** be maintained as assumed in the accident analysis as bus loads sequence on.

d. **NOT** reach rated voltage and frequency within design time limits. Bus voltage and frequency will **NOT** be maintained as assumed in the accident analysis as bus loads sequence on.

ANSWER: 40

a. reach rated voltage and frequency within design time limits. Bus voltage and frequency will be maintained as assumed in the accident analysis as bus loads sequence on.

Answer source: 2.2.20 p. 23, step 1.4.3.6

Distractors:

b. The droop mode does not affect start time or sequence.
c. The DG will shift to Isochronous mode automatically when an auto start signal exists. This overrides the switch position on the panel for droop.
d. The droop mode does not affect start time or sequence. The DG will shift to Isochronous mode automatically when an auto start signal exists. This overrides the switch position on the panel for droop.

55.43 section(s): (5)

SRO Justification: SRO persons assess plant conditions and determine compliance with Technical Specification and action required for non-compliance. SRO knowledge of diesel logic and plant accident analysis.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
90	46	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0070604	TRM - Fire Protection

Related Objectives	
INT0070604001010D	Given plant conditions, determine if the following TRM Limiting Conditions for Operation (TLCOs) are met: T.3.11.4 High Pressure Carbon Dioxide Extinguishing Systems
INT0070604001010F	Given plant conditions, determine if the following TRM Limiting Conditions for Operation (TLCOs) are met: T.3.11.6 Fire Hose Stations
INT0070604001030B	Given plant conditions and the TRM, determine the ACTIONS required per the following TLCOs: T.3.11.2 Fire Suppression Water System
INT0070604001010B	Given plant conditions, determine if the following TRM Limiting Conditions for Operation (TLCOs) are met: T.3.11.2 Fire Suppression Water System

Related References	
2.2.2	Procedure 2.2.2, Carbon Dioxide Systems
T3.11.6	Fire Hose Stations

Related Skills (K/A)	
286000.K6.02	Knowledge of the effect that a loss or malfunction of the following will have on the FIRE PROTECTION SYSTEM (CFR: 41.7 / 45.7) D. C . electrical distribution (2.8*/2.9*)

QUESTION: 46

The plant is operating at rated power when 125 VDC is lost to the CO_2 hose station subsystem.

What is the effect on the CO_2 hose stations and what are the **MINIMUM** actions required?

On loss of DC power to Cardox System, hose stations will . . .
(NOTE: Choices are listed in MINIMUM to MAXIMUM order.)

a. charge but not vent. Document on Fire Impairment Form that no compensatory actions required.

b. vent but not charge. Contact Fire Protection Group for determination of appropriate compensatory measure based on nature of impairment and its impact on safe shutdown timeliness.

c. vent but not charge. Establish 2 hour Fire Watch patrol in affected areas within 2 hours.

d. charge but not vent. Establish a continuous fire watch with backup fire suppression equipment for the unprotected area(s) within one (1) hour AND restore to OPERABLE status within 14 days.

ANSWER: 46

a. charge but not vent. Document on Fire Impairment Form that no compensatory actions required.

Answer source: 2.2.2 p.2, step 2.3
0.23 p. 13, Attachment 1

Distractors:

b. This is action for Non-Tech Spec Appendix R Safe Shutdown Capability. Fire hose stations are not Appendix R.
c. 0.23 required action for non-TRM fire detection equipment. This is not fire detection equipment.
d. TRM action for inoperable required fire detection equipment. This is not a TRM system and is not fire detection equipment.

55.43 section(s): (2) & (5)

SRO Justification: SRO persons assess plant conditions and determine compliance with Technical Requirements Manual (TRM) and action required for non-compliance. SRO use of TRM, administrative procedures and fire impairments.

Source: ***New***

Provide to Candidate: TRM section 3.11 and 0.23 "CNS FIRE PROTECTION PLAN".

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
91	47	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0070702	ODAM Specifications

Related Objectives	
INT00707020010200	Given the ODAM Appendix D and given plant condition, determine whether there is non-compliance with the ODAM Specification Sections 3.1 thru 3.5.

Related References	
D3.2.7	Primary Containment Venting and Purging

Related Skills (K/A)	
2.3.9	Knowledge of the process for performing a containment purge. (CFR: 43.4 / 45.10) (2.5/3.4)

QUESTION: 47

The plant has been shutdown for 30 days for a refueling outage. A plant startup is in progress with reactor power currently 25%. Preparations are underway to purge the primary containment with nitrogen.

When the primary containment is purged, should the purge flowpath utilize Standby Gas Treatment and why/why not?

a. Yes. The SBGT system shall be used to vent or purge the primary containment to ensure deflagration limits are not exceeded at the Elevated Release Point (ERP).

b. Yes. The SBGT system shall be used to vent or purge the primary containment while coolant temperature is greater than 200°F to ensure dose rates do not exceed small fractions of 10 CFR 100 limits.

c. No. To minimize the time that Oxygen concentration is above 4% to reduce the risk of a combustible atmosphere if a LOCA were to occur, the large purge and vent valves should be used.

d. No. To minimize the time that SBGT system is on line while coolant temperature is greater than 200°F to reduce the risk of damage to SBGT system if a LOCA were to occur with the main purge and vent valves open.

ANSWER: 47

d. No. To minimize the time that SBGT system is on line while coolant temperature is greater than 200°F to reduce the risk of damage to SBGT system if a LOCA were to occur with the main purge and vent valves open.

Answer source: ODAM bases p. 3.2-7

Distractors:

a. The large vent/purge valves should be used under these conditions.
b. The large vent/purge valves should be used under these conditions.
c. The basis for using the large valves is to minimize risk to SBGT.

55.43 section(s): (2) & (4)

SRO Justification: SRO persons assess plant conditions and determine compliance with Off site Dose Assessment Manual (ODAM) and action required for non-compliance.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
92	30	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0080613	OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL

Related Objectives	
INT00806130010900	Explain why HPCI but not RCIC must be secured at a primary containment water level of 11 feet.
INT00806130011100	Given plant conditions and EOP Flowchart 3A, PRIMARY CONTAINMENT CONTROL, determine required actions.

Related References	
EOP 3A, PCCP	EOP 3A Primary Containment Control Procedure

Related Skills (K/A)	
295030.EA1.05	Ability to operate and/or monitor the following as they apply to LOW SUPPRESSION POOL WATER LEVEL: (CFR: 41.7 / 45.6) HPCI (3.5/3.5)

QUESTION: 30

A LOCA has occurred with the following conditions:

- HPCI is injecting at 3000 gpm
- NO other high pressure injection source is available
- Reactor pressure is being maintained 800 to 1000 psig
- RPV water level is -102 inches (stable WR)
- HPCI suction is aligned to the torus
- Primary containment water level is 11.2 feet **AND** is dropping 6"/minute
- Average torus water temperature is 135°F **AND** is rising

What actions are required regarding HPCI operation?

a. Defeat HPCI high temperature interlocks.

b. Trip HPCI and place HPCI auxiliary oil pump in Pull-to-Lock.

c. Increase HPCI flow to restore and maintain RPV water level +3" to +54".

d. Reduce HPCI flow as necessary to maintain HPCI oil temperatures below 140°F.

ANSWER: 30

b. Trip HPCI and place HPCI auxiliary oil pump in Pull-to-Lock.

Answer source: EOP flowchart 3A, step SP/L-10

Distractors:

a. HPCI must be placed in PTL per EOP-3A. There is no need to defeat HPCI high temperature isolations.
c. HPCI must be placed in PTL irrespective of adequate core cooling.
d. The 140°F caution is for torus water temperature, not oil temperatures.

55.43 section(s): (5)

SRO Justification: SRO licensed personnel assess plant conditions during implementation of the EOPs and, based on that assessment, direct EOP actions.

Source: ***New***

Provide to Candidate: EOP-3A with cautions and entry conditions removed.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
93	34	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0080617	OPS FLOWCHART 5A - SECONDARY CONTAINMENT AND RADIOACTIVITY RELEASE CONTROL

Related Objectives	
INT00806170010600	Given plant conditions and EOP flowchart 5A, SECONDARY CONTAINMENT CONTROL and RADIOACTIVITY RELEASE CONTROL, determine required actions.

Related References	
EOP 5A, SCCP	EOP 5A, Secondary Containment Control

Related Skills (K/A)	
295032.EA2.01	Ability to determine and/or interpret the following as they apply to HIGH SECONDARY CONTAINMENT AREA TEMPERATURE: (CFR: 41.10 / 43.5 / 45.13) Area temperature (3.8*/3.8)

QUESTION: 34

The plant was operating at rated power when a steam leak on RCIC occurred. The crew attempted to isolate RCIC with no success. The reactor has been scrammed and the following conditions now exist:

- ➢ The Reactor Building is inaccessible due to the steam leak
- ➢ A complete loss of pneumatics (air and nitrogen) to the Reactor Building exists
- ➢ Reactor pressure is being maintained at 926 psig by DEH
- ➢ Reactor water level is at 25" (NR) and stable
- ➢ NE Quad temperature is 205°F (rising)
- ➢ SE Quad temperature is 170°F (rising)
- ➢ NW Quad temperature is 200°F (rising)
- ➢ SW Quad temperature is 165°F (rising)
- ➢ RWCU area temperature is 135°F (rising)
- ➢ 903' area temperature is 135°F (rising)

What action is required by the EOPs?

a. Emergency Depressurize the RPV.

b. Cooldown the RPV at < 100 °F/hr.

c. Place SRV control switches in AUTO.

d. Fully open main turbine bypass valves.

ANSWER: 34

a. Emergency Depressurize the RPV.

Answer source: EOP flowchart 5A, step SC-14

Distractors:

b. Emergency Depressurization is required when 2 areas exceed maximum safe operating value.

c. Emergency Depressurization is required when 2 areas exceed maximum safe operating value. 5.8.1 says to put the SRV switches in AUTO when there has been a loss of pneumatics, but the ED guidance supercedes.

d. Emergency Depressurization is required when 2 areas exceed maximum safe operating value. If "Emergency Depressurization is Required", then "Emergency Depressurization is NOT anticipated"

<u>55.43 section(s):</u> (5)

<u>SRO Justification:</u> SRO licensed personnel assess plant conditions during implementation of the EOPs and, based on that assessment, direct EOP actions.

<u>Source</u>: ***New***

<u>Provide to Candidate:</u> EOP-1A and EOP-5A with entry conditions and cautions removed.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
94	16569	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
INT0080618 EOP AND SAG GRAPHS

Related Objectives	
INT00806180010300	Given plant conditions and the EOP and SAG Graphs Flowchart, determine if operation is within the allowed region of a graph.
INT00806180010200	For each graph used in the flowcharts, identify the action(s) required if the parameters associated indicate operation in the restricted or prohibited area.

Related References	
EOP 3A, PCCP	EOP 3A Primary Containment Control Procedure
5.8	Procedure 5.8, Emergency Operating Procedures (EOPs)

Related Skills (K/A)	
295026.EA1.01	Ability to operate and/or monitor the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: (CFR: 41.7 / 45.6) Suppression pool cooling (4.1/4.1)

QUESTION: 16569

A LOCA has occurred with the following conditions:

- PMIS is unavailable.
- RHR pump "C" in the suppression pool cooling, torus spray and drywell spray mode.
- Torus pressure is 4 psig (stable)
- Torus average water temp is 185°F (rising slowly)
- Primary containment water level is 7 feet (stable)

The SS directs that all EOP Cautions be complied with.

What is the MAXIMUM "C" RHR pump flow allowed?

a. 0 gpm

b. 5,000 gpm

c. 6,500 gpm

d. 10,000 gpm

ANSWER: 16569

c. 6,500 gpm.

With 4 psig overpressure and 3 feet of water above the suctions there is 5.29 psig overpressure requiring that flow be reduced to no more than 7000 gpm for NPSH concerns. Flow is also in the unsafe region of the vortex limit curve, but this curve is less limiting than the NPSH curve for this case.

Answer source: EOP graph 5 and NOTE 3

Distractors:

a. Correct if wrong (0#) curve is used.
b. Correct if wrong curve (CS) is used.
d. Correct if wrong 5# curve is followed to end of graph.

55.43 section(s): (5)

SRO Justification: SRO licensed personnel must assess NPSH requirement if PMIS is unavailable. SRO licensed personnel assess plant conditions during implementation of the EOPs and, based on that assessment, direct EOP actions.

Source: ***Direct from bank***

Provide to Candidate: **All EOP graphs.**

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
95	28	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0320123	CNS Abnormal Procedures (RO) Reactivity

Related Objectives	
INT0320123F0F0100	Given plant condition(s), determine from memory the appropriate Abnormal/Emergency Procedure(s) to be utilized to mitigate the event(s).
INT0320123G0G0100	Given plant condition(s), determine from memory all immediate operator actions required to mitigate the event(s).

Related References	
2.4RXPWR	Procedure 2.4RXPWR, Reactor Power Anomalies

Related Skills (K/A)	
295014.AA2.03	Ability to determine and/or interpret the following as they apply to INADVERTENT REACTIVITY ADDITION: (CFR: 41.10 / 43.5 / 45.13) Cause of reactivity addition (4.0/4.3*)

QUESTION: 28

Reactor power is 20% with rods being pulled in preparation to sychronize the generator. Suddenly, Reactor power increases rapidly to 23% and stabilizes. There was no change in core plate differential pressure or reactor pressure ***prior*** to the power rise.

What caused the reactor power increase?

a. Dropped control rod.

b. Governor valve closure.

c. Loss of feedwater heating.

d. Runaway recirculation pump.

ANSWER: 28

a. Dropped control rod.

Answer source: 2.4RXPWR pp. 2 & 3
Reactor Theory

Distractors:

b. Governor valve closure would cause reactor pressure to rise slightly.
c. There is no feedwater heating prior to loading the turbine.
d. There was no change in core plate dp.

55.43 section(s): (5)

SRO Justification: Prediction of plant response and application with discrimination necessary to select the correct procedure to mitigate the event.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
96	5668	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0320102	CNS Administrative Procedures Site Services Procedures (Formal Classroom/Pre-OJT Training)

Related Objectives	
INT032010200C030A	Discuss the requirement associated with the following items: Visitor acces and departure

Related References	
5.4FIRE	General Fire Procedure
1.1	Station Security

Related Skills (K/A)			
2.4.25	Knowledge of fire protection procedures.	(CFR: 41.10 / 45.13)	(2.9/3.4)

QUESTION: 5668

The plant is operating at rated power at 0210 on a Tuesday morning when a fire started in the reactor building. The following conditions exist:

- ➢ The fire has been burning out of control for 20 minutes
- ➢ The fire brigade has been dispatched and is attempting to contain the fire
- ➢ The fire brigade leader has requested assistance from the Nemaha Fire Department
- ➢ The plant has been scrammed
- ➢ The Nemaha Fire Department has arrived at the plant

Which of the following **SHALL** occur to allow the members of the Nemaha Fire Department to access the plant?

a. Drive on site as soon as Security opens the gate.

b. They must sign the Visitor's Log, and have their vehicles searched.

c. Access is authorized by the Security Shift Supervisor, once vehicle search and headcount are complete.

d. They must obtain an emergency TLD from security, and have the Shift Supervisor/Security Shift Supervisor authorize vehicle access.

