Using the attached power to flow map, answer the following:

- The plant is operating at the 81% rodline.

Select the MINIMUM core flow at which the plant can operate at power and still be assured of avoiding power oscillations or instabilities.

a. 25%

b. 40%

c. 45%

d. 48%

Answer	c	**Exam Level**	B	**Cognitive Level**	Comprehension	**Facility:** Hope Creek	**Exam Date:**	06/17/2003
Tier:	Emergency and Abnormal Plant Evolutions	**RO Group**	2	**SRO Group**	2			295001K206
295001	Partial or Complete Loss of Forced Core Flow Circulation						**Record Number**	1

AK2. Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION and the following:

AK2.06 Reactor power 3.8 3.8

Explanation of Answer

Justification:

· 45%-Correct- IAW TS 3.4.1.1.b and HC.OP-AB.RPV-0002. 81% rodline with reduced rrp speeds, the boundry of the EXIT or Instability Region

· 48% -Incorrect- see TS 3.4.1.1.b and HC.OP-AB.RPV-0002

· 25% -Incorrect- see TS 3.4.1.1.b and HC.OP-AB.RPV-0002

· 40% - Incorrect- see TS 3.4.1.1.b and HC.OP-AB.RPV-0002

Reference Title

HC.OP-AB.RPV-0002

Learning Objectives

RECCONE012 (R) Given a copy of the Power-to-Flow map, describe the significance of all associated lines, IAW the Recirculation Flow Control System Lesson Plan.

Material Required for Examination HC.OP-AB.RPV-0002 Power to flow chart with region labeling and line nomenclature removed.

Question Source: Facility Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: Vision Bank QID# Q56742

Given the following conditions:

- A plant startup is in progress.
- Reactor power is 10.5 percent.
- Offgas recombiner train 0 just tripped.
- Offgas Recombiner train 1 is NOT available.

Which one of the following decribes the action required to allow use of the Mechanical Vacuum Pumps (MVP) and the bases for that power level?

a. Reduce reactor power by 5 percent; Combustible gas concentrations may cause an explosion in the SJAE Aftercondenser.

b. Reduce reactor power by 5 percent; Offsite radiological release may be above allowable limits at the South Plant Vent.

c. Reduce reactor power by 6 percent; Offsite radiological release may be above allowable limits at the North Plant Vent.

d. Reduce reactor power by 6 percent; Combustible gas concentrations may cause an explosion at the MVP suction.

Answer	d	Exam Level	S	Cognitive Level	Application	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Emergency and Abnormal Plant Evolutions	RO Group	2	SRO Group	2				295002A202
295002	Loss of Main Condenser Vacuum							Record Number	2

AA2. Ability to determine and/or interpret the following as they apply to LOSS OF MAIN CONDENSER VACUUM:

AA2.02 Reactor power: Plant-Specific 3.2 3.3

Explanation of Answer

JUSTIFICATION:55.43(4) & (5) SRO Radiation hazards and procedure actions required by assessment of facility conditions and procedure limitations.
Reduce reactor power by 6 percent; Combustible gas concentrations may cause an explosion at the MVP suction. Correct. Subsequent actions of HC.OP-AB.BOP-0006 direct power reduction to less than 5 percent. A 6 percent reduction for the given conditions will result in < 5 percent power. The explosion concern is the MVP inlet piping and pump.
Reduce reactor power by 5 percent; Combustible gas concentrations may cause an explosion in the SJAE Aftercondenser. Incorrect. 5 percent is the power limit. A 5 percent reduction will still result in >5 percent power. The explosion concern is the MVP inlet piping and pump.
Reduce reactor power by 5 percent; Offsite radiological release may be above allowable limits at the South Plant Vent. Incorrect. Wrong power level.
Reduce reactor power by 6 percent; Offsite radiological release may be above allowable limits at the North Plant Vent. Incorrect. Wrong bases. Wrong release path.

Reference Title
HC.OP-AB.BOP-0006 D.1
HC.OP-SO.CG-0001
NOH01AIRREM-01

Learning Objectives	
ABBOP6E007	(R) Explain the bases for Subsequent Actions and the information contained in the Discussion Bases Section of Main Condenser Vacuum.
IOP003E004	(R) Apply Precautions, Limitations and Notes while executing the STARTUP FROM COLD SHUTDOWN TO RATED POWER Integrated Operating Procedure.

Material Required for Examination	None

Question Source:	New	Question Modification Method:	
Question Source Comments:			

Given the following conditions:

- The plant is operating at 100 percent power.
- TACS is on the 'A' Loop of SACS.
- 'A' 1E 4.16 KV bus 10A401 has de-energized due to bus fault.

DELETED

Which one of the following describes a result of the bus fault and the reason for the result?

a. All RACS Pumps trip due to LO-LO Head tank level.

b. 'B' SACS Expansion Tank overflows due to power loss to TACS return valves.

c. RACS Head Tank overflows due to makeup valve power loss.

d. 'A' & 'C' SACS Pump trip due to LO-LO-LO Expansion Tank level.

Answer	c	Exam Level	B	Cognitive Level	Memory	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Emergency and Abnormal Plant Evolutions		RO Group	2	SRO Group	1			295003K105
295003	Partial or Complete Loss of A.C. Power							Record Number	3

AK1. Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER:

AK1.05 Failsafe component design 2.6 2.7

Explanation of Answer

Justification:
RACS Head Tank overflows due to makeup valve power loss. Correct. The makeup valve solenoid to RACS Head tank fails open on a loss of power to 10Y205 from 10A401 bus.
'B' SACS Expansion Tank overflows due to power loss to TACS return valves. Incorrect. SACS ET Makeup uses an MOV and inverter backed instrumentation. Sluicing causes "A" tank to overflow. B Tank lowers.
All RACS Pump trip due to LO-LO Head tank level. Incorrect. Head Tank Level goes high. Pumps trip on bus restoration.
A & C SACS Pump trip due to LO-LO-LO Expansion Tank level. A&C Pumps trip but for the wrong reasons. A trips due to loss of power; C trips on low pump differential pressure.

Reference Title
HC.OP-AB.ZZ-0170
HC.OP-GP.PB-0001

Learning Objectives	
0AB170E001	Recognize abnormal indications/alarms and/or procedural requirements for implementing, Loss of 4.16 KV Bus 10A401 A Channel, Abnormal Operating Procedure.

Material Required for Examination None

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following conditions:

- The plant is operating at 100% power.
- A LOP signal is generated due to a Loss of Off-site Power.
- Just prior to the EDG output breakers closing, a LOCA signal is generated due to a Loss of Coolant Accident.

Which of the following is the response of the LOP and LOCA sequencers for these conditions?

a. As soon as power is restored to the buses, the LOCA sequencer will control the restoration of all loads.

b. The LOCA sequencer will begin to sequence until the diesel generator output breakers close, then the LOP sequencer will complete load restoration.

c. As soon as power is restored the buses, the LOP sequencer will control the restoration of all loads.

d. The LOP sequencer will begin to sequence until the diesel generator output breakers close, then the LOCA sequencer will complete load restoration.

Answer a **Exam Level** B **Cognitive Level** Comprehension **Facility:** Hope Creek **Exam Date:** 06/17/2003

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 2 **SRO Group** 1 295003K204

295003 Partial or Complete Loss of A.C. Power **Record Number** 4

AK2. Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF A.C. POWER and the following:

AK2.04 A.C. electrical loads 3.4 3.5

Explanation of Answer

JUSTIFICATION:
CORRECT - As soon as power is restored to the buses, the LOCA sequencer will control the restoration of all loads. Both the LOP & LOCA sequencers start when power is available. The LOP sequencer will control until the LOCA signal is received. With a LOCA & a LOP signal present, the LOCA Sequencer has priority. When the LOCA signal is received, the LOCA sequencer will control load sequencing.
INCORRECT - The LOCA sequencer will begin to sequence until the diesel generator output breakers close, then the LOP sequencer will complete load restoration. Both the LOP & LOCA sequencers start when power is available. With a LOCA & a LOP signal present, the LOCA Sequencer has priority.
INCORRECT - As soon as power is restored the buses, the LOP sequencer will control the restoration of all loads. With a LOCA & a LOP signal present, the LOCA Sequencer has priority.
INCORRECT - The LOP sequencer will begin to sequence until the diesel generator output breakers close, then the LOCA sequencer will complete load restoration. Both the LOP & LOCA sequencers start when power is available. With a LOCA & a LOP signal present, the LOCA Sequencer has priority.

Reference Title

HC.OP-AB.ZZ-0135

Learning Objectives

0AB135E003 (R) Discuss the operational implications of the abnormal indications/alarms for system operating parameters related to Station Blackout/Loss Of Offsite Power Diesel Generator Malfunction, Abnormal Operating Procedure.

Material Required for Examination None

Question Source: Facility Exam Bank **Question Modification Method:** Direct From Source

Question Source Comments: Vision Bank QID#Q61290

Plant conditions are as follows:

- Reactor Power is 20 %
- Control rod 30-31 is selected at position 12.

Which one of the following describes the response of RMCS if the Main Turbine were to trip?

Reactor Manual Control will:

a. block all control rod movement because the reactor has scrammed.

b. allow control rod insertion using the Continuous Insert PB only.

c. automatically bypass RWM blocks due to the effects of colder feedwater.

d. actively enforce control rod blocks due to loss of First Stage Turbine pressure.

Answer c **Exam Level** B **Cognitive Level** Comprehension **Facility:** Hope Creek **Exam Date:** 06/17/2003

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 1 **SRO Group** 2 295005A103

295005 Main Turbine Generator Trip **Record Number** 5

AA1. Ability to operate and/or monitor the following as they apply to MAIN TURBINE GENERATOR TRIP:

AA1.03 Reactor manual control/rod control and information system 2.7 2.8

Explanation of Answer

Justification:
automatically bypass RWM blocks due to the effects of colder feedwater. Correct. 20 percent power is the upper limit of RWM Low Power Set Point or the power level that RWM enforces rod blocks. Above 20 percent, the blocks are bypassed and are indicated only. A turbine trip will cause reactor power to increase due to the positive reactivity effects of the loss of feedwater heating followng the turbine trip.
allow control rod insertion using the Continuous Insert Pb only. Incorrect. Continous Insert bypasses the Activity Control timer card, not the Rod Blocks.
block all control rod movement because reactor has scrammed. Incorrect. Reactor will not automatically scram from turbine trip at 20 percent power as a result of a turbine trip.
actively enforce control rod blocks due to loss of First Stage Turbine pressure. Incorrect First stage turbine pressure no longer input to cause rod blocks. Input removed by DCP.

Reference Title

HC.OP-SO.SF-0003

Learning Objectives

ABBOP2E004	Explain the reasons for how plant/system parameters respond when implementing Main Turbine.
RODMINE005	(R) Given plant conditions, determine if the conditions will cause the Rod Worth Minimizer to initiate a rod block signal.

Material Required for Examination None

Question Source: Facility Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments: INPO BANK QID# 861. Significantly modified.

Conditions are as follows:

- The plant is operating at 87% power
- It is near the end of a fuel cycle.
- Main Turbine Stop Valves (TSVs) are being tested to validate the EOC-RPT setpoints.
- Two TSVs initiate an EOC-RPT signal at 10% closed.
- Two TSVs initiate an EOC-RPT signal at 5% closed.

Which of the following is a safety implication (if any) of this condition?

a. There are no safety implications because the TSV EOC-RPT trip is a 1-out-of-2 logic.

b. There will be an excessive thermal margin upon EOC-RPT actuation if these TSVs close at power.

c. Reactor safety has been enhanced by the overly conservative trip value for TSV closure.

d. Void reactivity feedback may exceed control rod reactivity if these TSVs close at power.

Answer d **Exam Level** S **Cognitive Level** Comprehension **Facility:** Hope Creek **Exam Date:** 06/17/2003

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 1 **SRO Group** 2 295005G225

295005 Main Turbine Generator Trip **Record Number** 6

2.2 Equipment Control

2.2.25 Knowledge of bases in technical specifications for limiting conditions for operations and safety limits. 2.5 3.7

Explanation of Answer

Justification 55.43(2) IAW TS Bases 3/4.3.4, TS 3.3.4.2 & Table 3.3.4.2-2 - The purpose of the EOC-RPT system is to recover the loss of thermal margin that occurs at the end of a cycle. Void reactivity feedback due to a pressurization transient can add positive reactivity to the reactor at a faster rate than the control rods add negative scram reactivity.
Void reactivity feedback may exceed control rod reactivity if these TSVs close at power-Correct- If the TSVs generate an EOC-RPT signal at 10% closed vice the nominal 5% closed (7% allowable) then an excess positive reactivity will be added upon TSV closure.
There will be an excessive thermal margin upon EOC-RPT actuation if these TSVs close at power-Incorrect- If the TSVs generate an EOC-RPT signal at 10% closed vice the nominal ~ 5% closed (~ 7% allowable) then an excess positive reactivity will be added upon TSV closure.

Reactor safety has been enhanced by the overly conservative trip value for TSV closure-Incorrect-setpoint is 5% +2% not 10%
There are no safety implications because the TSV EOC-RPT trip is a 1-out-of-2 logic-Incorrect- The TSV closure uses various combinations, not 1 of 2 twice.

Reference Title

HCTS Bases.3.3.4.2

Learning Objectives

TECSPCE009	Explain the bases for Hope Creek Generating Station Technical Specification Safety Limits and Limiting Safety System Settings. (SRO/STA Only)
MNTURBE022	(R) Given a scenario of applicable operating conditions and access to Technical Specifications: a. Select those sections applicable to the Main Turbine. b. Evaluate Main Turbine operability and determine required action(s) based upon inoperability. c. Explain the bases for those technical specification items associated with the Main Turbine (SRO ONLY).

Material Required for Examination Technical Specification 3.3.4.2 and Table 3.3.4.2-2

Question Source: Facility Exam Bank

Question Modification Method: Editorially Modified

Question Source Comments: VISION BANK QID# Q62005

Given the following conditions:

- Preparations for a reactor startup from a refueling outage are in progress.
- Reactor Building ambient temperature is 74 degrees F.
- The Reactor Building Equipment Operator is charging the hydraulic control unit accumulators with nitrogen to a pressure of 590 psig
- Several days later with the Unit at 100% power, Reactor Building temperatures have stabilized at 92 degrees F

Which of the following describes the impact on the Control Rod Drive Hydraulic system operations for these conditions?
(Assume NO leakage)
(Refer to attached figure.)

The individual control rod scram speed will be:

a. faster and may result in mechanism damage.

b. slower and will result in MCPR LCO penalties.

c. slower and will result in reduced reactivity addition rates.

d. faster and may result in MCPR LCO penalties.

Answer a **Exam Level** B **Cognitive Level** Comprehension **Facility:** Hope Creek **Exam Date:** 06/17/2003

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 1 **SRO Group** 1 295006K205

295006 SCRAM **Record Number** 7

AK2. Knowledge of the interrelations between SCRAM and the following:

AK2.05 CRD mechanism 3.1 3.3

Explanation of Answer

Justification:
scram speeds will be faster and may result in mechanism damage. Correct. HC.OP-SO.BF-0002 Precaution 3.1.4. Interpretation of the graph places the N2 pressure too high. Warming up of the Reactor Building will cause the pressure to rise parallel to the desired precharge line and remaining too high.
scram speeds will be slower and will result in MCPR LCO penalties. Incorrect. Scram speeds will be faster.
scram speed will be slower and will result in reduced reactivity addition rates. Scram speeds will be higher.
scram speed will be faster and may result in MCPR LCO penalties. Incorrect. MCPR LCO penalties result from slow scram speeds.

Reference Title

HC.OP-SO.BF-0002

Learning Objectives

CRDHYDE021 (R) Given the "Accumulator Precharge Nitrogen Pressure Versus Ambient Temperature Curve", determine the proper accumulator precharge gas pressure, IAW HC.OP-SO.BF-0002.