ANSWER: 5668

d. They must obtain an emergency TLD from security, and have the Shift Supervisor/Security Shift Supervisor authorize vehicle access.

Answer source: 1.1 p. 3, step 3.2.2
1.1 p. 3, step 4.1
1.1 p. 6, step 5.2.3

Distractors:

a. Security must issue TLDs and either the Security Shift Supervisor or the Shift Supervisor must authorize access.
b. Vehicle search is not required and the signing the visitors log is not required.
c. Vehicle search is not required prior to access.

2.4.25 Knowledge of fire protection procedures. (CFR: 41.10 / 45.13) as it applies to 600000 Plant Fire on Site

55.43 section(s): (5)

SRO Justification: Shift Supervisor is one of the persons that can authorize access under these conditions (RO licensed personnel cannot) .

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
97	9015	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0070505	CNS Tech. Spec. 3.4, Reactor Coolant System (RCS)
INT0320136	CNS Abnormal Procedures (RO) Miscellaneous

Related Objectives	
INT00705050010800	From memory, in MODES 1, 2, or 3, state the actions required in less than one hour if RCS pressure and temperature (P/T) limits LCO is not met (LCO 3.4.9).
INT0320136P0P0100	Given plant condition(s), and the applicable Abnormal/Emergency Procedure, determine the correct subsequent actions required to mitigate the event(s).

Related References	
3.4.9	RCS Pressure and Temperature (P/T) Limits
5.1ASD	Shutdown From Outside The Control Room

Related Skills (K/A)	
295016.AA2.06	Ability to determine and/or interpret the following as they apply to CONTROL ROOM ABANDONMENT: (CFR: 41.10 / 43.5 / 45.13) Cooldown rate. (3.3/3.5)

QUESTION: 9015

Control room abandonment is required due to toxic gas in the control room. The CRS at the Alternate Shutdown Panel commences a reactor cooldown. The CRS lowers reactor pressure from 900 psig to 550 psig in 30 minutes.

If the cooldown continues at the SAME RATE (°F/minute) for the next thirty minutes, what will be the status of cooldown rate limits?

a. No limits will be exceeded.

b. Administrative limits will be exceeded; but no technical specifications will be violated.

c. Administrative limits and technical specifications will be exceeded; thirty minutes is the maximum time allowed to restore cooldown limits.

d. Administrative limits and technical specifications will be exceeded; one hour is the maximum time allowed to restore cooldown limits.

ANSWER: 9015

c. Administrative limits and technical specifications will be exceeded; thirty minutes is the maximum time allowed to restore cooldown limits.

Cooldown rate is 110ºF/hour (900 psig + 14.7 = 915 psia ----> 534ºF,
550 psig + 14.7 = 565 psia ----> 479ºF, 534ºF - 479ºF = 55ºF / 30 minutes or 110ºF/hr)

Answer source: Steam tables
Tech Spec 3.4.9

Distractors:

a. is incorrect; administrative limit is <or= 90ºF and is being exceeded.
b. is incorrect; technicial specification limit is <or= 100ºF and is being exceeded.
d. is incorrect; technical specifications only allows thirty minutes to restore cooldown rate.

55.43 section(s): (2) & (5)

<u>SRO Justification:</u> SRO licensed personnel perform actions at the Alternate Shutdown Panel and performs the cooldown from outside control room.

<u>Source:</u> ***Direct from Bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
98	5760	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
SKL0080102	OPS WATCHSTANDING PRINCIPLES FOR LICENSED OPERATORS

Related Objectives	
SKL00801020011300	State the proper response to an identified procedural deficiency which adversely affects the performance of the procedure.
SKL00801020011400	Describe the instant, non-intent procedure change process and the conditions under which it may be used.

Related References	
0.1	Introduction to CNS Operations Manual
0.26	Surveillance Program
0.4	Procedure Change Process

Related Skills (K/A)	
2.2.6	Knowledge of the process for making changes in procedures as described in the safety analysis report. (CFR: 43.3 / 45.13) (2.3/3.3)

QUESTION: 5760

Post Maintenance Testing is required to be performed by IAC on a pressure switch. The PMT requires the use of a portion of a surveillance procedure. The testing requirements cannot be met with the procedure as written. The procedure can be changed without changing the intent of the procedure, and the change must be made instantly due to pending work stoppage.

Who is responsible for approving the procedure change so the PMT may be completed?

a. Any two individuals holding an SRO license.

b. IAC Supervisor and the duty Shift Supervisor.

c. The on-shift Shift Supervisor with SORC concurrence.

d. The on-shift Shift Supervisor and a second SRO licensed individual.

ANSWER: 5760

d. The on-shift Shift Supervisor and a second SRO licensed individual.

Answer source: 0.4 p. 3, step 5.5

Distractors:

a. One must be on-shift SS.
b. IAC Supervisor not allowed.
c. SORC concurrence not required.

55.43 section(s): (3)

SRO Justification: SRO knowledge of instant procedure change requirements. Only SRO licensed personnel are allowed to make instant procedure changes.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
99	12218	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0320116	Health Physics Procedures

Related Objectives	
INT03201160000200	State the requirements an individual must meet in order to exceed any whole body exposure limits

Related References	
9.ALARA.1	Procedure 9.ALARA.1, Personnel Dosimetry Program
0.ALARA.7	Planned Special Exposure

Related Skills (K/A)	
2.3.4	Knowledge of radiation exposure limits and contamination control / including permissible levels in excess of those authorized. (CFR: 43.4 / 45.10) (2.5/3.1)

QUESTION: 12218

During the current calendar year a Station Operator had received a Planned Special Exposure (PSE) of 3.5 rem (TEDE) and an Occupational Exposure of 0.5 rem (TEDE).

What is the MAXIMUM dose (if any) this Station Operator can receive during the remainder of the calendar year ***without*** obtaining any additional written approval?

a. 0.0 rem

b. 0.5 rem

c. 1.5 rem

d. 2.5 rem

ANSWER: 12218

b. 0.5 rem

The PSE is separate from the allowed calendar year exposure. Special permission is not required as long as the calendar year exposure remains below a total of 1000 mrem (1 rem). Since the calendar year exposure is currently 0.5 rem any additional exposure that brings the total to 1.0 rem requires written approval.

Answer source: 9.ALARA.1, p. 12, Section 6.1
0.ALARA.7, p. 9, step 1.1

Distractors:

a can receive another 0.5 rem without approval.
c,d. is incorrect. Because written approval must be obtained before exceeding 1000 mrem.

55.43 section(s): (4)

SRO Justification: Authorization of exposure in excess of adminiatrative limits is a supervisory function. SRO licensed personnel are supervisors of ROs and SOs. SRO knowledge of PSE and TEDE requirements is required to assess and authorize extensions of a limit.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
100	9684	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons	
INT0231001	Shutdown Risk Management
SKL0110101	SRO Upgrade Self-Study Program

Related Objectives	
SKL0110101001350A	0.50, Outage Management Program: Define the following terms as they apply to Administrative Procedure 0.50, Outage Management Program: 1) Acceptable isolation barrier 2) Available 3) Contingency Plan 4) Decay Heat Removal (DHR) capability 5) Defense in Depth 6) Functional 7) Key Safety Function 8) Operations with Potential for Draining the Reactor Vessel (OPDRV's) 9) Primary System Boundary 10) Protected.

Related References	
0.50	Outage Management Program

Related Skills (K/A)	
2.4.21	Knowledge of the parameters and logic used to assess the status of safety functions including: 1. Reactivity control 2. Core cooling and heat removal 3. Reactor coolant system integrity 4. Containment conditions 5. Radioactivity release control. (CFR: 43.5 / 45.12) (3.7/4.3)

QUESTION: 9684

The plant is in day 9 of a scheduled refueling outage. The core off load has been delayed and the cavity has not yet been flooded. RHR loop A is operating in shutdown cooling with RHR pump A. RHR pump C is out of service due to an electrical fault that developed on the pump motor breaker during the previous shift. The following maintenance activities are in progress:

- RHR pumps "B" and "D" impeller replacement
- #2 Diesel Generator Main Bearing replacement
- Core Spray "B" is out of service for pump seal work
- RWCU is out of service to replace the pump seals
- Steam Separator lift

Which of the following activities would restore plant configuration to within guidelines?

a. Restore RWCU to service.

b. Restore "B" Core Spray to service.

c. Restore #2 Diesel Generator to service.

d. Flood the reactor cavity and remove the fuel pool gates.

ANSWER: 9684

d. Flood the reactor cavity and remove the fuel pool gates.

The current plant configuration does not provide a backup means of decay heat removal. By flooding the reactor cavity a recognized mode of backup decay heat removal is established.

Answer source: 0.50 p. 51, Attachment 4, Backup Systems.
0.50 p. 51, Attachment 4, RWCU item.
0.50 p. 51, Attachment 4, Fuel Pool/Gates item.

Distractors:

a. The RWCU system is not capable of providing backup decay heat removal at this time.
b. Core Spray does not provide decay heat removal.
c. Restoring the DG to service would not provide a mode of decay heat removal.

NOTE: No decay heat curves needed for this question. It is expected the operator should know that RWCU will not provide adequate decay heat removal 9 days after shutdown.

55.43 section(s): (6)

SRO Justification: SRO personnel authorize work during outages and assess risk and defense in depth continuously. SROs direct activities to restore the plant configuration to within the guidelines.

Source: ***Direct from bank***

Provide to Candidate: 0.50, Outage Management Program.

Cooper Nuclear Station June 2003 NRC Exam	Start "C" CBP, Loss of ESST, Steam Flow Inst Fails, Circ Water Pump Fails, Torus Leak

Description:

The plant is near the end of cycle at 70% power with SLC pump 1A, "C" Condensate Booster Pump and the "A" CRD pump out of service for maintenance. The "C" Condensate Booster Pump is expected to be returned to service at the beginning of the shift. An ice storm with accumulation is in progress.

After the crew takes the watch, the "C" Condensate Booster Pump will be returned to service. After the Booster Pump is returned to service, the dispatcher contacts the control room and requests that Cooper increase power to maximum to relieve overloading on the transmission lines. The crew increases power at the load dispatchers request.

After power is increased to 80%, a loss of the 69KV line occurs. The crew responds to the failure and addresses Technical Specifications.

After Technical Specifications are addressed for the loss of the 69KV line, the "C" Main Steam Line steam flow detector fails downscale. The crew responds per 2.4RXLVL.

A station operator will call the control room to report smoke and sparks coming out of the connection box on "B" Circulating Water Pump motor. The crew will rapidly remove the Circ Water Pump from service. There will be no fire or need for the fire brigade.

A Suppression Pool leak then develops resulting in a lowering torus level and entrance into EOP Flowchart 5A due to Secondary Containment Water Levels and 3A due to lowering Suppression Pool Level. When level decreases to 11', HPCI operation is prevented. Before SP water level drops to 9.6 ft., the crew initiates a manual reactor scram, which is unsuccessful. The crew is able to complete the scram using the Reactor Mode Switch, ARI or the RPS test switches.

The reactor is depressurized either by anticipation of emergency depressurization, emergency depressurization or both before primary containment water level goes below 9.6 ft. When the crew opens 6 SRVs, SRV C fails to open, requiring the crew to open a LLS valve.

When the reactor is depressurized and reactor water level is stable the scenario is terminated.

Cooper Nuclear Station June 2003 NRC Exam		Start "C" CBP, Loss of ESST, Steam Flow Inst Fails, Circ Water Pump Fails, Torus Leak	
Event No.	**Malf. No.**	**Type**	**Event Description**
1	N/A	N	Restore C CBP
2	N/A	R	Raise Power per the Load Schedule
3	ED21	C (BOP)	Loss of the 69KV line
4	FW13	I (RO)	Steam Flow Signal Failure to RVLC System
5	None	C (BOP)	"B" Circ Water Pump motor sparking leads
6	PC08	M	Suppression Pool Leak
7	Various	C (RO)	Electrical ATWS
8	Override	C (BOP)	SRV Fails to Open on Emergency Depressurization

Critical Tasks:

1. Scram prior to depressurization below 785 psig.
2. Initiate emergency depressurization before primary containment water level goes below 9.6 ft.

Critical Tasks are underlined in ***bold italics*** in the scenario.

Event Number		1
Event Description		Restore C CBP
Time	**Position**	**Applicant's Actions or Behavior**
5	BOP	Refer to procedure 2.2.6. Ensure the following (contact S.O.): ♦ Contact RW & verify adequate Condensate Demineralizer lineup. ♦ Ensure Auxiliary Oil Pump C has been operating for ≥ 5 minutes. ♦ Pump oil reservoir level is normal and flow exits through bearing sightglasses. ♦ TEC-44, CONDENSATE BOOSTER PUMP C OIL COOLER INLET, is open. ♦ TEC-45, CONDENSATE BOOSTER PUMP C OIL COOLER OUTLET, is throttled to maintain oil temperature as high as possible but < 110°F. ♦ MC-MO-303, CONDENSATE BOOSTER PUMP C DISCHARGE, is closed.
10	BOP	Place "C" CBP in service: ♦ Jog open for ~ 2 seconds, MC-MO-303. ♦ Announce start of "C" CBP. ♦ Start Condensate Booster Pump C. ♦ Check Condensate Booster Pump C AMPS are in green band. ♦ Check MC-FCV-12, MIN FLOW VLV, opens. ♦ When pump is up to speed and oil pressure is normal, place AUXILIARY OIL PUMP C switch to OFF to stop pump. ♦ Place AUXILIARY OIL PUMP C switch to AUTO. ♦ Slowly open MC-MO-303. ♦ Press SYS & PUMP MIN FCV RESET button to close MC-FCV-12 provided pump flow is > 500 gpm. ♦ Check Condensate Booster Pump C is operating properly. ♦ Throttle TEC-45 to maintain oil temperature between 100°F and 110°F.

Event Number		1
Event Description		Restore C CBP
Time	**Position**	**Applicant's Actions or Behavior**

Event Number		2
Event Description		Raise Power per the Load Schedule
Time	**Position**	**Applicant's Actions or Behavior**
20	CRS	Direct the Reactor Operator raise power using reactor recirculation flow at a rate not to exceed 10 MWe/min.
	RO	Raise power using reactor recirculation flow at a rate not to exceed 10 MWe/min
	BOP	Inform Station Operators of pending power increase.

Event Number		3
Event Description		Loss of the 69KV line
Time	**Position**	**Applicant's Actions or Behavior**
30	BOP	♦ Respond to annunciator C-1/G-6 Bkr 1FS Auto Closure Not Permitted ♦ Respond to annunciator C-4/G-1 Bkr 1GS Auto Closure Not Permitted ♦ Respond to annunciator C-2/C-10 Emergency Transformer Undervoltage ♦ Report loss of 69 KV power.
	CRS	♦ Suspend power ascension. ♦ Contact Load Dispatcher for prognosis. ♦ Determine the Emergency Transformer is inoperable. ♦ Refer to Technical Specification 3.8.1. ♦ Determine required action A.1 (1 hour/8 hour LCO) and A.3 (7 day LCO) apply.