Material Required for Examination Accumulator charging pressure graph from HC.OP-SO.BF-0002

Question Source: INPO Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: Inpo Bank QID# 14129 03/26/2001 Peach Bottom modified for Hope Creek

Given the following conditions:

- The plant is operating normally at 100 percent power.

- An EHC failure causes reactor pressure to rise 10 psig in 10 seconds.

Which one of the following describes reactor power response?
(Assume NO operator action)

a. Rises initially due to lower void fraction, then lowers rapidly due to scram.

b. Rises initially due to lower void fraction, then stabilizes at a slightly higher power level.

c. Lowers initially due to greater feedwater heating, then lowers rapidly due to scram.

d. Lowers initially due to greater feedwater heating, then stabilizes at a slightly higher power level.

Answer b **Exam Level** B **Cognitive Level** Comprehension **Facility:** Hope Creek **Exam Date:** 06/17/2003

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 1 **SRO Group** 1 295007K202

295007 High Reactor Pressure **Record Number** 8

AK2. Knowledge of the interrelations between HIGH REACTOR PRESSURE and the following:

AK2.02 Reactor power 3.8 3.8

Explanation of Answer

Justification:
Rises initially due to lower void fraction, then stabilizes at a slightly higher power level. Correct. Rising pressure causes reacor power to rise. 10 psig rise above normal 1005 psig will not reach scram setpoint of 1037 or manual scram threshold of >1030 psig of retainment override of AB-RPV-0005
Rises initially due to lower void fraction, then lowers rapidly due to scram. Incorrect. Would not reach scram threshold.
Lowers initially due to greater feedwater heating, then lowers rapidly due to scram. Incorrect. Rises initially.
Lowers initially due to greater feedwater heating, then stabilizes at a slightly higher power level. Incorrect. Rises initially.

Reference Title

HC.OP-AB.RPV-0005

Learning Objectives

ABRPV5E004 Explain the reasons for how plant/system parameters respond when implementing Reactor Pressure.

Material Required for Examination None

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following conditions:

- The plant is in Operational Condition 3 with RHR A in Shutdown Cooling operation.
- Reactor coolant temperature and pressure is slowly rising.
- The Shutdown Cooling Inboard and Outboard Isolation valves, HV-F009 and F008, have now closed, and the operating RHR pump has tripped.

The reason for these automatic actions is to prevent:

a. steam voiding in the RHR pump seals.

b. overpressurizing the RHR pump seals.

c. establishing a drain path from the RPV to the torus.

d. overpressurizing the shutdown cooling suction piping.

Answer d **Exam Level** B **Cognitive Level** Memory **Facility:** Hope Creek **Exam Date:** 06/17/2003

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 1 **SRO Group** 1 295007K305

295007 High Reactor Pressure **Record Number** 9

AK3. Knowledge of the reasons for the following responses as they apply to HIGH REACTOR PRESSURE:

AK3.05 Low pressure system isolation 3.0 3.2

Explanation of Answer

Justification:
overpressurizing the shutdown cooling suction piping. Correct. The piping isolates to minimze offsite radiological release to the environment if the low pressure piping fails.
overpressurizing the RHR pump seals. Incorrect. Wrong reason. Overpressurize piping
establishing a drain path from the RPV to the torus. Incorrect. Reason for isolation on low RPV level. Wrong reason. .
steam voiding in the RHR pump seals. Incorrect. Reason for seal coolers on RHR pumps.

Reference Title

HCGS TS Bases 3.3.2

Learning Objectives

NSSSS0E007 Given the design bases of the NSSSS, state the purpose of that design function IAW the NSSSS Lesson Plan.

Material Required for Examination None

Question Source: INPO Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: 8896 04/06/1998 Fermi 2 Modified for Hope Creek.

Given the following conditions:

- All 3 RFPT's are in manual speed control.
- RPV level is 35 inches.

Feedwater pump flows are as follows:
- 'A' 3.2 Mlbm/hr
- 'B' 3.6 Mlbm/hr
- 'C' 3.5 Mlbm/hr

Main Steam flows are as follows:
- 'A' 2.6 Mlbm/hr
- 'B' 2.5 Mlbm/hr
- 'C' 2.6 Mlbm/hr
- 'D' 2.4 Mlbm/hr

Based on these conditions, RFP speed demand ______________ must be applied to prevent the RPV Level _____ alarm.

a. raise, 4

b. lower, 4

c. raise, 7

d. lower, 7

Answer	d	Exam Level	B	Cognitive Level	Comprehension	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Emergency and Abnormal Plant Evolutions			RO Group	2	SRO Group	2		295008K103
295008	High Reactor Water Level							Record Number	10

AK1. Knowledge of the operational implications of the following concepts as they apply to HIGH REACTOR WATER LEVEL:

AK1.03 Feed flow/steam flow mismatch 3.2 3.2

Explanation of Answer

Justification:
decrease, 7; Correct. Total Feed Flow is 10.3 Mlbm/hr. Total Steam flow is 10.1 Mlbm/hr. this mismatch will result in an increasing RPV water level. RPV speed demand must lower to prevent the Level 7 alarm. The Level 7 alarm will occur if no action is taken.
increase, 4; Incorrect. Wrong action; wrong alarm.
decrease, 4; Incorrect. Correct action; wrong alarm.
increase, 7; Incorrect. Wrong action; correct alarm.

Reference Title

HC.OP-SO.AE-0001

Learning Objectives

FWCONTE006 (R) Given a set of conditions and a drawing of the controls, instrumentation and/or alarms located in the Main Control Room, identify the status of the Feedwater Control System or its components by evaluation of the controls/instrumentation/alarms, IAW the Feedwater Control System Lesson Plan.

Material Required for Examination None

Question Source:	New
Question Source Comments:	

Question Modification Method:

Using the attached transient analysis plots of a reactor scram, which one of the following failures caused the scram?

a. EHC Pressure Regulator Failure to 0%.

b. Master Level Control Setpoint Failure to 0 inches.

c. EHC Pressure Regulator Failure to 130%.

d. Master Level Control Setpoint Failure to +60 inches.

Answer b **Exam Level** S **Cognitive Level** Comprehension **Facility:** Hope Creek **Exam Date:** 06/17/2003

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 1 **SRO Group** 1 295009A201

295009 Low Reactor Water Level **Record Number** 11

AA2. Ability to determine and/or interpret the following as they apply to LOW REACTOR WATER LEVEL:

AA2.01 Reactor water level 4.2 4.2

Explanation of Answer

Justification:
55.43(1) Conditions and limitation of the HC UFSAR
Master Level control fails to 0 inches - correct. RPV level drops with no rpv pressure rise.
Master Level control fails to +60 inches - incorrect. RPV Level lowers
EHC Press reg fails to 130 %- incorrect. RPV pressure remains steady until scram on low level.
EHC Press reg fails to 0 %- incorrect. RPV pressure remains steady until scram on low level.
Provide UFSAR figure 15.2-8 with title block removed

Reference Title

UFSAR 15.2

Learning Objectives

EO101LE006 (R) Given any step of the procedure, describe the reason for performance of that step and/or expected system response to control manipulation prescribed by that step.

Material Required for Examination Provide FSAR figure 15.2-8 with title block removed.

Question Source: Facility Exam Bank **Question Modification Method:** Direct From Source

Question Source Comments: Vision bank QID# Q56686

Given the following conditions:

- The plant is operating normally at 100 percent power.
- 'B' Reactor Feedwater pump trips.
- Assume NO operator actions taken.

What is the response of the plant and the reason for that response?

a. Full Reactor Recirc Runback. Reduce Recirc loop flow to ensure adequate NPSH to the Recirc Pumps.

b. Intermediate Reactor Recirc Runback. Reduce power to control level.

c. Full Reactor Recirc Runback. Reduce core flow to ensure adequate NPSH to the Jet Pumps.

d. Intermediate Reactor Recirc Runback. Reduce power to prevent power oscillations.