Event Number		4
Event Description		Steam Flow Signal Failure to RVLC System
Time	**Position**	**Applicant's Actions or Behavior**
35	RO	♦ Recognize and report FI-88B (panel 9-5) reads downscale. ♦ Recognize and report RPV water level is lowering.
	CRS	♦ Enter and direct the activities of 2.4RXLVL. ♦ Direct critical parameter monitoring of RPV water level ♦ Direct action points for reactor scram.
NOTE:		**The operator may not place the controllers in a manual mode if RPV water level has already stabilized.**
	RO	♦ Place RFC-LC-83, RFC-CS-RFPTA and/or RFC-CS-RFPTB in a manual mode as necessary to control RPV water level. ♦ Enter 2.4RXLVL Attachment 2. ♦ Ensure RFC-LC-83 (or RFC-CS-RFPTA and RFC-CS-RFPTB) is/are in MAN. ♦ Place 1 OR 3 ELEMENT LVL CONT SELECT to 1. ♦ Return RPV water level control to automatic per 2.2.28.1.

Event Number		5
Event Description		“B” Circ Water Pump motor sparking leads
Time	**Position**	**Applicant’s Actions or Behavior**
40	CRS	Direct “B” Circulating Water Pump be removed from service.
	BOP	♦ Place pump control switch to PULL-TO-LOCK, leave in PULL-TO-LOCK or place to NORMAL AFTER STOP, and check discharge valve closes and pump trips after valve has partially closed. ♦ Monitor condenser vacuum.

Event Number		6
Event Description		Suppression Pool Leak
Time	**Position**	**Applicant's Actions or Behavior**
45	BOP	Respond per annunciator S-1/A-1, REACTOR BLDG A SUMP HI-HI LEVEL
	CRS	♦ Enter and direct the activities of EOP-5A ♦ Direct sump pump operation be verified. ♦ Direct all quad coolers be started.
	BOP	♦ Verify sump pump operation. ♦ Start all quad coolers.
	RO	Report alarm 9-3-2/G-5, SUPPR POOL NR/WR LOW LEVEL
	CRS	♦ When Torus Level reaches -2", enter and direct the activities of EOP-3A. ♦ Direct PC water level be maintained above 11' per 5.8.14.
	RO	♦ Perform suppression pool makeup per Procedure 5.8.14. ♦ Report makeup water systems in-service and Suppression Pool Water Level still lowering.
	BOP	Once it is determined the torus leak cannot be isolated, place control switch for following TORUS AREA DR VLVs to CLOSE: ♦ RW-AO-767. ♦ RW-AO-768. ♦ RW-AO-769.
53	CRS	When it has been determined that PC level cannot be maintained above 11', direct HPCI be stopped and prevented.
	BOP	Stop and prevent HPCI operation.
55	CRS	When it has been determined that PC level cannot be maintained above 9.6', direct the reactor be scrammed (***Scram the reactor prior to depressurization below 785 psig.***)(see Event 7).
	RO	♦ Scram the reactor as directed (***Scram the reactor prior to depressurization below 785 psig.***) (see Event 7). ♦ Report all control rods inserted.

Event Number		6
Event Description		Suppression Pool Leak
Time	**Position**	**Applicant's Actions or Behavior**
	CRS	♦ Enter EOP 1A ♦ Assign RPV level / pressure control
	BOP	Ensure each of following initiated: ♦ PCIS Group 2,3 and 6 isolations. ♦ ECCS initiations
NOTE:	**Due to single element control, RPV water level may rise above the main turbine and reactor feed pump high level trip setpoint.**	
	RO	Restore and maintain RPV water level between +15 and +40".
	BOP	Maintain Rx pressure 800 to 1000 psig with DEH.
60	CRS	IF Emergency Depressurization is anticipated, direct Main Turbine Bypass valves be fully opened.
	BOP	IF directed, fully open Main Turbine Bypass valves.
65	CRS	***When it has been determined that PC level cannot be maintained above 9.6', direct the RPV be Emergency Depressurized.*** ♦ Exit 1A pressure leg, enter 2A. ♦ Check PC level above 6' ♦ Direct 6 SRVs be opened. ♦ Direct RPV water level be maintained +15 to +40".
	BOP	***Place 6 SRV switches in OPEN***. ♦ Report "C" SRV did not open (See Event 8) ♦ Place an additional SRV switch in OPEN.

Event Number		7
Event Description		Electrical ATWS
Time	**Position**	**Applicant's Actions or Behavior**
55	CRS	***Direct a manual scram be inserted.***
	RO	♦ Manually scram the reactor as directed ♦ Report ony ½ scram occurred. ♦ ***Place Reactor Mode Switch in shutdown (before depressurization below 785 psig).*** ♦ Report all control rods inserted.

Event Number		8
Event Description		SRV Fails to Open on Emergency Depressurization
Time	**Position**	**Applicant's Actions or Behavior**
65	BOP	♦ Report "C" SRV did not open ♦ Place an additional SRV switch in OPEN.

Critical Tasks	SAT	UNSAT
1. Scram prior to depressurization below 785 psig.		
2. Initiate emergency depressurization before primary containment water level goes below 9.6 ft.		

Booth Instructor Activities

Event #1: Restore C CBP

ROLE PLAY:

If asked as Radwaste, report adequate Condensate Demineralizers are in service to support power increase (7 F/D's precoated).

ROLE PLAY:

When asked as S.O. to verify "C" CBP ready to start, report:

- ♦ Pump oil reservoir level is normal and flow exits through bearing sightglasses.
- ♦ TEC-44, CONDENSATE BOOSTER PUMP C OIL COOLER INLET, is open.
- ♦ TEC-45, CONDENSATE BOOSTER PUMP C OIL COOLER OUTLET, is throttled to maintain oil temperature as high as possible but < 110°F.
- ♦ MC-MO-303, CONDENSATE BOOSTER PUMP C DISCHARGE, is closed.

ROLE PLAY:

IF asked, report lube oil pressure is 10 psig.

ROLE PLAY:

When asked as S.O. to jog open MC-MO-303, modify **Remote Function** FW15, MC-MO-303, CBP "C" Discharge Valve" to 5%. Report valve position as indicated on the computer one-line diagram.

ROLE PLAY:

When asked as S.O. to fully open MC-MO-303, modify **Remote Function** FW15, MC-MO-303, CBP "C" Discharge Valve" to 100%. Report valve position as indicated on the computer one-line diagram.

Booth Instructor Activities

Event #2: Raise Power per the Load Schedule

ROLE PLAY:
When directed by the Chief Examiner, call as the Load Dispatcher and request maximum MW output from Cooper.

ROLE PLAY:
If contacted as Reactor Engineer for power raising guidance, report the control rods are at the target rod pattern. Recommend use of reactor recirculation flow control for power increase.

Event #3: Loss of the 69KV line

ACTION:
When directed by the Chief Examiner (after power has been increased to 80%), activate **TRIGGER E3** to de-energize the Emergency Transformer.

ROLE PLAY:
If asked to investigate the loss of the 69 KV line:
- As load dispatcher, report a fault at CNS.
- As S.O., report no visible damage or cause of the failure.

ROLE PLAY:
IF asked, report OCB 1604 is closed and disconnects are closed.

Booth Instructor Activities

Event #4: Steam Flow Signal Failure to RVLC System

ACTION:
When directed by the Chief Examiner, activate **TRIGGER E4** to fail "B" Steam Flow detector downscale.

ROLE PLAY:
If contacted to investigate the instrument racks, report that nothing appears abnormal at any of the instrument racks.

ROLE PLAY:
If contacted as maintenance, report there is no maintenance in progress in the field at this time.

Event #5: "B" Circ Water Pump motor sparking leads

ROLE PLAY:
When directed by the Chief Examiner, call the control room as Turbine Building station operator and report sparks and smoke are coming out of the connection box on the "B" Circulating Water pump motor.

ROLE PLAY:
If asked, there is no fire and no need for the fire brigade.

ROLE PLAY:
When the Circ water breaker is tripped, report sparking has stopped and smoke is dissipating.

Booth Instructor Activities

<u>Event #6: Suppression Pool Leak</u>

<u>ACTION:</u>
When directed by the Chief Examiner, activate **TRIGGER E6** to start the Torus leak.

<u>ACTION</u>
When SO is directed to investigate the leak or A, (B, D) sump levels, utilize **TRIGGER E20** to simulate NW torus area door being opened (annunciator A-1/A-2).

<u>ROLE PLAY:</u>
When directed as SO to investigate, wait several minutes, then report that a welded seam on the Torus just inside the NW door has failed. A substantial amount of water is flowing out of the break. The leak is low on the torus.

<u>ACTION</u>
When done performing the investigation of the quads, delete annunciator override for annunciator A-1/A-2.(NW Torus Door)

<u>Event #7: Electrical ATWS</u>

No preplanned actions or role play for this event.

<u>Event #8: SRV Fails to Open on Emergency Depressurization</u>

No preplanned actions or role play for this event.

<u>TERMINATING CONDITIONS:</u>

<u>ACTION</u>:
When the following conditions are reached, place the simulator in **FREEZE**:

1. 6 SRV switches are in OPEN.
2. RPV water level has been stabilized.
3. Chief Examiner directs the termination of the scenario.

<u>CAUTION</u>: Do not reset the simulator until directed by the Chief Examiner.

SIMULATOR SET-UP

A. Materials required

None

B. Initialize the Simulator in IC-20 (Full Power, End of Cycle)

Batch File name A:\124818.

C. Change the Simulator conditions from those of the IC as follows:

1. Triggers

Number	File Name	Description
	None	

2. Malfunctions

Number	Title	Trigger	TD	Severity	Ramp	Initial
RD27	Automatic ARI Failure	A	N/A	N/A	N/A	N/A
ED21	Loss of the 69KV line	E3	0	0	0	As Is
FW13B	"B" Steam Flow Signal Failure	E4	0	100	0	As Is
PC08	Suppression Pool Leak	E6	0	50	7:00	As Is

3. Remotes

Number	Title	Trigger	TD	Value	Ramp
HV01	Outside Air Temperature (10 m)	A	0	26	0
HV02	Wind Speed	A	0	21	0
HV03	Wind Direction	A	0	5	0
HV05	River Water Temperature	A	0	40	0
HV17	Air Temperature (100 m)	A	0	27.2	0
HV19	Air Temperature (60 m)	A	0	26.5	0
FW15	MC-MO-303, CBP “C” Discharge Valve	A	0	0	0

4. Overrides

Instrument	Tag	Trigger	TD	Value	Ramp
“A” SLC pump green light (labeled as squib valve ready light)	ZLOSLCSWS1A[1]	A	N/A	OFF	N/A
“A” Squib Valve Ready Light	ZLOSLCSWS1A[3]	A	N/A	OFF	N/A
“A” SLC Pump Control Switch	ZDISLCSWS1A[1]	A	N/A	STOP	N/A
“A” Squib Continuity Alarm	RA:MUX09C072	A	N/A	ON	N/A
A3/F-3 Cond Stor Tank B Temp Low	RA:MUX06032	A	N/A	OFF	N/A
“A3G-2 Cond Stor tank A Heater On	RA:MUX10C070	A	N/A	OFF	N/A
“A” Manual Scram Pushbutton	ZDIRPSSWS3A[1]	A	0	PUSH-OUT	N/A
“C” SRV Control Switch	ZDIMSSWS1C[1]	A	0	AUTO	N/A
Control Switch for CRD Pump A	ZDICRDSWS3A[2]	A	N/A	NASP	N/A
Control Switch for CRD Pump A green light	ZLOCRDSWS3A[1]	A	N/A	OFF	N/A
5.8.3 RPS Jumper, channel A1 (EOP PTM #31)	ZDIRPSSCRMJA1	A	0	IN	N/A
5.8.3 RPS Jumper, channel A2 (EOP PTM #32)	ZDIRPSSCRMJA2	A	0	IN	N/A
5.8.3 RPS Jumper, channel B1 (EOP PTM #33)	ZDIRPSSCRMJB1	A	0	IN	N/A
5.8.3 RPS Jumper, channel B2 (EOP PTM #34)	ZDIRPSSCRMJB2	A	0	IN	N/A
RX Bldg NW Torus Door	RA:MUX10C107	E20	N/A	ON	N/A

5. Panel Set-up

a. Reduce reactor power to ~ 70% using reactor recirculation pumps (~ 46% speed) .

b. Trip “C” Condensate Booster pump.

c. Load Batch file.

d. Ensure “B” CRD pump is in service & “A” CRD pump is secured.

e. Hang a red tag on “A” CRD pump.

f. Hang a red tag on “A” SLC pump control switch.

g. Advance all charts and alarm typers to clean paper.

h. Adjust APRM GAFs.

i. Adjust 9A/9B for ~ 15000 gpm condensate flow.

j. Balance the Main Generator regulator.

Turnover Information

A. Plant Status:

1. The plant is operating at approximately 70% power near the end of the operating cycle.

2. Rod Sequence Information:

Page:	2
Rod:	18-27
Notch:	26

3. Tech. Spec. Limitations in effect:

   Day 2 of 7 day LCO (3.1.7.A.1) for “A” SLC pump. The pump tripped during a surveillance. Maintenance is investigating.

4. Significant problems/abnormalities:

   ♦ “C” CBP is out of service due to a hot motor connection. Maintenance has repaired the connection and the pump is ready to be placed in service.
   ♦ An ice storm with accumulation is in progress.

5. Evolutions/maintenance for the on-coming shift:

   ♦ Continue investigation of “A” SLC pump.
   ♦ Restore “C” CBP to service.
   ♦ Raise power to 100% after “C” CBP has been returned to service per load schedule.

LOAD SCHEULE

♦ Maintain ~ 500 Mwe
♦ As soon as CBP started, then raise for:
  - 1st hour 600 Mwe
  - 2nd hour 700 Mwe
  - 3rd hour RATED full power

Cooper Nuclear Station June 2003 NRC Exam	Shift Bus Duct Fans, Core Spray Line Break, Air Compressor Trip, Loss of Vacuum, LOOP & LOCA

Description:

The plant is operating at 100% power near the end of the fuel cycle. Slight grid instabilities are present due to record grid loading. The load schedule requires that CNS maintain 100% power. SLC pump 1A is tagged out for corrective maintenance.

The crew shifts the operating bus duct fan as the running fan was making loud noises last shift.

After the crew completes the swap of the bus duct fans, “B” Core Spray line breaks inside the RPV. The crew will respond per annunciator procedures and Technical Specifications.

After Technical Specifications have been assessed, the running air compressor trips. The crew responds to the loss of the air compressor and initiates an investigation into the cause.

After the air system is stabilized, condenser vacuum begins to slowly lower due to increased air in-leakage. The crew responds to the lowering vacuum and commences a reduction in power in order to maintain vacuum. These efforts are initially successful in the maintenance of condenser vacuum but the condenser air in-leakage eventually increases to the point that requires the turbine to be tripped. After the turbine trip, air in-leakage continues to increase resulting in complete loss of vacuum and Group I isolation.

Shortly after the turbine is tripped, a loss of off-site power (LOOP) occurs due to collapse of the grid following the loss of Cooper’s generation. Both DGs fail to auto start following the loss of offsite power, but are both manually started by the crew.

The transient caused by the turbine trip, scram and Group 1 isolation results in a LOCA. The LOCA results in a rising containment temperature and pressure. EOP 1A and 3A are entered. HPCI fails to auto start, but can be manually initiated for RPV level control. The crew stabilizes level with HPCI and RCIC initially, then with low pressure systems.

Torus and drywell sprays are initiated by the crew. When the sprays are in service with a stable reactor water level utilizing low pressure systems, the scenario is terminated.

Cooper Nuclear Station June 2003 NRC Exam		**Shift Bus Duct Fans, Core Spray Line Break, Air Compressor Trip, Loss of Vacuum, LOOP & LOCA**	
Event No.	**Malf. No.**	**Type**	**Event Description**
N/A	ED20	N/A	Grid Instabilities
1	N/A	N	Shift the Bus Duct Cooling Fans
2	CS03B	I (RO)	“B” Core Spray Line Break
3	IA04C	C (BOP)	Air Compressor Trip
4	MC01	R	Lowering Condenser Vacuum, Power Reduction Following Lowering Vacuum
5	ED05 ED06	M	Loss of Offsite Power following turbine trip
6	DG06	C (BOP)	Failure of Both DGs to Start.
7	RR31	M	LOCA
8	HP01	C (RO)	HPCI Fails to Auto Start

CRITICAL TASKS:

1. Start the Diesel Generators prior to reactor water level reaching TAF and prior to exceeding PSP.
2. Initiate Drywell sprays prior to exceeding PSP.
3. Maintain RPV water level above -25" (corrected Fuel Zone).

Event Number		1
Event Description		Shift the Bus Duct Cooling Fans
Time	**Position**	**Applicant's Actions or Behavior**
5	CRS	Direct Bus Duct Cooling fans be shifted.
	BOP	• Obtain a copy of 2.2.53 • On PANEL C, place GEN BUS DUCT FAN 1B control switch to RUN. • Place GEN BUS DUCT FAN 1A to OFF. • Verify Annunciator C-3/F-2, MAIN GEN BUS DUCT NO AIR FLOW, alarms and clears. • Place GEN BUS DUCT FAN 1A from OFF to STDBY.

Event Number		2
Event Description		"B" Core Spray Line Break
Time	**Position**	**Applicant's Actions or Behavior**
8	RO	• Respond per 9-3-3/A5, CS B Break Detection. • Dispatch SO to read dp's locally.
	CRS	• Determine "B" Core Spray is inoperable • Refer to Tech Spec 3.5.1 • Determine REQUIRED ACTION A.1 applies (7 day LCO).

Event Number		3
Event Description		Air Compressor Trip
Time	**Position**	**Applicant's Actions or Behavior**
15	BOP	• Respond to and report alarm A-4/E-4, STATION AIR COMPRESSOR C TRIP. • Dispatch SO to standby compressor and verify it is operating properly. • Direct SO to reset compressor alarms and trips per Procedure 2.2.59. • If service or instrument air pressure lowering, enter Procedure 5.2AIR.
NOTE:		**The crew may not need to enter 5.2AIR depending on the timing of the actions above. If so, the following actions do not apply.**
	CRS	• Enter and direct the activities of 5.2AIR. • Direct operator perform actions of 5.2AIR
	BOP	Update the crew that 5.2AIR has scram actions.
	CRS	• Assign critical parameter monitoring. • Assign action points.
	BOP	Make following announcement twice:"All personnel using breathing equipment supplied by plant air shall move to an area with a clean atmosphere."
	CRS	Contact maintenance to investigate and repair "C" Air Compressor.

Event Number		4
Event Description		Lowering Condenser Vacuum, Power Reduction Following Lowering Vacuum
Time	**Position**	**Applicant's Actions or Behavior**
25	BOP	• Recognize rising SJAE flow. • Recognize and report degrading condenser vacuum.
	CRS	• Enter and direct the activities of 2.4VAC. • Direct an operator perform actions of 2.4VAC.
	BOP	Update crew that 2.4VAC has scram actions.
	CRS	• Direct critical parameter monitoring of condenser vacuum. • Direct action point for scram based on vacuum.
	BOP	• Enter 2.4VAC Attachment 1 • Ensure required number of circulating water pumps running per Procedure 2.2.3. • Check circulating water valve line-up is correct on Panel A. • Determine operating vacuum limitations
	CRS	Direct power reduction per Procedure 2.1.10 as necessary to maintain condenser vacuum > 23" Hg.
	RO	Reduce power per Procedure 2.1.10 as necessary to maintain condenser vacuum > 23" Hg.
	BOP	• Determine operating vacuum limitations • Check SJAE driving steam pressure is ~ 300 psig. • Ensure required condensate pumps are running to supply cooling water to SJAE inter and after condensers. • Ensure SJAE condenser inlet and outlet valves on Panel A are OPEN. • Ensure AR-MO-150, VACUUM BREAKER, is closed. • Ensure SJAE valve lineup is correct. • Check operation of Gland Sealing System per Procedure 2.2.75. • Ensure RWCU-MO-56 and RWCU-MO-57 are not open at same time.
	BOP	Report condenser vacuum recovering during power reduction.

Event Number		4
Event Description		Lowering Condenser Vacuum, Power Reduction Following Lowering Vacuum
Time	**Position**	**Applicant's Actions or Behavior**
	CRS	Brief crew on plans.
35	BOP	Report condenser vacuum is rapidly degrading.
	CRS	Direct a manual reactor scram be inserted.
	RO	Scram the reactor as directed.

Event Number		5
Event Description		Loss of Offsite Power following turbine trip
Time	**Position**	**Applicant's Actions or Behavior**
36	BOP	• Report loss of all off-site power. • Report failure of DG1 and DG2 to automatically start (see event 6). • ***Manually start DG1 and DG2.***
	CRS	• Direct operator perform actions of 5.3EMPWR. • Enter and direct the activities of EOP-1A • Direct RPV water level be restored and maintained +15" to +40". • Direct Reactor pressure be maintained 800 to 1000 psig.
	RO	• Report all control rods fully inserted. • Restore and maintain RPV water level +15" to +40". • Maintain reactor pressure 800 to 1000 psig.
	BOP	• Perform 5.3EMPWR. • Restore Service Water. • Restore REC cooling to the drywell. • Restore air compressors to service. • Monitor DG loading.
	RO	Recognize and report HPCI failure to automatically start (see event 8).

Event Number		6
Event Description		Failure of Both DGs to Start.
Time	**Position**	**Applicant's Actions or Behavior**
36	BOP	• Report failure of DG1 and DG2 to automatically start. • ***Manually start DG1 and DG2.***

Event Number		7
Event Description		LOCA
Time	**Position**	**Applicant's Actions or Behavior**
40	ALL	• Recognize rapid rise in primary containment pressure. • Recognize lowering trend on RPV water level.
	CRS	• Re-enter EOP-1A on Drywell pressure. • Enter and direct the activities of EOP-3A • Direct operator place Drywell FCUs in OVERRIDE. • Direct RHR be placed in torus spray mode.
	BOP	Place RHR in torus spray mode.
	CRS	• When torus spray is in service, verify PC level and DWSIL and ***direct Drywell Sprays be placed in service.*** • Brief crew on transition of injection sources as Rx pressure drops. • ***Direct RPV water level be maintained above -25" (corrected fuel zone)*** (preferably +15" to +40").
	RO	***Maintain RPV water level above -25" (corrected fuel zone)*** (preferably +15" to +40").
	BOP	***Initiate Drywell spray as directed.***
55	RO/BOP	Transition to low pressure ECCS for injection.

Event Number		8
Event Description		HPCI Fails to Auto Start
Time	**Position**	**Applicant's Actions or Behavior**
40	RO	• Recognize and report failure of HPCI to automatically start. • Manually start and operate HPCI.

Critical Tasks	SAT	UNSAT
1. Start the Diesel Generators prior to reactor water level reaching TAF and prior to exceeding PSP.		
2. Initiate Drywell sprays prior to exceeding PSP.		
3. Maintain RPV water level above -25" (corrected Fuel Zone).		

Booth Instructor Activities

<u>Event #1: Shift the Bus Duct Cooling Fans</u>

<u>ROLE PLAY</u>
IF asked, report the standby fan is ready to start and the running fan is still very noisy.

<u>Event #2: "B" Core Spray Line Break</u>

<u>ACTION:</u>
When directed by the Chief Examiner, activate **TRIGGER E2** to fail "B" Core Spray line.

<u>ROLE PLAY</u>
When asked to read the Core Spray dp's, report DPIS 43B reads +4.0 psid and DPIS 43A reads -3.5 psid (normal).

<u>Event #3: Air Compressor Trip</u>

<u>ACTION:</u>
When directed by the Chief Examiner, activate **TRIGGER E3** to trip "C" Air compressor.

<u>ROLE PLAY</u>:
When sent to check out "C" air compressor, report there is no obvious problem and there is no air leak.

<u>ROLE PLAY</u>:
When sent to check the now running "A" (previously standby) compressor, report it seems to be operating normally.

<u>ROLE PLAY</u>:
IF directed to reset "C" air compressor, report the high temperature trip will not reset.

<u>ACTION:</u>
If asked to change Tendamatic setting, use **REMOTE FUNCTION** IA01.

<u>NOTE:</u>
The "B" compressor may start on the tendamatic swap. If asked, you were slow moving

Booth Instructor Activities

the switch. The compressor started when you went through B-C-A.

Event #4: Lowering Condenser Vacuum, Power Reduction

ACTION:
When directed by the Chief Examiner, activate **TRIGGER E4** to start the condenser air leakage.

ROLE PLAY:
When Mwe has dropped by 10, call the control room as Load Dispatcher and ask them why they are lowering power.

ACTION:
Adjust **malfunction MC01** as necessary to maintain condenser vacuum degraded and require a power reduction, but above 23" (At 8%, vacuum rises very slowly).

ROLE PLAY:
IF the crew dispatches station operators to look for leaks, report back that you cannot find any.

ACTION:
When directed by Chief Examiner (it is desired to initiate the turbine trip), **increase malfunction MC01** to 100% ramped over 0 minutes.

Event #5: Loss of Offsite Power following turbine trip

NOTE:
The loss of off-site power automatically occurs after the turbine trip.

ROLE PLAY:
Contact the Control Room as Doniphan and report there are faults at CNS that prevent restoring power to the site.

ROLE PLAY:
If sent to check out the transformers, report there is no visible damage.

Booth Instructor Activities

Event #6: Failure of Both DGs to Start.

ROLE PLAY:
If sent to check out the DG's, report they are running normally.

Event #7: LOCA

ACTION:
When directed by the Chief Examiner, activate **TRIGGER E7** to initiate the LOCA.

ACTION:
When reactor pressure reaches 300 psig, increase **malfunction RR31A** to 10% ramped over 0 minutes.

Event #8: HPCI Fails to Auto Start

There are no prescripted actions or role play for this event.

TERMINATING CONDITIONS:

ACTION:
When the following conditions are reached, place the simulator in **FREEZE**:

1. The reactor has been depressurized.
2. RPV water level is under control.
3. Chief Examiner directs the termination of the scenario.

CAUTION: **Do not reset the simulator until directed by the Chief Examiner.**

SIMULATOR SET-UP

A. Materials required

None

B. Initialize the Simulator in IC-20 (Full Power, End of Cycle)

Batch File name A:\124819.

C. Change the Simulator conditions from those of the IC as follows:

1. Triggers

Number	File Name	Description
E5	None	trgset 5 "zloegcs3310[1]==1 .and. zloegcs3312[1]==1" 3310 and 3312 green lights

2. Malfunctions

Number	Title	Trigger	TD	Severity	Ramp	Initial
DG06A	Failue of DG1 to Automatically Start	A	0	N/A	N/A	N/A
DG06B	Failue of DG2 to Automatically Start	A	0	N/A	N/A	N/A
HP01	HPCI Fails to Automatically Start	A	0	N/A	N/A	N/A
ED20	Grid Instabilities	A	0	1	N/A	N/A
CS03B	“B” Core Spray Discharge Line Break (Inside Vessel)	E2	0	N/A	0	As Is
IA04C	“C” Air Compressor Trip	E3	0	N/A	N/A	N/A
MC01	Main Condenser Air In-Leakage	E4	0	15	3:00	As Is

Number	Title	Trigger	TD	Severity	Ramp	Initial
ED05	Loss of Power (Startup Transformer)	E5	10	N/A	N/A	N/A
ED06	Loss of Power (Emergency Transformer)	E5	30	N/A	N/A	N/A
RR31A	Reactor Recirculation Suction Loop Ruptue	E7	0	1.6	N/A	As Is

3. Remotes

Number	Title	Trigger	TD	Value	Ramp
	None				

4. Overrides

Instrument	Tag	Trigger	TD	Value	Ramp
"A" SLC pump green light (labeled as squib valve ready light)	ZLOSLCSWS1A[1]	A	N/A	OFF	N/A
"A" Squib Valve Ready Light	ZLOSLCSWS1A[3]	A	N/A	OFF	N/A
"A" SLC Pump Control Switch	ZDISLCSWS1A[1]	A	N/A	STOP	N/A
"A" Squib Continuity Alarm	RA:MUX09C072	A	N/A	ON	N/A

5. Panel Set-up

a. Place a Red Tag on "A" SLC pump control switch.

b. Advance all charts and alarm typers to clean paper.

c. Adjust APRM GAFs.

d. Balance the Main Generator regulator.

Turnover Information

A. Plant Status:

1. The plant is operating at approximately 100% power near the end of the operating cycle.

2. Rod Sequence Information:

Page:	2
Rod:	18-27
Notch:	26

3. Tech. Spec. Limitations in effect:

   Day 2 of 7 day LCO (3.1.7.A.1) for “A” SLC pump. The pump tripped during a surveillance. Maintenance is investigating.

4. Significant problems/abnormalities:

   - Record grid loads have resulted in slight instabilities. The load dispatcher is attempting to stabilize the grid.
   - The running bus duct cooling fan is making loud noises.

5. Evolutions/maintenance for the on-coming shift:

   - 5.3GRID has been entered and steps 4.1 & 4.2 have been completed.
   - Shift bus duct cooling fans as soon as the crew has the watch. The running fan was making loud noises last shift.
   - Continue investigation of “A” SLC pump.
   - Maintain maximum Megawatt output until further notice.

Cooper Nuclear Station June 2003 NRC Exam	Shift CRD Pumps, SW Pump Trip, Vessel Flange Leak, Loss of FW Heating, HPCI Leak, Fuel Failure, MSOT 2 Areas

Description:

The plant is initially operating at 100% power with SLC pump 1A out of service for corrective maintenance. After the crew takes the watch, B CRD pump is started and the A CRD pump is removed from service to support maintenance on the pump. The BOP operator must be the person who swaps the CRD pumps.

Following the shift of the CRD pumps, the A SW pump trips. The crew responds to the loss of the SW pump and evaluates Technical Specifications for the loss of the SW pump.

After the crew has addressed the Technical Specifications for the SW pumps, the Vessel Flange Seal Leakage annunciates. The crew responds per procedure 4.6.3.

Following the initial actions for the flange seal leak, a loss of feedwater heating occurs resulting in a power transient and small amount of fuel failure (delayed response). The crew reduces power in response to the power excursion cause by the loss of FW heating.

After power is stabilized and the APRM operability is assessed, a HPCI steam line break occurs and the HPCI isolation valves fail to close automatically. Manual attempts to close the valves are unsuccessful. The crew inserts a manual scram prior to exceeding Maximum Safe Operating Levels for temperature. The group 6 isolation fails to occur automatically and the crew manually isolates group 6 and starts SGT.