Answer b **Exam Level** R **Cognitive Level** Memory **Facility:** Hope Creek **Exam Date:** 06/17/2003

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 1 **SRO Group** 1 295009K301

295009 Low Reactor Water Level **Record Number** 12

AK3. Knowledge of the reasons for the following responses as they apply to LOW REACTOR WATER LEVEL:

AK3.01 Recirculation pump run back: Plant-Specific 3.2 3.3

Explanation of Answer

Correct. Intermediate Reactor Recirc Runback. Reduce power to control level. The RFP trip runback reduces reactor power and demand on the Feedwater system to within the capability of 2 RFP operation.
Intermediate Reactor Recirc Runback. Reduce power to prevent power oscillations. Incorrect. Wrong reason. Intermediate RB on RPT Trip is to control level.
Full Reactor Recirc Runback. Reduce core flow to ensure adequate NPSH to the Jet Pumps. Incorrect. Wrong Runback. Wrong Reason. Bases of Total Feedwater flow RB.
Full Reactor Recirc Runback. Reduce Recirc loop flow to ensure adequate NPSH to the Recirc Pumps. Incorrect. Wrong Runback Wrong reason. Bases for Suction valve position interlock.

Reference Title

NOH01RECCON-01

Learning Objectives

RECCONE002 From memory, state the purpose of the following Recirculation Flow Control components IAW the Recirculation Flow Control System Lesson Plan:

a. Recirc MG drive motor
b. Fluid Coupler
c. Generator
d. Scoop Tube Positioning Unit
e. Exciter
f. Tachometer
g. Individual pump speed controllers
h. Speed limiters #1 and #2
i. Startup signal generator
j. Error limiting circuit
k. Scoop tube positioning unit
l. Signal Failure Detector
m. Master Speed Controller

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following conditions:

- A LOCA has occurred
- All control rods did not fully insert
- Reactor Power is 3%, and lowering
- Reactor Pressure is 1000 psig, controlled by SRV's
- Reactor Water Level is 0 inches, steady
- Drywell Temperature is 350°F, and rising
- Drywell Pressure is 33 psig, and rising
- Suppression Pool Temperature is 120°F, and rising
- Suppression Pool Level is 80 inches, steady
- Suppression Chamber Pressure is 31.5 psig, and rising
- No operator actions have been taken

Which one of the following is the appropriate action for the conditions above in accordance with the Emergency Operating Procedures?

a. Initiate drywell sprays ONLY.

b. Initiate Emergency Depressurization and Drywell Sprays.

c. Initiate Emergency Depressurization and Suppression Pool Cooling.

d. Initiate Suppression Pool Cooling and Drywell Sprays.

Answer c **Exam Level** S **Cognitive Level** Application **Facility:** Hope Creek **Exam Date:** 06/17/2003

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 2 **SRO Group** 2 295012G406

295012 High Drywell Temperature **Record Number** 13

2.4 Emergency Procedures and Plan

2.4.6 Knowledge symptom based EOP mitigation strategies. 3.1 4.0

Explanation of Answer

Justification: SRO 55.43(4)

CORRECT - Initiate Emergency Depressurization and suppression pool cooling. With SP temperature above 95°F and with SRVs controlling RPV pressure, Suppression Pool Cooling is required. At a Drywell pressure of 33 psig and temperature of 350°F, Drywell Spray is precluded and ED is required IAW DW/T-4 thru DW/T-6.

INCORRECT - Initiate drywell sprays ONLY. At a Drywell pressure of 33 psig and temperature of 350°F, Drywell Spray is precluded.

INCORRECT - Initiate Emergency Depressurization and drywell sprays. At a Drywell pressure of 33 psig and temperature of 350°F, Drywell Spray is precluded.

INCORRECT - Initiate suppression pool cooling and drywell sprays. At a Drywell pressure of 33 psig and temperature of 350°F, Drywell Spray is preclude

Reference Title

HC.OP-EO.ZZ-0102, Step DW/T-3 thru DW/T-6 & DW/P-6

Learning Objectives

EO102PE007 (R) Given any step of the procedure, determine the reason for performance of that step and/or predict expected system response to control manipulations prescribed by that step IAW the Primary Containment Control - Drywell Lesson Plan.

Material Required for Examination	EOP flowcharts
Question Source:	Facility Exam Bank
Question Modification Method:	Editorially Modified
Question Source Comments:	Vision Bank QID Q56006 editorially modified.

Given the following conditions:

- The plant is operating at 100 percent power.
- Turbine Building Chiller AK111 suffers an evaporator tube break.
- All Turbine Building Chilled Water pumps trip on low flow from Freon in the pump casings.
- Attempts to crosstie Chilled Water have failed.
- Drywell temperature and pressure are rising.

Which one of the following actions is required?

a. Manually scram the reactor at 1.5 psig Drywell pressure.

b. Manually scram the reactor at 135 F Drywell temperature.

c. Shutdown Recirc pumps and Drywell Coolers and initiate Drywell Sprays at 1.5 psig Drywell pressure.

d. Shutdown Recirc pumps and Drywell Coolers and initiate Drywell Sprays at 135 F Drywell temperature.

Answer a **Exam Level** B **Cognitive Level** Memory **Facility:** Hope Creek **Exam Date:** 06/17/2003

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 2 **SRO Group** 2 295012K102

295012 High Drywell Temperature **Record Number** 14

AK1. Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL TEMPERATURE:

AK1.02 Reactor power level control 3.1 3.2

Explanation of Answer

Justification:
Manually scram the reactor at 1.5 psig Drywell pressure. Correct. Retainment override step I.a. of HC.OP-AB.CONT-0001 Drywell Pressure.
Manually scram the reactor at 135 F Drywell temperature. HCTS LCO 3.6.1.7 Limit. EOP-102 decision point DW/T-2 which directs SD of RRPs and DW Coolers to spray DW IF under curve DWP-T limit. DW pressure is still too low.
Shutdown Recirc pumps and Drywell Coolers and initiate Drywell Sprays at 1.5 psig Drywell Pressure.Incorrect. DW Pressure too low. Sprays can be inservice at this pressure only if initiated at higher pressure then pressure subsequently lowers.
Shutdown Recirc pumps and Drywell Coolers and initiate Drywell Sprays at 135 F Drywell temperature. Incorrect. Saturation temperature for DWT too low. Step DWT-3 allows spraying the DW between 135 and 340 degF only if DWP falls in SAFE area of curve DWT-P. This corresponds to about 3.2 psig. DWT rising due to a loss of DW cooling, not a LOCA.

Reference Title

HC.OP-AB.CONT-0001

Learning Objectives

EO102PE007	(R) Given any step of the procedure, determine the reason for performance of that step and/or predict expected system response to control manipulations prescribed by that step IAW the Primary Containment Control - Drywell Lesson Plan.
ABCNT5E006	(R) Explain purpose of and the bases for Retainment Override Steps in Irradiated Fuel Damage.

Material Required for Examination None

Question Source: New **Question Modification Method:**

Question Source Comments:

The following plant conditions exist:

- The reactor is at full power.
- Torus Cooling is in operation and average temperature is increasing.
- HPCI testing is in progress.

The required action is to immediately stop HPCI testing if torus temperature exceeds (1), or immediately place the mode switch in shutdown if torus temperature exceeds (2) .

a. (1) 95 F; (2) 110 F

b. (1) 105 F; (2) 110 F

c. (1) 105 F; (2) 120 F

d. (1) 110 F; (2) 120 F

Answer b **Exam Level** R **Cognitive Level** Application **Facility:** Hope Creek **Exam Date:** 06/17/2003

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 2 **SRO Group** 1 295013A102

295013 High Suppression Pool Temperature **Record Number** 15

AA1. Ability to operate and/or monitor the following as they apply to HIGH SUPPRESSION POOL TEMPERATURE:

AA1.02 Systems that add heat to the suppression pool 3.9 3.9

Explanation of Answer

Justification:
(1) 105; (2) 110 Correct. HPCI testing is required to be terminated at 105 degF. If Temp exceeds 110 degF, place the mode switch to shutdown.
(1) 95; (2) 110 Incorrect. 95 is LCO for normal / non-heat adding conditions.
(1) 105; (2) 120 Incorrect.
(1) 110; (2) 120 Incorrect.