Following the scram, the fuel failure caused by the power excursion becomes evident, resulting in an off-site release of radiation if the release is not isolated (group 6).

Emergency Depressurization is required due to the primary system discharging into the secondary containment, resulting in 2 areas exceeding Maximum Safe Operating temperature.

The scenario ends when the group 6 isolation is actuated by the operators, RPV Depressurization has taken place and RPV level is being maintained.

Cooper Nuclear Station June 2003 NRC Exam		**Shift CRD Pumps, SW Pump Trip, Vessel Flange Leak, Loss of FW Heating, HPCI Leak, Fuel Failure, MSOT 2 Areas**	
Event No.	**Malf. No.**	**Type**	**Event Description**
1	N/A	N (BOP)	Shift CRD pumps
2	SW01C	C (BOP)	SW Pump Trip
3	RR21	I (RO)	Vessel Head Inner Seal Leakage
4	various	C (RO/BOP)	Loss of Feedwater Heating
5	N/A	R	Reduce Power Due to Loss of Feedwater Heating
6	HP06	M	HPCI Steam Leak
7	CR01	C (BOP)	Fuel Failure
8	RP05	C (BOP)	Group 6 isolation failure

Critical Tasks:

1. Scram prior to emergency depressurization.
2. Emergency Depressurize within 10 minutes of Exceeding Maximum Safe Operating Temperature in two areas.
3. Manually isolate reactor building release prior to reactor building vent radiation monitor exceeding 49 mrem.

Event Number		1
Event Description		Shift CRD pumps
Time	**Position**	**Applicant's Actions or Behavior**
5	CRS	Direct operator to swap CRD pumps.
	RO	! Establish communications between Control Room and CRD pump area. ! Ensure » CRD-130, CRD PUMP 1A SUCTION (R-881-SE Quad), is open. » CRD-13, CRD PUMP 1A MINIMUM FLOW (R-881-SE Quad), is open. » CRD-11, CRD PUMP 1A DISCHARGE (R-881-SE Quad), is open. » Pump and motor oil levels normal. » Pump casing and suction filter vented. ! At Panel 9-5, balance CRD-FIC-301, CRD FLOW CONTROL, and place to MANUAL. ! Start "A" CRD pump. ! Check locally for normal pump operation. ! Slowly adjust manual control on CRD-FC-301 to obtain flow of 50 gpm. ! Balance CRD-FC-301 and place to BAL. ! At Panel 9-5, check charging water pressure and drive water ΔP, and adjust if needed.

Event Number		2
Event Description		"C" SW Pump Trip
Time	**Position**	**Applicant's Actions or Behavior**
10	BOP	! Report and respond per alarm A-4/B-7, Service Water Pump C Trip ! Maintain SW System pressure > 38 psig on SW-PI-2715A as follows: » Place available non-running SW pump MODE SELECTOR switch(es) in MAN, as necessary. » Start available SW pump(s), as necessary. ! Ensure MODE SELECTOR switches aligned per Procedure 2.2.71. ! Dispatch Station Operator(s) to investigate the pump trip.
	CRS	! Enter and direct the activities of 5.2SW. ! Direct operator perform subsequent actions of 5.2SW.
	BOP	Update crew on scram actions in 5.2SW.
	CRS	! Assign critical parameters. ! Assign action points.
	BOP	! Maintain system pressure > 38 psig in both loops (SW-PI-2715A(B)) as follows: » Place available SW pump MODE SELECTOR switch(es) to MAN, as necessary (done). » Start available SW pump(s), as necessary (done). » If pressure still < 38 psig, close SW-MO-116, SCREENWASH SUPPLY. » If pressure is > 38 psig, ensure MO-36 and/or MO-37 are open. » Ensure MODE SELECTOR switch(es) aligned per Procedure 2.2.71 (done) ! Return system to a normal line-up per Procedure 2.2.71..

Event Number		2
Event Description		“C” SW Pump Trip
Time	**Position**	**Applicant’s Actions or Behavior**
	CRS	! Declare “C” SW pump inoperable. ! Determine REQUIRED ACTION 3.7.2.A.1 (30 day LCO) applies. ! Determine do NOT need to cascade to specifications 3.8.1 or 3.4.7.

Event Number		3
Event Description		Vessel Head Inner Seal Leakage
Time	**Position**	**Applicant's Actions or Behavior**
20	BOP	! Respond per alarm 9-4-1/F-5, VESSEL FLANGE SEAL LEAK ! Enter 4.6.3, Reactor Vessel Top Head Flange Leak Detection. !
	CRS	Station personnel at location of valves to ensure timely restoration of valve, when required by Control Room Operator, and require continuous communications with Control Room Operator.
27		! Determine leak rate by: » Remove seal and open PC-559, 560, 565 & 566. » Place REACTOR FLANGE LEAKOFF switch (Panel 9-4) to OPEN and verify following: - INLET NBI-736AV closes. - DRAIN NBI-737AV opens. - Annunciator 9-4-1/F-5 clears. - NBI-PI-101 (R-931-NW LR 25-5) indicates 0 pressure. » Place REACTOR FLANGE LEAKOFF switch (Panel 9-4) to CLOSE and verify following: - INLET NBI-736AV opens. - DRAIN NBI-737AV closes. » Close and seal PC-559, 560, 565, 566. ! Ensure appropriate entries are made in PC Manual Isolation Valve and Cap Log.
	CRS	! If vessel seal leakage alarms come in frequently, record time interval between each drainage and alarm and determine rate of leakage. ! Monitor equipment sump flow rate. ! Evaluate leakage rate limits. ! Determine no limit is exceeded at this time.

Event Number		4
Event Description		Loss of Feedwater Heating
Time	**Position**	**Applicant's Actions or Behavior**
30	RO	Report rising reactor power.
	CRS	MAY enter 2.4RXPWR until reason for power rise is identified.
	BOP	! Report and respond to alarm **A-2/C-4** HEATER LOW LEVEL alarm for B3, B2 and B1 heaters. ! Direct SO to check affected heater level locally. ! Check applicable heater-to-heater and heater-to-condenser valves (CD-AO-LCV) are fully closed and turbine-to-heater valves (ES-AO-NRV) are open. ! Check PMIS point NSSRP617, for a loss of feedwater heating and enter Procedure 2.4EX-STM, if required.
	RO/BOP	Report lowering feedwater temperature.
	CRS	! Enters 2.4EX-STM ! Determines in the loss of FW Heating Region ! Required to restore within 2 hours or reduce power to < 25%
	RO	! Reduce power with recirculation flow per Procedure 2.1.10, as necessary, to restore power to level it was before feedwater temperature lowered (see event 5) ! Insert control rods as necessary per Procedure 10.13 to maintain rod line < 120%.
	BOP	! Directs SO to check out FW Heaters ! Determines APRM GAFs are out of specification. ! Determines APRMs are not allowed to be adjusted per procedure 10.1 (due to being outside Att. 4 graph).
	CRS	! Declare APRMs with GAFs out of specification inoperable. ! Enter appropriate condition of LCO 3.3.1.1. ! Enter appropriate condition of TLCO 3.3.1

Event Number		5
Event Description		Reduce Power Due to Loss of Feedwater Heating
Time	**Position**	**Applicant's Actions or Behavior**
30	CRS	! Direct power be reduced with recirculation flow per Procedure 2.1.10, as necessary, to restore power to level it was before feedwater temperature lowered ! Direct control rods be inserted as necessary per Procedure 10.13 to maintain rod line < 120%.
	RO	! Reduce power with recirculation flow per Procedure 2.1.10, as necessary, to restore power to level it was before feedwater temperature lowered ! Insert control rods as necessary per Procedure 10.13 to maintain rod line < 120%.

Event Number		6
Event Description		HPCI Steam Leak
Time	**Position**	**Applicant's Actions or Behavior**
40	BOP	Report Rx Bldg ARMs alarming.
	CRS	! Enter and direct the activities of EOP-5A, Secondary Containment Control. ! Direct all quad coolers be started. ! Direct verification of sump pump operation. ! Direct area temperatures be monitored. ! Direct HPCI be isolated
	BOP	! Start quad coolers (including REC flow) ! Verify sump pump operation. ! Monitor area temperatures. ! Attempt to isolate HPCI, report valves will not close. ! Reports temperatures from Pnl 9-21 or PMIS.
	CRS	Monitor for any secondary containment parameter reaches its maximum safe operating value: • Temperatures (TABLE 9) • Radiation (TABLE 10) • Water levels (TABLE 11)
48	BOP	Report alarm 9-3-1/E-10, Area High Temp.
53	CRS	! Re-enters EOP-5A ! Determine a manual scram is required BEFORE a Maximum Safe Operating Value has been exceeded (already complete). ! Enter EOP-1A ! Direct a manual scram be inserted (***prior to depressurization***).
	RO	! Scram the reactor as directed (***prior to depressurization***). ! Report all control rods fully inserted.
	CRS	! Direct RPV pressure be maintained 800 to 1000 psig. ! Direct RPV water level be maintained +15" to +40".

Event Number		6
Event Description		HPCI Steam Leak
Time	**Position**	**Applicant's Actions or Behavior**
	BOP/RO	! Maintain RPV pressure 800 to 1000 psig. ! Maintain RPV water level +15" to +40".
60	CRS	! May determine anticipation of Emergency Depressurization is required. ! May direct operator fully open Main Turbine Bypass valves.
	BOP	Fully open Main Turbine Bypass valves if directed by CRS.
NOTE:		**If Main Turbine Bypass valves are opened early enough, temperatures may not exceed MSOT in 2 areas. If so, the following actions do not apply and the preceding steps satisfy the Critical Task.**
	CRS	***When 2 areas exceed Maximum Safe Operating Values, direct an Emergency Depressurization be performed.*** ! Verify PC level is > 6'. ! Direct 6 SRVs be opened.
	BOP	***Place 6 SRV control switches in OPEN.***
	CRS	! When torus temperature reaches 95˚F, enter and direct the activities of EOP-3A. ! Direct torus cooling be placed in service.

Event Number		7
Event Description		Fuel Failure
Time	**Position**	**Applicant's Actions or Behavior**
58	BOP	! Respond per annunciator 9-4-1/C-5, Offgas Hi Rad ! Inform Chemistry to sample reactor coolant. ! Enter 2.4OG and 5.2FUEL.
	BOP	! Reports alarm **9-3-1/A-9**, REACTOR BLDG HIGH RAD ! Reports numerous Rx Bldg ARMs alarming and levels from Pnl 9-11. ! Notify Plant personnel to evacuate Rx Bldg. ! Enter Procedure 5.1RAD.

Event Number		8
Event Description		Group 6 isolation failure
Time	**Position**	**Applicant's Actions or Behavior**
53	ALL	Recognize Group 2 and Group 6 isolation failure.
	CRS	***Direct operator manually insert Group 2 and Group 6 isolations before exceeding 49 mr/hr on reactor building vent rad monitors.***
	BOP	***Manually insert Group 2 and Group 6 isolations before exceeding 49 mr/hr on reactor building vent rad monitors.***

Critical Tasks	SAT	UNSAT
1. Scram prior to emergency depressurization.		
2. Emergency Depressurize within 10 minutes of Exceeding Maximum Safe Operating Temperature in two areas **OR** rapidly depressurize the RPV to prevent 2 areas from exceeding Maximum Safe Operating Values.		
3. Manually isolate reactor building release prior to reactor building vent radiation monitor exceeding 49 mrem.		

Booth Instructor Activities

Event #1: Shift CRD pumps

ROLE PLAY:
When asked for status:
- CRD-130, CRD PUMP 1A SUCTION (R-881-SE Quad), is open.
- CRD-13, CRD PUMP 1A MINIMUM FLOW (R-881-SE Quad), is open.
- CRD-11, CRD PUMP 1A DISCHARGE (R-881-SE Quad), is open.
- Pump and motor oil levels normal.
- Pump casing and suction filter vented.

ROLE PLAY:
When asked for status of “A” CRD pump, report the pump is operating normally.

Event #2: “C” SW Pump Trip

ACTION:
When directed by the Chief Examiner, activate **TRIGGER E2** to trip the “C” SW pump.

ROLE PLAY:
When asked to investigate the “C” SW pump, report the motor is very hot.

ROLE PLAY:
When asked to investigate the “C” SW pump motor at the switchgear, report the 51 device has tripped for the pump.

Booth Instructor Activities

Event #3: Vessel Head Inner Seal Leakage

NOTE:
It takes ~ 10 minutes before the alarm comes in. The malfunction is active when the batch file is loaded. The annunciator is overridden off in the setup. TRIGGER E3 will delete the override and restore the malfunction to normal operation. This is necessary to avoid a 10 minute wait.

ACTION:
When directed by the Chief Examiner, activate **TRIGGER E3** to cause the Seal to leak.

ROLE PLAY:
When asked to establish continuous communications, perform action as requested.

NOTE:
It will take all 3 station operators to support this action.

ROLE PLAY/ACTION:
When asked to remove seals and open valves locally, activate **TRIGGER 20** to unisolate air to the flange seal leakoff valves.

ROLE PLAY:
When asked for pressure on NBI-PI-101, report 0 psig.

ROLE PLAY/ACTION:
When asked to close the air supply, modify **Remote Function RR18** to CLOSE.

Booth Instructor Activities

Event #4: Loss of Feedwater Heating

ROLE PLAY:
When directed by the Chief Examiner, activate **TRIGGER E4** to simulate Extraction Steam Dump Valves DV-3, 4, 7, 8, & 10 opening.

ROLE PLAY
As SO report Lights out for DV-3, 4, 7, 8, & 10 and NRV 2, 4 ,7, 8, 11,& 12. **IF** asked, extraction Steam panel LR-102 supply fuse for CCP-2B smells like burnt insulation (cannot tell status visually by design).

ROLE PLAY
IF asked, report heaters B1, B2 and B3 have a very low level.

ROLE PLAY
IF told to go into the Heater Bay, wait 2 min and report Ex Steam Dump Valves DV-3, DV-4, DV-7, DV-8 and DV-10 are **OPEN**. (Override SW Htr Bay Door alarms as appropriate.)

ROLE PLAY
IF contacted as I&C or Electrician, report fuse blown due to hard ground on circuit 2 of CCP-2B. You are investigating the ground.

ROLE PLAY
IF Reactor Engineer contacted, recommend following rod insertion sequence in book if rod line needs to be adjusted.

Event #5: Reduce Power Due to Loss of Feedwater Heating

ROLE PLAY
IF Reactor Engineer contacted, recommend following rod insertion sequence in book if rod line needs to be adjusted.

Booth Instructor Activities

Event #6: HPCI Steam Leak

NOTE: ***It will take ~ 6 minutes after malfunction activation until the high temperature alarm is received.***

ROLE PLAY:

When directed by the Chief Examiner, activate **TRIGGER E6** to start the HPCI steam leak.

ROLE PLAY:

When the first high temperature alarm is received (or if you have been sent to investigate), call the control room and report steam coming out of the HPCI room.

NOTE:

It takes ~ 8 minutes from the high temperature alarm until the first maximum safe temperature is reached (if they do not depressurize).