Reference Title

HCGS TS 3.6.2.1

HC.OP-IS.BJ-0001

EOP-102 flowchart

Learning Objectives

EOP102E004 From memory, recall the reason why average suppression pool temperature is used for determining the entry condition and subsequent actions IAW the Primary Containment Control - Suppression Pool Lesson Plan.

TECSPCE008 Given specific plant operating conditions, and a copy of the Hope Creek Generating Station Technical Specifications, determine the following:
a. If a Limiting Condition for Operation has been exceeded.
b. If a Limiting Safety System Setting has been reached and/or exceeded.
c. If a Safety Limit has been violated.

Material Required for Examination HCGS Tech Specs section 3.6; EOP Flow charts

Question Source: INPO Exam Bank **Question Modification Method:** Direct From Source

Question Source Comments: 19639 06/14/2001 Fermi 2

A startup is in progress with the following conditions:

- Reactor pressure is stable at 170 psig.
- Two turbine bypass valves are full open.
- Control rods are being withdrawn.
- IRMs are between 30 and 70 on range 8

Which of the following would occur if all turbine bypass valves were to fail closed and why? (Assume NO operator action)

a. The reactor would scram due to high flux.

b. The reactor would scram due to high pressure.

c. Reactor power would increase and stabilize due to the change in coolant temperature.

d. Reactor power would decrease and stabilize due to the change in void fraction.

Answer a **Exam Level** B **Cognitive Level** Comprehension **Facility:** Hope Creek **Exam Date:** 06/17/2003

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 1 **SRO Group** 1 295014K201

295014 Inadvertent Reactivity Addition **Record Number** 16

AK2. Knowledge of the interrelations between INADVERTENT REACTIVITY ADDITION and the following:

AK2.01 RPS 3.9 4.1

Explanation of Answer

The reactor would scram due to high flux. Correct. TBV closure will cause pressure rise and void collapse. Power will rise. With IRMs not ranged, a reactor scram would result.

The reactor would scram due to high pressure. Incorrect. HI-HI IRM flux will trp first.

Reactor power would increase and stabilize due to the change in coolant temperature. Incorrect. Would not stabilize with TBVs failed closed.

Reactor power would decrease and stabilize due to the change in void fraction. Incorrect. Reactor power would increase.

Reference Title

HC.OP-AB.RPV-0005

Learning Objectives

RXINSTE014 (R) Given changes in the following parameters, evaluate the affect on each RPV level indication.
Reactor Pressure
Drywell Temperature
Steam Flow

Material Required for Examination None

Question Source: INPO Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: INPO BANK QID #21232 Dresden 06/14/2002 Modified for Hope Creek.

Given the following conditions:

- The plant was operating at 100% power when the reactor scrammed.

The operator observes the following indications:

- Reactor Pressure: 900 psig
- All Scram valves open
- RWM: All Rods In: NO
  Shutdown: YES
  Rods Not Full In: 040

The reactor is:

a. in a cold shutdown rod configuration with forty control rods at position 02.

b. in a cold shutdown rod configuration with forty control rods out further than 02.

c. only subcritical at the present reactor temperature with forty rods at position 02.

d. only subcritical at the present reactor temperature with forty rods out further than 02.

Answer a **Exam Level** R **Cognitive Level** Comprehension **Facility:** Hope Creek **Exam Date:** 06/17/2003

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 1 **SRO Group** 1 295015K101

295015 Incomplete SCRAM **Record Number** 17

AK1. Knowledge of the operational implications of the following concepts as they apply to INCOMPLETE SCRAM:

AK1.01 Shutdown margin 3.6 3.9

Explanation of Answer Justification:
in a cold shutdown rod configuration with forty control rods at position 02. Correct. Shutdown will be yes if all rods are at 02 or less.
in a cold shutdown rod configuration with forty control rods out further than 02. Incorrect. 02 or less
only subcritical at the present reactor temperature with forty rods at position 02. Incorrect. SDM is assured.
only subcritical at the present reactor temperature with forty rods out further than 02.Incorrect. SDM is assured at 02 or less.

Reference Title

EOP 102

HC.OP-SO.SF-0003 5.4.3

Learning Objectives

EO101AE004	Explain the significance of "Maximum Subcritical Banked Withdrawal Position" and state its value.
RODMINE003	(R) Given a labeled drawing of, or access to, the RWM Operator Display on 10C651, or the RWM Computer Display (in the Computer Room): a. Explain the function of each indicator. b. Assess plant conditions, which will cause the indicator to light or extinguish. c. Determine the effect of each control on the Rod Worth Minimizer. d. Assess plant conditions or permissives required for the control switches/pushbuttons to perform their intended functions.

Material Required for Examination None

Question Source: INPO Exam Bank **Question Modification Method:** Direct From Source

Question Source Comments: 7937 Hatch 03/14/1997.

Given the following conditions:

- Security reports that a tanker in the river has run aground and is leaking a large cloud of green gas vapor.
- The wind is carrying this gas towards the plant.
- Security officers report a strong Chlorine odor outside.

Based on these conditions, place CREF in service with CREF boundry dampers in _________ / __________ Mode and operators must ________________

a. OA; ISOLATE; remain in the Control Room.

b. OA; NORMAL; establish control at the RSP.

c. RECIRC; ISOLATE; remain in the Control Room.

d. RECIRC; NORMAL; establish control at the RSP.

Answer c **Exam Level** B **Cognitive Level** Memory **Facility:** Hope Creek **Exam Date:** 06/17/2003

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 2 **SRO Group** 1 295016K203

295016 Control Room Abandonment **Record Number** 18

AK2. Knowledge of the interrelations between CONTROL ROOM ABANDONMENT and the following:

AK2.03 Control room HVAC 2.9 3.1

Explanation of Answer

Justification:
RECIRC; ISOLATE; remain in the Control Room.. Correct. Retainment override step I of HC.OP-AB.HVAC-0002 Control Room Environment
OA; ISOLATE; remain in the Control Room. Incorrect. Allows gas to enter Control Room Envelope. Mode for High Radiation response.
OA; NORMA; establish control at the RSP Incorrect. Allows gas to enter Control Room Envelope.
RECIRC; NORMAL; establish control at the RSP. Incorrect. Closes off outside air flowpath but dampers are in wrong alignment.

Reference Title

HC.OP-AB.HVAC-0002

Learning Objectives

ABHVC1E004 Explain the reasons for how plant/system parameters respond when implementing HVAC.

Material Required for Examination None

Question Source: New **Question Modification Method:**

Question Source Comments:

Due to a fire in the Control Room console, the Control Room Supervisor orders the Control Room immediately evacuated. The reactor was scrammed remotely.

Which of the following statements describes how a scram is verified in accordance with Shutdown from Outside the Control Room, HC.OP-IO.ZZ-008?

a. HCU accumulator pressure verified to be 950 - 1000 psig at each HCU.

b. SPDS display terminal "Rods Full In" in the TSC.

c. Reactor vessel pressure stable at 920 psig.

d. RPS Backup Scram Air Solenoids verifed de-energized.

Answer b **Exam Level** B **Cognitive Level** Memory **Facility:** Hope Creek **Exam Date:** 06/17/2003

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 2 **SRO Group** 1 295016K301

295016 Control Room Abandonment **Record Number** 19

AK3. Knowledge of the reasons for the following responses as they apply to CONTROL ROOM ABANDONMENT:

AK3.01 Reactor SCRAM 4.1 4.2

Explanation of Answer

Justification:
IAW IO-0008 step 5.1.3 "If the Rx scram was not verified prior to evacuating the Control Room, then verify Rods Full In (SPDS/CRIDS (TSC) or Activity Control Cards OR Other)
SPDS display terminal "Rods Full In" in the TSC. Correct. IAW IO-0008 step 5.1.4
HCU accumulator pressure verified to be 950 - 1000 psig at each HCU. Incorrect. 950 - 1000 psig is still within the normal charged range of an HCU. The USFAR states "Observing the local nitrogen side pressure indicator for each hydraulic control unit scram accumulator for a low (post scram) pressure indication."
RPS Backup Scram Air Solenoids verifed de-energized. Incorrect. Rx pressure at 920 indicates the reactor is at low thermal power level but not necessarily scrammed. The USFAR states "By manually cycling a safety/relief valve from the RSP (after RSP takeover) and observing an appropriate cooldown as indicated by a reduction in steady state reactor pressure following the steam discharge. Pressure indication can be used since pressure and temperature are directly related in a saturated system. If the reactor were critical, pressure and, correspondingly, temperature, would return to approximately their initial values since the reactor would see this evolution as a power transient."
RPS Backup Scram Air Solenoids verifed de-energized. Incorrect BU scram valves are de-energized with the scram reset. If they were energized, that would indicate a reactor scram.

Reference Title

HC.OP-IO.ZZ-0008

UFSAR 7.4.1.4

Learning Objectives

IOP008E002 Determine if all Prerequisites have been met prior to implementation of the SHUTDOWN FROM OUTSIDE THE CONTROL ROOM Integrated Operating Procedure.

Material Required for Examination None

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following conditions:

- A plant startup is in progress following a forced outage.
- The plant has been operating with a known fuel leak.
- The plant scrammed 40 hours ago.
- 'A' Mechanical Vacuum Pump (MVP) is placed in service with the suction valve throttled.
- The Main Condenser Vacuum Breakers are closed.

Which one of the following actions is required by HC.OP-SO.CG-0001, Condenser Air Removal System Operation, if the South Plant Vent (SPV) RMS Effluent reaches the HIGH level?

a. Throttle MVP Suction valve further closed to reduce effluent levels in the SPV.

b. Swap to the standby MVP to reduce effluent levels in the SPV.

c. Stop the MVP to stop release to the SPV.

d. Open the Main Condenser Vacuum Breakers to stop release to the SPV.

Answer c **Exam Level** B **Cognitive Level** Memory **Facility:** Hope Creek **Exam Date:** 06/17/2003

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 2 **SRO Group** 1 295017G132

295017 High Off-Site Release Rate **Record Number** 20

2.1 Conduct of Operations

2.1.32 Ability to explain and apply system limits and precautions. 3.4 3.8

Explanation of Answer

Justification:
Stop the MVP to stop release to the SPV. Correct. Required because the HIGH Setpoint is reached. HC.OP-SO.CG-0001 Caution 5.8.13 .
Throttle MVP Suction valve further closed to reduce effluent levels in the SPV. Incorrect. HC.OP-SO.CG-0001 Caution 5.8.13 states the MVP does not need to be stopped if the MVP suction is throttled until the HIGH alarm setpoint is reached.
Swap to the standby MVP to reduce effluent levels in the SPV. Incorrect. Swapping MVPs will not lower release rate.
Open the Main Condenser Vacuum Breakers to stop release to the SPV. Incorrect. Not required. Would increase effluent flow.

Reference Title

HC.OP-SO.CG-0001 Caution 5.8.13

HC.OP-AB.CONT-0004

Learning Objectives

ABBOP6E007 (R) Explain the bases for Subsequent Actions and the information contained in the Discussion Bases Section of Main Condenser Vacuum.

CNDLEKE001 From memory, summarize/identify the purpose of Condenser Leak Detection IAW the Lesson Plan.

Material Required for Examination None

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following conditions:

- The plant is operating at 100 percent power.
- A large leak has occurred on the Instrument Air header supplying the CRD Scram Air Header.
- The header pressure is lowering rapidly.

At what point is a reactor manual scram required and why?

a. When the first control rod drifts due to Low Accumulator Pressure IAW HC.OP-SO.BF-0002 Individual HCU Operation.

b. When a second control rod drifts due to the Cooling Water Flow Control Valve failing open IAW HC.OP-SO.BF-0001 CRD System Operation.

c. When the first control rod drifts due to its Scram Inlet Valve opening IAW HC.OP-AR.ZZ-0011 Attachment E3 for Control Rod Drift.

d. When a second control rod drifts due to its Scram Outlet Valve opening IAW HC.OP-AB.COMP-0001 Instrument and/or Service Air.

Answer d **Exam Level** S **Cognitive Level** Memory **Facility:** Hope Creek **Exam Date:** 06/17/2003

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 2 **SRO Group** 2 295019A201

295019 Partial or Complete Loss of Instrument Air **Record Number** 21

AA2. Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR:

AA2.01 Instrument air system pressure 3.5 3.6

Explanation of Answer

SRO UNIQUE - RO LEVEL QUESTION.
Justification:
When a second control rod drifts due to its Scram Outlet Valve opening IAW HC.OP-AB.COMP-0001 Instrument and/or Service Air. Correct. On loss of header air pressure, the Scram Outlet valves open. This causes the rods to drift inward. More than one rod drifitng in requires the Mode Switch locked in Shutdown.
When a second control rod drifts due to the Cooling Water Flow Control Valve failing open IAW HC.OP-SO.BF-0001 CRD System Operation. Incorrect. Wrong reason. The Cooling Water Flow control valve fails closed on a loss of air.
When the first control rod drifts due to its Scram Inlet Valve opening IAW HC.OP-AR.ZZ-0011 Attachment E3 for Control Rod Drift. Incorrect. Wrong action. Need more than one rod drifting/scrammed.
When the first control rod drifts due to Low Accumulator Pressure IAW HC.OP-SO.BF-0002 Individual HCU Operation. Incorrect. Wrong reason. Low accumulator pressure is a result of a rod scram, not the cause.

Reference Title

HC.OP-AB.COMP-0001

Learning Objectives

ABCMP1E007 (R) Explain the bases for Subsequent Actions and the information contained in the Discussion Bases Section of Instrument and/or Service Air.

Material Required for Examination None

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following conditions:

- The plant is operating at 100 percent power.
- A complete loss of the service and instrument air systems occurs.
- Operators are trying to restart a compressor as air header pressure lowers.

Which one of the following is the effect on the Condensate/Feedwater System?

a. Secondary Condensate pump Min Flow valves fail closed.

b. SJAE/SPE Bypass Valve PDV-1719 fails open.

c. Primary Condensate pump Min Flow valve HV-1710 fails open.

d. Feedwater heater dump valves fail closed.

Answer b **Exam Level** B **Cognitive Level** Memory **Facility:** Hope Creek **Exam Date:** 06/17/2003

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 2 **SRO Group** 2 295019K207

295019 Partial or Complete Loss of Instrument Air **Record Number** 22

AK2. Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR and the following:

AK2.07 Condensate system 3.2 3.2

Explanation of Answer

Justification:
SJAE/SPE Bypass Valve PDV-1719 fails open. Correct. The bypass valve fails open on loss of air.
Secondary Condensate pump Min Flow valves fail closed. Incorrect. SCP min flows fail open.
Primary Condensate pump Min Flow valve HV-1710 fails open. Incorrect. Motor Operated Valve.
Feedwater heater dump valves fail closed. Cascading drain valves fail close, but high level dump valves fail open.

Reference Title

HC.OP-AB.COMP-0001

Learning Objectives

ABCMP1E004 Explain the reasons for how plant/system parameters respond when implementing Instrument and/or Service Air.

Material Required for Examination None

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following conditions:

- An ATWS occurs from 100 percent power.
- As corrective actions are being taken, the MSIVs inadvertantly isolate from a spurious high steam tunnel temperature signal.
- Other MSIV closure interlocks are clear.
- Visual inspection of the steam tunnel show NO abnormalities.
- Main condenser vacuum is 3 InHgA.
- Reactor coolant activity levels are normal.

Based on these conditions, which one of the following will allow the MSIVs to be re-opened?

a. Reactor power is 10 percent.

b. Suppression pool temperature is rising towards HCTL.

c. RPV Level is less than -129 inches.

d. Emergency depressurization is anticipated.