ACTION

If requested to isolate HPCI from the ASD room, activate **TRIGGER E18.** Place HPCI-MO-15/16 Isolate switch to ISOL. Call the control room and report the valves will not close from the ASD room.

NOTE:

The crew should attempt to close MO-15 from MCC-R due to the steam leak rather than MO-16 which is in the Quad

ROLE PLAY

If asked to close HPCI MO-16 manually (local), wait 2 minutes, then report steam in the Rx Bldg prohibits access to the valve.

Event #7: Fuel Failure

NOTE:

This failure automatically inserts itself on the reactor scram.

Booth Instructor Activities

Event #8: Group 6 isolation failure

There are no prescripted actions or role plays for this event.

TERMINATING CONDITIONS:

ACTION:
When the following conditions are reached, place the simulator in **FREEZE**:

1. The reactor has been depressurized.
2. RPV water level is under control.
3. Chief Examiner directs the termination of the scenario.

CAUTION: **Do not reset the simulator until directed by the Chief Examiner.**

SIMULATOR SET-UP

A. Materials required

None

B. Initialize the Simulator in IC-20 (Full Power, End of Cycle)

Batch File name A:\124820.

C. Change the Simulator conditions from those of the IC as follows:

1. Triggers

Number	File Name	Description
E3	None	trg 3 "DOR RA:MUX17C051" Delete override on Vessel Flange Leak annunciator when TRIGGER 3 is activated.
E7	None	trgset 7 "zlorpsds1a==0 .and. zlorpsds1b==0" "A1" scram light off and "B1" scram light off

2. Malfunctions

Number	Title	Trigger	TD	Severity	Ramp	Initial
RP05	Group 2 Isolation Failure	A	0	N/A	N/A	N/A
RP07	Group 6 Hi Rad Isolation Failure	A	0	N/A	N/A	N/A
HP09	HPCI Failure to Auto Isolate	A	0	N/A	N/A	N/A
RR21	Vessel Head Inner Seal Leakage	A	0	100	0	As Is
SW01C	Service Water Pump Trip	E2	0	N/A	N/A	N/A
HP06	HPCI Steam Line Break	E6	0	6	25:00	0.5
CR01	Fuel Cladding Failure	E7	0	50	2:00	0

3. Remotes

Number	Title	Trigger	TD	Value	Ramp
FW22	FW Heater Bypass Valve	E4	0	Open	N/A
FW25	FW Heater Bypass Valve	E4	0	Open	N/A
RR18	Isolate Air to Flange Leakoff Valves	E20	0	Open	N/A

4. Overrides

Instrument	Tag	Trigger	TD	Value	Ramp
“A” SLC pump green light (labeled as squib valve ready light)	ZLOSLCSWS1A[1]	A	N/A	OFF	N/A
“A” Squib Valve Ready Light	ZLOSLCSWS1A[3]	A	N/A	OFF	N/A
“A” SLC Pump Control Switch	ZDISLCSWS1A[1]	A	N/A	STOP	N/A
“A” Squib Continuity Alarm	RA:MUX09C072	A	N/A	ON	N/A
Stm Supp Outbd Isol Vlv MO-15	ZDIHPCISWS1[2]	A	0	OPEN	N/A
Stm Supp Inbd Isol Vlv MO-16	ZDIHPCISWS2[2]	A	0	OPEN	N/A
HPCI Manual Isolation PB	ZDIHPCISWS32	A	0	OFF	N/A
Annunciator 9-4-1/F-5, VESSEL FLANGE SEAL LEAK	RA:MUX17C051	A	0	OFF	N/A
B-3 Heater Low Level alarm	RA:MUX05C027	E4	01:20	ON	N/A
B-2 Heater Low Level alarm	RA:MUX05C026	E4	01:35	ON	N/A
B-1 Heater Low Level alarm	RA:MUX05C025	E4	01:55	ON	N/A
ASD Room Outer Door	RA:MUX15C035	E18	0	ON	N/A

5. Panel Set-up

a. Place a Red Tag on “A” SLC pump control switch.

b. Advance all charts and alarm typers to clean paper.

c. Adjust APRM GAFs.

d. Balance the Main Generator regulator.

Turnover Information

A. Plant Status:

1. The plant is operating at approximately 100% power near the end of the operating cycle.

2. Rod Sequence Information:

Page:	2
Rod:	18-27
Notch:	26

3. Tech. Spec. Limitations in effect:

   Day 2 of 7 day LCO (3.1.7.A.1) for “A” SLC pump. The pump tripped during a surveillance. Maintenance is investigating.

4. Significant problems/abnormalities:

   None.

5. Evolutions/maintenance for the on-coming shift:

   - The BOP operator is to shift CRD pumps as soon as the crew takes the watch.
   - Continue investigation of “A” SLC pump.

Cooper Nuclear Station June 2003 NRC Exam	Manual Scram Test, Single Rod Scram, SDV Drain Valve Failure, TEC Pump Trip, High Main Turbine Vibration, Stop and Prevent ATWS

Description:

The plant is operating at 90% power with SLC pump 1A is out of service for corrective maintenance.

After the crew takes the watch, they perform surveillance 6.1RPS.301 "Manual Scram Functional Test (DIV 1)." When the half scram is inserted for the surveillance, control rod 22-23 scrams due to a blown fuse for the 118 valve (if fuses checked prior to the test, all fuses indicate continuity). The crew responds per 2.4CRD. The reactor engineer recommends that the crew reduce reactor power to 70% to recover the control rod.

After the crew has addressed the Technical Specifications for the control rod, one of the drain valves for the South SDV fails.

After the crew responds to drain valve failure, an operating TEC pump trips requiring the crew to start the standby pump.

Following the loss of the TEC pump, a main turbine bearing failure occurs with associated high vibration. Eventually the vibration reaches a level that requires a scram and turbine trip.

When the manual scram is attempted very little rod movement occurs. Crew attempts to insert the control rods with ARI also fail. The crew will be required to stop and prevent injection to suppress power oscillations. After the crew stops and prevents injection, a steam leak develops in the steam tunnel resulting in a high steam tunnel temperature, the crew must manually close the MSIVs due to a failure of the automatic group 1 isolation.

RPV pressure control is established with SRVs and control rods are inserted using Alternate Rod Insertion methods IAW ESP 5.8.3.

The scenario ends when all control rods are inserted.

Cooper Nuclear Station June 2003 NRC Exam		**Manual Scram Test, Single Rod Scram, SDV Drain Valve Failure, TEC Pump Trip, High Main Turbine Vibration, Stop and Prevent ATWS**	
Event No.	**Malf. No.**	**Type**	**Event Description**
1	N/A	N	Manual Scram Surveillance
2	RD14	C (RO)	Single Rod Scram
3	N/A	R	Reduce Reactor Power to Recover Rod
4	RD01A	C (RO)	South Scram Discharge Volume Drain Valve Failure
5	SW07	C (BOP)	TEC Pump Trip / Start Standby Pump
6	TU03	C (BOP)	Turbine Bearing Failure/High Vibration
7	RD02, Various	M	Turbine Trip, ATWS
8	MS03	C (BOP)	Steam Leak
9	RP04	I (RO)	Group 1 Isolation Failure

Critical Tasks:

1. Fully insert all control rods.
2. Close MSIVs prior to 2 secondary containment areas exceeding maximum operating temperature.
3. Inhibit ADS prior to exceeding cooldown rate limit.
4. Stop and prevent injection except for CRD, RCIC and boron injection prior to HCTL.

Critical Tasks are underlined in ***bold italics*** in the scenario.

<table>
<tr><th colspan="2">Event Number</th><td>1</td></tr>
<tr><th colspan="2">Event Description</th><td>Manual Scram Surveillance</td></tr>
<tr><th>Time</th><th>Position</th><th>Applicant’s Actions or Behavior</th></tr>
<tr><td>5</td><td>CRS</td><td>Direct RO perform 6.1RPS.301 “Manual Scram Functional Test (DIV 1)”</td></tr>
<tr><td></td><td>RO</td><td>♦ Ensure SCRAM INDICATIONS GROUP A and GROUP B lights (Panel 9-5) <u>or</u> all SCRAM GROUP lights (Panels 9-15 and 9-17) are on.
♦ Press RX SCRAM CH A button (Panel 9-5)
♦ Report alarm 9-5-1/C-4 Rod Drift.
♦ Report rod 30-23 has scrammed.
♦ Stop the surveillance</td></tr>
<tr><td colspan="3">NOTE: The crew may elect to NOT resume the surveillance. If so, the following steps do not apply and procedure 2.1.5 would provide appropriate guidance for resetting the ½ scram.</td></tr>
<tr><td></td><td>RO</td><td>Verify:

♦ 9-5-2/A-1, RX SCRAM CHANNEL A, alarms
♦ Alarm CRT displays 2650 RX SCRAM (MANUAL) CHAN A3 TRIP.
♦ SCRAM INDICATIONS GROUP A lights turn off.
♦ Light in RX SCRAM CH A button turns on.
♦ PMIS indicates D528 MANUAL SCRAM CHANNEL A TRIP.</td></tr>
<tr><td></td><td>RO</td><td>Reset scram and check:

♦ Annunciator 9-5-2/A-1 clears.
♦ Alarm CRT displays 2650 RESET.
♦ SCRAM INDICATIONS GROUP A and GROUP B lights (Panel 9-5) are on.
♦ Light in RX SCRAM CH A turns off.
♦ PMIS indicates D528 RSET.</td></tr>
<tr><td></td><td></td><td></td></tr>
</table>

Event Number		2
Event Description		Single Rod Scram
Time	**Position**	**Applicant's Actions or Behavior**
NOTE:		**The CRS may declare rod 30-23 inoperable prior to replacing the fuse. If he does, he should take action per LCO 3.1.3.**
7	CRS	♦ Direct the surveillance be stopped. ♦ Enter and direct the activities of 2.4CRD. ♦ Direct operator perform subsequent actions of 2.4CRD.
	RO	♦ Inform crew of scram actions contained in 2.4CRD. ♦ Troubleshoot per 2.2.8 (no new guidance). ♦ Contact Reactor Engineer and system engineer to investigate the single rod scram.

Event Number		3
Event Description		Reduce Reactor Power to Recover Rod
Time	**Position**	**Applicant's Actions or Behavior**
15	CRS	Direct power reduction at ≤ 10 Mwe/min to 70% power.
	RO	Reduce power at ≤ 10 Mwe/min to 70%.

Event Number		4
Event Description		South Scram Discharge Volume Drain Valve Failure
Time	**Position**	**Applicant's Actions or Behavior**
22	RO	Respond per annunciator **9-5-1/G-5**, SOUTH SDIV NOT DRAINED ♦ Check CRD-AO-CV33 and CRD-AO-CV35, SOUTH SDV DRAIN VLV AO 33/35, are open. ♦ Dispatch SO to check locally for HCU scram valve leakage. ♦ Check on Panel 9-5 for scram valve open indication. ♦ Check for drain line blockage. ♦ Report drain valve 33 is closed.
	CRS	♦ Direct maintenance investigate the drain valve failure. ♦ Declare the SDV Drain Valve inoperable. ♦ Determine Technical Specification LCO 3.1.8, REQUIRED ACTION A.1 (7 day LCO) applies. ♦ Determine inop valve may be opened periodically under administrative control to drain the SDV (NOTE 2)

Event Number		5
Event Description		TEC Pump Trip / Start Standby Pump
Time	**Position**	**Applicant's Actions or Behavior**
30	BOP	♦ Respond per alarm M-2/C-5 TEC PUMP C TRIP and M-2/A-6, TEC SYSTEM LOW PRESSURE ♦ Start additional TEC pumps if available. ♦ Monitor TEC System pressure.
	CRS	♦ If no high temperature alarms are received on any TEC cooled component, determine entry into 2.4TEC is NOT required. ♦ Contact maintenance to investigate and repair "C" TEC pump.
	BOP	♦ Dispatch Station Operator(s) to investigate the trip of "C" TEC pump. ♦ Refer to 2.2.76 and report to crew that precaution "TEC Pumps A and B should <u>not</u> be operated together, as loss of 480V Bus 1A will cause a loss of both pumps." cannot be met.

Event Number		6
Event Description		Turbine Bearing Failure/High Vibration
Time	**Position**	**Applicant's Actions or Behavior**
38	BOP	♦ Respond per alarm B-1/C-2, TG ROTOR HI VIBRATION ♦ Validate alarm by checking other bearings, and observing locally. ♦ Update the crew on the high vibration scram action in the Annunciator procedure. ♦ Enter 2.4TURB
	CRS	♦ Enter and direct actions of 2.4TURB. ♦ Establish turbine vibration as critical parameter. ♦ Set action point (< 14 mils) for scram action. ♦ Direct operator perform subsequent actions of 2.4TURB. ♦ When red DANGER LED is on, direct power be slowly lowered per Procedure 2.1.10 while observing vibration. ♦ When vibrations continue to rise while lowering power, direct power reduction be stopped.
	RO	Lower power as directed by CRS.
	BOP	♦ Monitor and report turbine vibration. ♦ Contact Turbine Engineering Group for data analysis ♦ Report turbine vibration when it exceeds the action point.
	CRS	♦ Direct the reactor be scrammed. ♦ Direct the main turbine be tripped.
	RO	♦ Scram the reactor. ♦ Report an ATWS.
47	BOP	Trip the main turbine.

Event Number		7
Event Description		Turbine Trip, ATWS
Time	**Position**	**Applicant's Actions or Behavior**
47	CRS	♦ Enter EOP-1A. ♦ Exit EOP-1A, enter EOP-6A and EOP-7A. ♦ Direct reactor mode switch be placed in SHUTDOWN. ♦ Direct ARI be initiated ♦ Direct Recirculation flow be reduced to minimum. ♦ Direct Recirculation pumps be tripped. ♦ ***<u>Direct ADS be inhibited</u>***. ♦ Direct SLC be initiated. ♦ Direct ARI be reset. ♦ ***<u>Direct control rods be inserted by 5.8.3</u>***.
	RO	♦ Attempt to initiate ARI, report initiation failure. ♦ Reduce Recirculation flow to minimum ♦ Trip Recirculation pumps (one at a time) ♦ Initiate SLC, report initial tank level. ♦ Reset ARI. ♦ ***<u>Insert control rods per 5.8.3.</u>***
	BOP	♦ ***<u>Inhibit ADS (prior to exceeding the cooldown rate limit).</u>***
	CRS	♦ Direct reactor pressure be maintained 800 to 1000 psig ♦ Direct low level Group 1 isolation be defeated. ♦ ***<u>Direct injection into the RPV be stopped and prevented except RCIC, CRD and boron injection (prior to exceeding HCTL).</u>***
	BOP	♦ Maintain reactor pressure 800 to 1000 psig with bypass valves and SRVs. ♦ Defeat low level Group 1 isolation if directed. ♦ ***<u>Stop and prevent injection into the RPV except CRD, RCIC and boron injection (prior to exceeding HCTL).</u>***

Event Number		7
Event Description		Turbine Trip, ATWS
Time	**Position**	**Applicant's Actions or Behavior**
	RO	♦ Direct EOP PTM 61 and 62 be installed. ♦ Defeat automatic reactor scrams. ♦ Reset the scram. ♦ Start second CRD pump. ♦ Place CRD FCV to MAN. ♦ Insert control rods with RMCS.
54	CRS	When RPV water level reaches +100" (corrected FZ), direct RPV water level be maintained in a band somewhere between -25" (corrected FZ) and +100" (corrected FZ).
	BOP	Maintain RPV water level in band directed by CRS.
	RO	♦ When SDV has drained, insert manual scram. ♦ Report control rod movement. ♦ Reset scram. ♦ When SDV has drained a second time, insert manual scram. ♦ Report all control rods fully inserted.
	CRS	♦ Direct SLC injection be secured. ♦ Exit EOP-6A and EOP-7A, enter EOP-1A. ♦ Direct RPV water level be restored and maintained +15" to +40"
	RO	Secure SLC injection.
	BOP	Restore and maintain RPV water level +15" to +40".