* **Answer** a **Exam Level** B **Cognitive Level** Application **Facility:** Hope Creek **Exam Date:** 06/17/2003

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 2 **SRO Group** 2 295020K102

295020 Inadvertent Containment Isolation **Record Number** 23

AK1. Knowledge of the operational implications of the following concepts as they apply to INADVERTENT CONTAINMENT ISOLATION:

AK1.02 Power/reactivity control 3.5 3.8

Explanation of Answer

Justification:

Reactor power is 10 percent. Correct. Step RC/P-17 allows reopening if boron injection is required; (>4% power) and Main condenser available; (3 InHgA), and No indication of gross fuel failure or steam line break; (normal activity; NSSSS conditions clear)

Suppression pool temperature is rising towards HCTL. Incorrect. Drives pressure reduction, but does not give permission to open MSIVs.

RPV Level is less than -129 inches. Incorrect. EOP 301 must be performed.

Emergency depressurization is anticipated. - Depressurization through TBV's is not allowed due to the reactor is not shutdown under all conditions without boron.

Reference Title

HC.OP-EO.ZZ-0101A

Learning Objectives

EO101AE008 (R) Given any step of the procedure, explain the reason for performance of that step and/or evaluate the expected system response to control manipulations prescribed by that step.

Material Required for Examination EOP Flowcharts

Question Source: New **Question Modification Method:**

Question Source Comments:

* Answer key changed to both "a" and "b" as correct.

Given the following conditions:

- The plant is operating normally at 100 percent power.
- An inadvertent High Drywell Pressure Core Spray Manual Channel D initiation signal occurs.

What effect does this have on Drywell Cooling?

a. Drywell Cooler fans A1 through H1 trip but can be restarted.

b. Drywell Cooler fans A2 through H2 trip and CANNOT be restarted.

c. Drywell Chilled Water Isolation valves close but all Drywell Cooling fans remain running.

d. Drywell Cooler fans A2 through H2 trip and Chilled Water Isolation valves close.

Answer c **Exam Level** B **Cognitive Level** Memory **Facility:** Hope Creek **Exam Date:** 06/17/2003

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 2 **SRO Group** 2 295020K203

295020 Inadvertent Containment Isolation **Record Number** 24

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

AK2.03 Drywell/containment ventilation/cooling: Plant- Specific 3.1 3.3

Explanation of Answer

Justification:
Drywell Chilled Water Isolation valves close but all Drywell Cooling fans remain running. Correct. Cooler fans powered from A and B channels.
Drywell Cooler fans A2 through H2 trip and Chilled Water Isolation valves close. Incorrect. CS Manual D closes Isolation valves and A2 through H2 fans trip.
Drywell Cooler fans A1 through H1 trip but can be restarted. Incorrect. A2 through H2 fans trip.
Drywell Cooler fans A2 through H2 trip and CANNOT be restarted. Incorrect. Can be restarted if Load Shed breaker to MCC re-closed.

Reference Title

HC.OP-SO.SM-0001

Learning Objectives

DWVENTE008 (R) From memory, summarize the interrelationship between the Drywell Ventilation System and the following systems IAW the Drywell Ventilation System Lesson Plan:
a. Chilled Water System
b. Reactor Auxiliaries Cooling System (RACS)
c. Electrical Power Supply
d. Plant Leak Detection System
e. Process Computer

Material Required for Examination None

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following conditions:

- The plant is in Operational Condition 4 at 180 degF.
- The reactor scrammed on startup from a refueling outage.
- RHR Loop "A" operating in Shutdown Cooling.
- The "B" RHR pump is Cleared & Tagged for motor replacement.
- The "A" RHR pump develops a high vibration and trips on overcurrent.
- BC-HV-F008 has spuriously closed and will NOT reopen.
- HC.OP-AB.RPV-0009, Shutdown Cooling, is entered.

Which of the following will be adequate as an Alternate Decay Heat Removal method for the conditions above?

a. Crosstie "C" or "D" RHR pump for heat removal.

b. Inject with Condensate Transfer and reject with RWCU.

c. Use natural circulation and Drywell coolers for heat removal.

d. Inject with one Core Spray pump from the CST to the RPV.

Answer b **Exam Level** B **Cognitive Level** Comprehension **Facility:** Hope Creek **Exam Date:** 06/17/2003

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 3 **SRO Group** 2 295021K302

295021 Loss of Shutdown Cooling **Record Number** 25

AK3. Knowledge of the reasons for the following responses as they apply to LOSS OF SHUTDOWN COOLING:

AK3.02 Feeding and bleeding reactor vessel 3.3 3.4

Explanation of Answer

Justification
Inject with Condensate Transfer and reject with RWCU. Correct. Feed and bleed will maintain temperatures at low decay heat loads such as at BOL.
Crosstie "C" or "D" RHR pump for heat removal. Incorrect. Need F008 open.
Use natural circulation and Drywell coolers for heat removal. Incorrect. Will remove some decay heat but not enough to be included if RPV-0009 actions.
Inject with one Core Spray pump from the CST to the RPV. Incorrect. Need one loop of Core Spray.

Reference Title

HC.OP-AB.RPV-0009

Learning Objectives

ABRPV9E007 (R) Explain the bases for Subsequent Actions and the information contained in the Discussion Bases Section of Shutdown Cooling.

Material Required for Examination None

Question Source: Facility Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments: Vision Bank QID # Q61332 significantly modified.

The plant is operating at 100% power when a CRD Temperature High alarm is received.

Which one of the following could have caused the CRD high temperature condition?

a. Eroded CRD cooling water orifice.

b. Stabilizing valve failed fully open.

c. CRD pump low discharge pressure.

d. CRD flow control valve failed fully open.

Answer c **Exam Level** B **Cognitive Level** Memory **Facility:** Hope Creek **Exam Date:** 06/17/2003

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 2 **SRO Group** 2 295022K302

295022 Loss of CRD Pumps **Record Number** 26

AK3. Knowledge of the reasons for the following responses as they apply to LOSS OF CRD PUMPS:

AK3.02 CRDM high temperature 2.9 3.1

Explanation of Answer

Justification:

HC.OP-BD.IC-0001 Effects of Loss of CRD regulating Function section 3 "CRD High temperatures can be caused by abnormal drive pressure and any of the following:

1. leaking scram discharge valve
2. low cooling water flow
3. defective thermocouple circuit
4. plugged CRD cooling water orifice

Justification:

CRD pump low discharge pressure.-correct per HC.OP-BD.IC-0001. Would cause low cooling water flow.

Stabilizing valve failed fully open-incorrect- this is the normal position, failure to close would possibly cause hunting of the FCV during rod motion

Eroded CRD cooling water orifice-incorrect opposite effect of plugged orifice low temperatures

CRD flow control valve failed fully open-incorrect high cooling water flow opposite effect lower temperatures

Reference Title

HC.OP-BD.IC-0001

Learning Objectives

ABIC01E007 (R) Explain the bases for Subsequent Actions and the information contained in the Discussion Bases Section of Control Rod.

Material Required for Examination None

Question Source: Facility Exam Bank **Question Modification Method:** Direct From Source

Question Source Comments: Vision Bank QID# Q54263

Given the following conditions:

- The plant is in OPCON 5.
- Core offload is in progress.
- A spent fuel bundle is full up on the main hoist over the core.
- Subsequently, the refuel bridge spotter notices the fuel bundle has become unlatched and has fallen into the vessel.
- The water clarity has degraded significantly.
- A short time later, the following Refuel Floor Rad Monitors alarm:
    - Spent Fuel Pool ARM.
    - New Fuel Criticality ARM.

Based on these conditions, what IMMEDIATE operator action is required?

a. Suspend all refueling operations.

b. Remove the Fuel Pool Cooling System from service to reduce Reactor Building radiation levels.

c. Initiate action to establish Secondary Containment within 1 hour.

d. Determine the location of the dropped bundle, inform the CRS, and evacuate the Refuel Floor.