Event Number	8	
Event Description	Steam Leak	
Time	**Position**	**Applicant's Actions or Behavior**
52	BOP	Respond to and report leak detection alarm. Monitor area temperatures on PMIS or on back panels.
	CRS	♦ When Max normal water level exceeded in secondary containment, enter and direct the activities of EOP-5A ♦ Direct sump pump operation be verified. ♦ Direct all quad coolers be started.
	BOP	♦ Verify sump pump operation. ♦ Start all quad coolers.
	CRS	Once leak location has been identified, determine if MSIVs are required to support EOPs (unless temp is already > 195˚F, then direct closing MSIVs).

Event Number		9
Event Description		Group 1 Isolation Failure
Time	**Position**	**Applicant's Actions or Behavior**
53	CRS	***Direct the MSIVs be closed when the group 1 isolation setpoint has been exceeded (prior to maximum safe temperature in 2 areas).***
	BOP	***Close the MSIVs be closed when the group 1 isolation setpoint has been exceeded (prior to maximum safe temperature in 2 areas).***

Critical Tasks	SAT	UNSAT
1. Fully insert all control rods.		
2. Close MSIVs prior to 2 secondary containment areas exceeding maximum operating temperature.		
3. Inhibit ADS prior to exceeding cooldown rate limit.		
4. Stop and prevent injection except for CRD, RCIC and boron injection prior to HCTL.		

Booth Instructor Activities

<u>Event #1: Manual Scram Surveillance</u>

<u>ROLE PLAY:</u>
If asked prior to the surveillance, report all fuses at the HCUs for the 117 and 118 valves have been checked and have continuity.

<u>Event #2: Single Rod Scram</u> ***NOTE:*** ***This event is tied to a conditional trigger that goes active on the ½ scram.***

<u>ROLE PLAY:</u>
When asked to investigate as Rx Bldg Station Operator, report fuse 118 is blown on HCU for rod 30-23.

<u>ACTION:</u>
When the ½ scram is reset OR the 118 fuse has been replaced, **delete malfunction RD14**.

<u>ROLE PLAY:</u>
When contacted as System Engineer, recommend replacing the 118 fuse.

<u>ROLE PLAY:</u>
If asked to replace the 118 fuse, wait 2 minutes,and report to the control room that you have replaced the fuse.

<u>ROLE PLAY:</u>
When contacted as Reactor Engineer, request power be reduced to 70% to recover the control rod. Once power has been reduced to 80%, the control rod may be single notch withdrawn to notch 12, then continuous to its original position. You will be up in a few minutes with the rod movement sheet.

<u>Event #3: Reduce Reactor Power to Recover Rod</u>

No prescripted actions or role play.

Booth Instructor Activities

Event #4: South Scram Discharge Volume Drain Valve Failure

ACTION:
When directed by the Chief Examiner, activate **TRIGGER E4** to fail the South SDV drain valve (V-33).

NOTE:
There is a 3 minute time delay until the "Not Drained" annunciator is received.

ROLE PLAY:
If asked to investigate the valve, report that the "33" valve has failed closed. There is no obvious reason why.

Event #5: "C" TEC Pump Trip / Start Standby Pump ("B")

NOTE: This event is tied to a conditional trigger that activates thehydraulic ATWS.

ACTION:
When directed by the Chief Examiner, activate **TRIGGER E5** to trip the "C" TEC pump.

ROLE PLAY:
When contacted as Station Operator to investigate, report relay TEC-REL-TECPC(74) is tripped for "C" TEC pump. The motor is hot to the touch.

Event #6: Turbine Bearing Failure/High Vibration

NOTE: The high vibration alarm comes in at ~ 43% severity.

ACTION:
When directed by the Chief Examiner, insert **TRIGGER E6** to activate the bearing high vibration malfunction.

Booth Instructor Activities

Event #7: Turbine Trip, ATWS

NOTE: The ATWS is tied to a conditional trigger that goes active when "C" TEC pump trips.

ACTION:
After the scram, delete the overrides on the south SDV drain valve.

ROLE PLAY:
When directed to install EOP PTM 61 and 62, wait 2 minutes, then **INSERT REMOTE FUNCTION RD18**, EOP PTMs 61 & 62 at REMOVE.

ACTION:
After the scram has been reset (the first time), reduce **malfunction RD02** to 90%.

ACTION:
When the reactor scram has been reset the second time, **DELETE** malfunction RD02.

Event #8: Steam Leak

ACTION:
When directed by the Chief Examiner, activate **TRIGGER E8** to initiate the steam leak in the steam tunnel.

Event #9: Group 1 Isolation Failure

No prescripted actions or role play.

TERMINATING CONDITIONS:

ACTION:
When the following conditions are reached, place the simulator in **FREEZE**:

1. All control rods inserted.
2. RPV water level is under control.
3. Chief Examiner directs the termination of the scenario.

CAUTION: **Do not reset the simulator until directed by the Chief Examiner.**

SIMULATOR SET-UP

A. Materials required

None

B. Initialize the Simulator in IC-20 (Full Power, End of Cycle)

Batch File name A:\124821.

C. Change the Simulator conditions from those of the IC as follows:

1. Triggers

Number	File Name	Description

E2	None	trgset 2 "zlorpsds1a==0" “A1” scram light off
E7	None	trgset 7 “zlotecswtecpc[1]==1” “C” TEC pump green light
E20	None	trgset 20 “RA:MUX09C059==1” South SDV not drained annunciator trg 20 “set rdlsdv1inst = 30"

2. Malfunctions

Number	Title	Trigger	TD	Severity	Ramp	Initial
RP04	Group 1 Isolation Failure	A	0	N/A	N/A	As Is
RD02	ATWS	E7	0	98	N/A	As Is
RD14	Single Rod Scram (Rod 30-23)	E2	0	N/A	N/A	As Is
SW07C	“C” TEC pump trip	E5	0	N/A	N/A	As Is
TU03A	Main Turbine Bearing High Vibration	E6	0	70	7:00	40
MS03B	Steam Rupture Outside Primary Containment (Steam Tunnel)	E8	0	1.5	5:00	As Is

3. Remotes

Number	Title	Trigger	TD	Value	Ramp
RD18	EOP PTMs 61 & 62	E18	0	INSTALL	N/A

4. Overrides

Instrument	Tag	Trigger	TD	Value	Ramp
"A" SLC pump green light (labeled as squib valve ready light)	ZLOSLCSWS1A[1]	A	N/A	OFF	N/A
"A" Squib Valve Ready Light	ZLOSLCSWS1A[3]	A	N/A	OFF	N/A
"A" SLC Pump Control Switch	ZDISLCSWS1A[1]	A	N/A	STOP	N/A
"A" Squib Continuity Alarm	RA:MUX09C072	A	N/A	ON	N/A
Scram Discharge Volume Isol Drain Valve V-33 Close Green light	ZLORPLT5ADS4G	E4	0	ON	N/A
Scram Discharge Volume Isol Drain Valve V-33 Open Red light	ZLORPLT5ADS5R	E4	10	OFF	N/A
9-5-1/G-5, South SDV Not Drained Alarm	RA:MUX09C059	E4	3:00	ON	N/A

5. Panel Set-up

a. Reduce reactor power to ~ 90% using reactor recirculation pumps.

b. Place a Red Tag on "A" SLC pump control switch.

c. Advance all charts and alarm typers to clean paper.

d. Adjust APRM GAFs.

e. Balance the Main Generator regulator.

Turnover Information

A. Plant Status:

1. The plant is operating at approximately 90% power near the end of the operating cycle.

2. Rod Sequence Information:

Page:	2
Rod:	18-27
Notch:	26

3. Tech. Spec. Limitations in effect:

   Day 2 of 7 day LCO (3.1.7.A.1) for “A” SLC pump. The pump tripped during a surveillance. Maintenance is investigating.

4. Significant problems/abnormalities:

   None.

5. Evolutions/maintenance for the on-coming shift:

   - Perform 6.1RPS.301 as soon as the crew takes the watch.
   - Continue investigation of “A” SLC pump.
   - Restore power to rated when contacted by the Load Dispatcher.

2003 CNS NRC **Reactor Operator** Examination

Question Number	Record Number	Description	Source	Cognitive Level	References Required
1	1111	COR0010102, AC Electrical Distribution	Direct	3	
2	14036	COR0010102, AC Electrical Distribution	New	2	
3	1099	COR0010102, AC Electrical Distribution	direct	2	
4	14035	COR0011302, OPS MAIN GENERATOR AND AUXILIARIES	New	2	
5	14670	COR0011402, OPS MAIN TURBINE	direct	1	
6	19124	COR0011802, OPS Radiation Monitoring	direct	2	
7	5084	COR0012002, OPS Reactor Water Cleanup	modified	1	
8	14043	COR0012402, OPS TURBINE EQUIPMENT COOLING SYSTEM	New	1	
9	14045	COR0013001, HWC Gas Generation System	New	2	
10	2931	COR0020302, CONTAINMENT	direct	2	
11	10081	COR0020302, CONTAINMENT	direct	3	
12	5155	COR0020402, CONTROL ROD DRIVE HYDRAULICS	direct	2	
13	14040	COR0020602, CORE SPRAY	New	2	
14	14050	COR0020702, OPS DC ELECTRICAL DISTRIBUTION	New	1	
15	19090	COR0020702, OPS DC ELECTRICAL DISTRIBUTION	direct	3	
16	1507	COR0020802, Diesel Generators	direct	1	
17	14032	COR0021102, OPS High Pressure Coolant Injection (HPCI)	New	2	
18	19051	COR0021102, OPS High Pressure Coolant Injection (HPCI)	direct	2	
19	16513	COR0021402, Digital Electro-Hydraulic Control	direct	2	
20	1058	COR0021502, NUCLEAR BOILER INSTRUMENTATION	direct	2	
21	5425	COR0021602, OPS NUCLEAR PRESSURE RELIEF	New	3	
22	5608	COR0021602, OPS NUCLEAR PRESSURE RELIEF	direct	2	
23	14679	COR0021702, PLANT MANAGEMENT INFORMATION SYSTEM	direct	2	
24	14451	COR0021802, OPS Reactor Core Isolation Cooling (RCIC)	direct	1	
25	14027	ACD0070307, Thermal Hydraulics (GP)	New	3	Calculator, Steam Tables, GFES Formula sheet.
26	2521	COR0021902, REACTOR EQUIPMENT COOLING	direct	2	
27	14039	COR0021902, REACTOR EQUIPMENT COOLING	Modified	2	
28	14024	COR0022002, OPS REACTOR MANUAL CONTROL SYSTEM	New	1	
29	1208	COR0022102, REACTOR PROTECTION SYSTEM	Modified	1	
30	19096	COR0022202, REACTOR RECIRCULATION	direct	1	
31	1744	COR0022302, RESIDUAL HEAT REMOVAL	direct	1	
32	4029	COR0022302, RESIDUAL HEAT REMOVAL	modified	2	
33	14028	COR0022302, RESIDUAL HEAT REMOVAL	New	1	
34	2127	COR0022602, OPS ROD WORTH MINIMIZER	direct	1	
35	14033	COR0022802, OPS STANDBY GAS TREATMENT	New	2	
36	14025	COR0022902, STANDBY LIQUID CONTROL	New	1	
37	5348	COR0023002, SOURCE RANGE MONITOR SUBSYSTEM	direct	1	

Question Number	Record Number	Description	Source	Cognitive Level	References Required
38	19084	COR0023102, TRAVERSING IN-CORE PROBE	direct	1	
39	14004	COR0023202, OPS REACTOR VESSEL LEVEL CONTROL	New	1	
40	14014	COR0023402, Alternate Shutdown (LO)	direct	1	
41	8970	INT0070501, OPS Introduction to Technical Specifications	modified	3	
42	3995	INT0070502, CNS Tech. Spec. 3.1, Reactivity Control Systems	direct	2	
43	14048	INT0070504, CNS Tech. Spec. 3.3, Instrumentation	New	1	
44	14042	INT0070505, CNS Tech. Spec. 3.4, Reactor Coolant System (RCS	New	2	T.S. 3.0 section and bases, T.S. LCO 3.4.4 and bases, T.S. LCO 3.4.5 and bases.
45	6226	INT0320128, CNS Abnormal Procedures (RO) Containment	direct	1	
46	5247	INT0080605, OPS FLOWCHART 1A - RPV CONTROL/RPV PRESSURE	direct	2	
47	13407	INT0080605, OPS FLOWCHART 1A - RPV CONTROL/RPV PRESSURE	direct	2	
48	14000	INT0080605, OPS FLOWCHART 1A - RPV CONTROL/RPV PRESSURE	New	2	
49	2403	COR0020902, Digital Electro-Hydraulic Control	direct	1	
50	19034	INT0080605, OPS FLOWCHART 1A - RPV CONTROL/RPV PRESSURE	direct	2	EOP Graphs
51	14001	INT0080606, FLOWCHART 6A - RPV PRESSURE/POWER FAILURE-TO-SCR	New	2	
52	16472	INT0080606, FLOWCHART 6A - RPV PRESSURE/POWER FAILURE-TO-SCR	direct	2	
53	16485	INT0080612, FLOWCHART 7B - RPV FLOODING FAILURE-TO-SCRAM	Modified	3	EOP-1A, EOP-2A, EOP-2B, EOP-6A, EOP-7A EOP-1A, EOP-2A, EOP-2B, EOP-6A, EOP-7A and EOP Graphs with entry conditions and cautions removed.and EOP Graphs.
54	54	INT0080610, OPS FLOWCHART 7A - RPV LEVEL FAILURE-TO-SCRAM	New	1	
55	14047	INT0080613, OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL	New	2	EOP-1A and EOP-3A with entry conditions and cautions removed, EOP graphs.
56	5268	INT0080613, OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL	modified	2	EOP-3A with entry conditions and cautions removed.
57	5332	INT0080613, OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL	direct	2	EOP-3A with cautions and entry conditions removed.

Question Number	Record Number	Description	Source	Cognitive Level	References Required
58	5333	INT0080613, OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL	direct	1	
59	14023	INT0080613, OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL	New	1	EOP-3A with cautions and entry conditions removed.
60	19082	INT0080613, FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL	New	2	EOP-3A with cautions and entry conditions removed, 5.9H2O2.
61	5730	INT0080617, OPS FLOWCHART 5A - SECONDARY CONTAINMENT AND RAD	modified	1	
62	14019	INT0080617, OPS FLOWCHART 5A - SECONDARY CONTAINMENT AND RAD	New	1	EOP 5A with entry conditions and cautions removed.
63	14021	INT0080617, OPS FLOWCHART 5A - SECONDARY CONTAINMENT AND RAD	New	1	
64	16483	INT0080617, OPS FLOWCHART 5A - SECONDARY CONTAINMENT AND RAD	direct	2	
65	19068	INT0080617, OPS FLOWCHART 5A - SECONDARY CONTAINMENT AND RAD	direct	1	
66	14046	INT0080618, EOP AND SAG GRAPHS	New	2	EOP Graphs
67	4243	INT0320101 CNS ADMINISTRATIVE PROCEDURES (RO)	direct	1	
68	12215	INT0320102, CNS Administrative Procedures Site Services Proc	direct	2	
69	12209	INT0320103, CNS Administrative Procedures Conduct of Operati	direct	2	
70	13673	INT0320104, CNS Administrative Procedures General Operating	direct	1	
71	16796	INT0320126, CNS Abnormal Procedures (RO) Cooling Water	direct	2	
72	14426	INT0320132, CNS Abnormal Procedures (RO) Off Gas/Vaccuum	direct	2	
73	4214	INT0320134, OPS CNS Abnormal Procedures (RO) - Fire	direct	1	
74	5127	SKL0080101, WATCHSTANDING PRINCIPLES	direct	1	
75	2167	RO only COR0010502, FIRE PROTECTION SYSTEM	direct	1	
76	1290	RO only COR0010602, FUEL POOL COOLING AND DEMINERALIZING SYS	direct	1	
77	5092	RO only COR0010602, FUEL POOL COOLING AND DEMINERALIZING SYS	direct	1	
78	3724	RO only COR0010802, HEATING, VENTILATION, AIR CONDITIONING	direct	3	
79	14041	RO only COR0011602, Off Gas	New	1	
80	3973	RO only COR0011602, Off Gas	direct	1	
81	3085	RO only COR0020102, AVERAGE POWER RANGE MONITOR	direct	1	

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Question Number	Record Number	Description	Source	Cognitive Level	References Required
82	18311	RO only COR0020202, OPS CONDENSATE AND FEED	direct	2	
83	14053	RO only COR0020202, OPS CONDENSATE AND FEED	New	2	
84	3302	RO only COR0020302, CONTAINMENT	Modified	2	
85	14038	RO only COR0020402, CONTROL ROD DRIVE HYDRAULICS	New	1	
86	14496	RO only COR0020702, OPS DC ELECTRICAL DISTRIBUTION	direct	2	
87	14034	RO only COR0020802, DIESEL GENERATORS	New	1	
88	3286	RO only COR0021502, NUCLEAR BOILER INSTRUMENTATION	direct	1	
89	1238	RO only COR0022002, OPS REACTOR MANUAL CONTROL SYSTEM	New	1	
90	2816	RO only COR0022302, RESIDUAL HEAT REMOVAL	Modified	3	
91	19100	RO only SKL0124223, OPS RESIDUAL HEAT REMOVAL	direct	1	
92	5401	RO only COR0022402, ROD BLOCK MONITOR	direct	1	
93	5070	RO only COR0022902, STANDBY LIQUID CONTROL	direct	1	
94	16441	RO only INT0070501, OPS Introduction to Technical Specificat	direct	2	
95	14003	RO only INT0070504, CNS Tech. Spec. 3.3, Instrumentation	New	3	T.S. 3.0 section and bases, T.S. LCO 3.3.1.1 and bases.
96	14026	RO only INT0070504, CNS Tech. Spec. 3.3, Instrumentation	New	2	T.S. 3.0 section and bases, T.S. LCO 3.3.3.1 and bases.
97	19123	RO only INT0080617, OPS FLOWCHART 5A - SECONDARY CONTAINMENT	direct	1	
98	16466	RO only INT0320104, CNS Administrative Procedures General Op	direct	3	
99	13406	RO only INT0320124, CNS Abnormal Procedure (RO) Reactor Reci	direct	2	
100	12222	RO only INT0320115, CHEMISTRY PROCEDURES	Direct	2	
	58	**Direct**		**45**	**Memory (Cognitive 1)**
	10	**Modified**		**44**	**Comprehension (Cognitive 2)**
	32	**New**		**11**	**Analysis (Cognitive 3)**
	100	**total**			

Question Number	Record Number	Description	Source	Cognitive Level	References Required
1	1111	COR0010102, AC Electrical Distribution	Direct	3	
2	14036	COR0010102, AC Electrical Distribution	New	2	
3	1099	COR0010102, AC Electrical Distribution	direct	2	
4	14035	COR0011302, OPS MAIN GENERATOR AND AUXILIARIES	New	2	
5	14670	COR0011402, OPS MAIN TURBINE	direct	1	
6	19124	COR0011802, OPS Radiation Monitoring	direct	2	
7	5084	COR0012002, OPS Reactor Water Cleanup	modified	1	
8	14043	COR0012402, OPS TURBINE EQUIPMENT COOLING SYSTEM	New	1	
9	14045	COR0013001, HWC Gas Generation System	New	2	
10	2931	COR0020302, CONTAINMENT	direct	2	
11	10081	COR0020302, CONTAINMENT	direct	3	
12	5155	COR0020402, CONTROL ROD DRIVE HYDRAULICS	direct	2	
13	14040	COR0020602, CORE SPRAY	New	2	
14	14050	COR0020702, OPS DC ELECTRICAL DISTRIBUTION	New	1	
15	19090	COR0020702, OPS DC ELECTRICAL DISTRIBUTION	direct	3	
16	1507	COR0020802, Diesel Generators	direct	1	
17	14032	COR0021102, OPS High Pressure Coolant Injection (HPCI)	New	2	
18	19051	COR0021102, OPS High Pressure Coolant Injection (HPCI)	direct	2	
19	16513	COR0021402, Digital Electro-Hydraulic Control	direct	2	
20	1058	COR0021502, NUCLEAR BOILER INSTRUMENTATION	direct	2	
21	5425	COR0021602, OPS NUCLEAR PRESSURE RELIEF	New	3	
22	5608	COR0021602, OPS NUCLEAR PRESSURE RELIEF	direct	2	
23	14679	COR0021702, PLANT MANAGEMENT INFORMATION SYSTEM	direct	2	
24	14451	COR0021802, OPS Reactor Core Isolation Cooling (RCIC)	direct	1	
25	14027	ACD0070307, Thermal Hydraulics (GP)	New	3	Calculator, Steam Tables, GFES Formula sheet.
26	2521	COR0021902, REACTOR EQUIPMENT COOLING	direct	2	
27	14039	COR0021902, REACTOR EQUIPMENT COOLING	Modified	2	
28	14024	COR0022002, OPS REACTOR MANUAL CONTROL SYSTEM	New	1	
29	1208	COR0022102, REACTOR PROTECTION SYSTEM	Modified	1	
30	19096	COR0022202, REACTOR RECIRCULATION	direct	1	
31	1744	COR0022302, RESIDUAL HEAT REMOVAL	direct	1	
32	4029	COR0022302, RESIDUAL HEAT REMOVAL	modified	2	
33	14028	COR0022302, RESIDUAL HEAT REMOVAL	New	1	
34	2127	COR0022602, OPS ROD WORTH MINIMIZER	direct	1	
35	14033	COR0022802, OPS STANDBY GAS TREATMENT	New	2	
36	14025	COR0022902, STANDBY LIQUID CONTROL	New	1	
37	5348	COR0023002, SOURCE RANGE MONITOR SUBSYSTEM	direct	1	

Question Number	Record Number	Description	Source	Cognitive Level	References Required
38	19084	COR0023102, TRAVERSING IN-CORE PROBE	direct	1	
39	14004	COR0023202, OPS REACTOR VESSEL LEVEL CONTROL	New	1	
40	14014	COR0023402, Alternate Shutdown (LO)	direct	1	
41	8970	INT0070501, OPS Introduction to Technical Specifications	modified	3	
42	3995	INT0070502, CNS Tech. Spec. 3.1, Reactivity Control Systems	direct	2	
43	14048	INT0070504, CNS Tech. Spec. 3.3, Instrumentation	New	1	
44	14042	INT0070505, CNS Tech. Spec. 3.4, Reactor Coolant System (RCS	New	2	T.S. 3.0 section and bases, T.S. LCO 3.4.4 and bases, T.S. LCO 3.4.5 and bases.
45	6226	INT0320128, CNS Abnormal Procedures (RO) Containment	direct	1	
46	5247	INT0080605, OPS FLOWCHART 1A - RPV CONTROL/RPV PRESSURE	direct	2	
47	13407	INT0080605, OPS FLOWCHART 1A - RPV CONTROL/RPV PRESSURE	direct	2	
48	14000	INT0080605, OPS FLOWCHART 1A - RPV CONTROL/RPV PRESSURE	New	2	
49	2403	COR0020902, Digital Electro-Hydraulic Control	direct	1	
50	19034	INT0080605, OPS FLOWCHART 1A - RPV CONTROL/RPV PRESSURE	direct	2	EOP Graphs
51	14001	INT0080606, FLOWCHART 6A - RPV PRESSURE/POWER FAILURE-TO-SCR	New	2	
52	16472	INT0080606, FLOWCHART 6A - RPV PRESSURE/POWER FAILURE-TO-SCR	direct	2	
53	16485	INT0080612, FLOWCHART 7B - RPV FLOODING FAILURE-TO-SCRAM	Modified	3	EOP-1A, EOP-2A, EOP-2B, EOP-6A, EOP-7A EOP-1A, EOP-2A, EOP-2B, EOP-6A, EOP-7A and EOP Graphs
54	54	INT0080610, OPS FLOWCHART 7A - RPV LEVEL FAILURE-TO-SCRAM	New	1	
55	14047	INT0080613, OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL	New	2	EOP-1A and EOP-3A with entry conditions and cautions removed, EOP graphs.
56	5268	INT0080613, OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL	modified	2	EOP-3A with entry conditions and cautions removed.
57	5332	INT0080613, OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL	direct	2	EOP-3A with cautions and entry conditions removed.
58	5333	INT0080613, OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL	direct	1	

Question Number	Record Number	Description	Source	Cognitive Level	References Required
59	14023	INT0080613, OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL	New	1	EOP-3A with cautions and entry conditions removed.
60	19082	INT0080613, FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL	New	2	EOP-3A with cautions and entry conditions removed, 5.9H2O2.
61	5730	INT0080617, OPS FLOWCHART 5A - SECONDARY CONTAINMENT AND RAD	modified	1	
62	14019	INT0080617, OPS FLOWCHART 5A - SECONDARY CONTAINMENT AND RAD	New	1	EOP 5A with entry conditions and cautions removed.
63	14021	INT0080617, OPS FLOWCHART 5A - SECONDARY CONTAINMENT AND RAD	New	1	
64	16483	INT0080617, OPS FLOWCHART 5A - SECONDARY CONTAINMENT AND RAD	direct	2	
65	19068	INT0080617, OPS FLOWCHART 5A - SECONDARY CONTAINMENT AND RAD	direct	1	
66	14046	INT0080618, EOP AND SAG GRAPHS	New	2	EOP Graphs
67	4243	INT0320101 CNS ADMINISTRATIVE PROCEDURES (RO)	direct	1	
68	12215	INT0320102, CNS Administrative Procedures Site Services Proc	direct	2	
69	12209	INT0320103, CNS Administrative Procedures Conduct of Operati	direct	2	
70	13673	INT0320104, CNS Administrative Procedures General Operating	direct	1	
71	16796	INT0320126, CNS Abnormal Procedures (RO) Cooling Water	direct	2	
72	14426	INT0320132, CNS Abnormal Procedures (RO) Off Gas/Vaccuum	direct	2	
73	4214	INT0320134, OPS CNS Abnormal Procedures (RO) - Fire	direct	1	
74	5127	SKL0080101, WATCHSTANDING PRINCIPLES	direct	1	
75	768	SRO Only COR0012102, Refueling	Modified	1	
76	5101	SRO Only COR0012102, Refueling	Direct	1	
77	5468	SRO Only COR0020902, Digital Electro-Hydraulic Control	New	1	
78	14052	SRO Only COR0020302, CONTAINMENT	New	3	
79	32	SRO Only COR0020302, CONTAINMENT	New	1	
80	36	SRO Only COR0020302, CONTAINMENT	New	1	
81	44	SRO Only SKL0124223, OPS RESIDUAL HEAT REMOVAL	New	1	
82	4002	SRO Only INT0070501, OPS Introduction to Technical Specific	Direct	2	
83	38	SRO Only INT0070501, OPS Introduction to Technical Specific	New	2	T.S. 3.0 section and bases, T.S. LCO 3.3.3.1 and bases, T.S. LCO 3.8.1 and bases, LCO 3.8.3 and bases.
84	48	SRO Only INT0070502, CNS Tech. Spec. 3.1, Reactivity Contro	New	2	T.S. 3.0 section and bases, T.S. LCO 3.1.3 and bases, 3.1.4 and bases.
85	114	SRO Only INT0070502, CNS Tech. Spec. 3.1, Reactivity Contro	Direct	1	T.S. 3.0 section and bases, 3.3.2.1 and bases.

Question Number	Record Number	Description	Source	Cognitive Level	References Required
86	42	SRO Only INT0070505, CNS Tech. Spec. 3.4, Reactor Coolant S	New	2	
87	16415	SRO Only INT0070507, CNS Tech. Spec. 3.6, Containment Syste	Direct	1	T.S. 3.0 section and bases, T.S. LCO 3.6.1.3 and bases, 3.6.2.3 and bases, 6.1RHR.201.
88	24	SRO Only INT0070507, CNS Tech. Spec. 3.6, Containment Syste	New	1	
89	40	SRO Only INT0070509, CNS Tech. Spec. 3.8, Electrical Power	New	1	
90	46	SRO Only INT0070604, TRM - Fire Protection	New	3	TRM section 3.11 and 0.23 "CNS FIRE PROTECTION PLAN".
91	47	SRO Only INT0070702, ODAM Specifications	New	1	
92	30	SRO Only INT0080613, OPS FLOWCHART 3A - PRIMARY CONTAINMENT	New	1	EOP-3A with cautions and entry conditions removed.
93	34	SRO Only INT0080617, OPS FLOWCHART 5A - SECONDARY CONTAINME	New	1	EOP-1A and EOP-5A with entry conditions and cautions removed.
94	16569	SRO Only INT0080618, EOP AND SAG GRAPHS	Direct	2	All EOP graphs.
95	28	SRO Only INT0320123, CNS Abnormal Procedures (RO) Reactivit	New	3	
96	5668	SRO Only INT0320134, OPS CNS Abnormal Procedures (RO) - Fir	Direct	2	
97	9015	SRO Only INT0320136, CNS Abnormal Procedures (RO) Miscellan	Direct	1	
98	5760	SRO Only INT0320101, CNS Administrative Procedures Volume 0	Direct	3	
99	12218	SRO Only INT0320116, Health Physics Procedures	Direct	2	
100	9684	SRO Only SKL0110101, SRO Upgrade Self-Study Program	Direct	2	0.50, Outage Management Program.
	51	**Direct**		**45**	**Memory (Cognitive 1)**
	9	**Modified**		**44**	**Comprehension (Cognitive 2)**
	40	**New**		**11**	**Analysis (Cognitive 3)**
	100	**Total**			