Answer	a	**Exam Level**	S	**Cognitive Level**	Memory	**Facility:**	Hope Creek	**Exam Date:**	06/17/2003
Tier:	Emergency and Abnormal Plant Evolutions			**RO Group**	3	**SRO Group**	1		295023G449
295023	Refueling Accidents							**Record Number**	27

2.4 Emergency Procedures and Plan

2.4.49 Ability to perform without reference to procedures those actions that require immediate operation of system components and controls. 4.0 4.0

Explanation of Answer

Justification:
SRO 10CFR55.43 (7) Fuel handling facilities and procedures.
SRO 55.43 (6) Procedures and limitations involved in alterations in core configuration.
SRO 55.43 (4) Radiation hazards that may arise during normal and abnormal situations.
Correct- Suspend all refueling operations. AB.CONT-0005 is entered due to the New Fuel Criticality ARM alarm. This is the Immediate Operator action.
Incorrect - Remove the Fuel Pool Cooling System from service to reduce Reactor Building radiation levels. CONT-0005 subsequent action C discusses the increased rad levels in the FPCC System but no direction to remove the system from service is provided.
Incorrect - Initiate action to establish Secondary Containment within 1 hour - Secondary containment is required to be in place during all fuel moves there is no time limit if lost. Step 4.3 requires Secondary Containment to be verified.
Incorrect - Determine the location of the dropped bundle, inform the CRS and evacuate the Refuel Floor. The operator should not wait to determine the location of the dropped bundle..

Reference Title

HC.OP-AB.CONT-0005

Learning Objectives

ABCNT5E003 (R) From memory, recall the Immediate Operator Actions for Irradiated Fuel Damage.

Material Required for Examination None

Question Source:	Facility Exam Bank	Question Modification Method:	Editorially Modified
Question Source Comments:	VISION Bank QID# Q55912		

Which of the following indications will positively identify a criticality event in progress while a fuel bundle is being lowered into the core during refueling operations?

a. Source range monitor spiking repeatedly.

b. A sustained upward trend on the nearest source range instrument to the fuel bundle location.

c. The high refuel floor radiation alarm sounds.

d. Refuel bridge hoist motion interlock activates.

Answer b **Exam Level** B **Cognitive Level** Memory **Facility:** Hope Creek **Exam Date:** 06/17/2003

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 3 **SRO Group** 1 295023K103

295023 Refueling Accidents **Record Number** 28

AK1. Knowledge of the operational implications of the following concepts as they apply to REFUELING ACCIDENTS:

AK1.03 Inadvertent criticality 3.7 4.0

Explanation of Answer

Justification:
A sustained upward trend on the nearest source range instrument to the fuel bundle location. Correct. Responsibility of the Control Room refueling monitor is to monitor for unexpected increasing count rate.
Source range monitor spiking repeatedly. Incorrect. Indications of a detector failure.
The high refuel floor radiation alarm sounds. Incorrect. Would be correct if in New Fuel Vault.
Refuel bridge hoist motion interlock activates. Incorrect. No hoist interlocks will activate as a result of a criticality in the core.

Reference Title

HC.RE-AP.ZZ-0049 3.3.1.D

Learning Objectives

ADMPROE071 From memory State the responsibilities of the following personnel:
a. Refueling SRO.(SRO ONLY)
b. Refueling Bridge Operator
c. Control Room Refuel Monitor IAW NC.NA-AP.ZZ-0049.

Material Required for Examination None

Question Source: INPO Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments: INPO Bank QID # 2147 Quad Cities 10/11/1996 Modified for Hope Creek.

Given the following conditions:

- The reactor has scrammed (all control rods are at position 00) on high drywell pressure.
- Reactor pressure is 35 psig.
- Reactor level is -120 inches rising.
- Suppression pool level is 75 inches.
- Suppression pool temp is 120°F.
- Suppression chamber temp is 100°F.
- Suppression chamber press is 15 psig.
- Drywell temp is 280°F.
- Drywell pressure is 17 psig.

To control the primary containment under these conditions the operator should monitor and control hydrogen concentration in the Supp Chamber and the Drywell and:

a. place one loop of RHR in drywell spray and the other loop of RHR in drywell and suppression chamber spray.

b. place one loop of RHR in suppression pool cooling and the other loop of RHR in drywell and suppression chamber spray.

c. place one loop of RHR in suppression pool cooling and suppression chamber spray and the other loop of RHR in drywell spray.

d. place one loop of RHR in suppression pool cooling and the other loop of RHR in drywell spray, and vent the suppression chamber.

Answer c **Exam Level** B **Cognitive Level** Application **Facility:** Hope Creek **Exam Date:** 06/17/2003

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 1 **SRO Group** 1 295024A201

295024 High Drywell Pressure **Record Number** 29

EA2. Ability to determine and/or interpret the following as they apply to HIGH DRYWELL PRESSURE:

EA2.01 Drywell pressure 4.2 4.4

Explanation of Answer

Justification:
Place one loop of RHR in Suppression pool cooling and suppression chamber spray and the other loop of RHR in drywell spray.-Correct-adequate core cooling is assured, SP temp requires SP cooling/spray, DWSIL curve is satisfied DW Spray is appropriate.
Place one loop of RHR in Suppression pool cooling and the other loop of RHR in drywell and suppression chamber spray-Incorrect- DW spray is always on a loop by itself, if available the second loop is placed in SP cooling/spray to prevent inadvertent bypass of the containment on the operating pump trip.
place one loop of RHR in drywell spray and the other loop of RHR in drywell and suppression chamber spray -incorrect- never place both loops in DW spray this may exceed the makeup capacity of the vacuum breakers and draw the containment negative.
Place one loop of RHR in Suppression pool cooling and the other loop of RHR in drywell spray, and vent the suppression chamber-incorrect- venting the containment is only requires if pressure cannot be maintained below the design limit of 65 psig, and only after attempts top lower pressure with DW spray.

Reference Title

HC.OP-EO-ZZ-0102

Learning Objectives	
EO102PE007	(R) Given any step of the procedure, determine the reason for performance of that step and/or predict expected system response to control manipulations prescribed by that step IAW the Primary Containment Control - Drywell Lesson Plan.

Material Required for Examination EOP Flowcharts

Question Source: Facility Exam Bank

Question Modification Method: Direct From Source

Question Source Comments: VISION Bank QID# Q56010

Given the following conditions:

- A LOCA has resulted from a seismic event.
- Reactor water level is -20 inches and rising.
- Reactor pressure is 850 psig and slowly lowering.
- Drywell Pressure is 31 psig and slowly rising.
- Drywell temp is 275 °F and slowly rising.
- Suppression Chamber pressure is 30 psig and slowly rising.
- Suppression Pool water level is 77 inches.
- 1BD417 1E 125 VDC distribution panel is de-energized.
- "A" RHR Loop is in Drywell Spray.
- The Main Condenser is NOT available.
- All control rods are full in.

Based on the above conditions, when is Emergency Depressurization of the reactor required?

a. Immediately using all Turbine Bypass Valves.

b. Immediately using 5 ADS valves.

c. When Drywell Temp reaches 310F using 5 ADS valves.

d. When Drywell Press reaches 35 psig using all Turbine Bypass Valves.

Answer b **Exam Level** S **Cognitive Level** Application **Facility:** Hope Creek **Exam Date:** 06/17/2003

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 1 **SRO Group** 1 295024A204

295024 High Drywell Pressure **Record Number** 30

EA2. Ability to determine and/or interpret the following as they apply to HIGH DRYWELL PRESSURE:

EA2.04 Suppression chamber pressure: Plant-Specific 3.9 3.9

Explanation of Answer

JUSTIFICATION:
SRO 55.43(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergengcy situations.
CORRECT - Immediately using 5 ADS valves. Emergency de-pressurization must occur now for exceeding the PSP curve.
INCORRECT - Immediately using Turbine Bypass Valves. Emergency de-pressurization can no longer be anticipated. Emergency de-pressurization must occur now for exceeding the PSP curve.
INCORRECT - When Drywell Temp reaches 310°F using 5 ADS valves. Emergency de-pressurization must occur now before exceeding the PSP curve. Emergency de-pressurization for high Drywell temperature does not occur until 340°F.
INCORRECT - When Drywell Press reaches 35 psig using Turbine Bypass Valves. Emergency de-pressurization can no longer be anticipated. Emergency de-pressurization must occur now for exceeding the PSP curve.

Reference Title

EOP Flow chart 102

Learning Objectives

EO102PE006 (R) Given plant conditions and access to the following curves determine the region of acceptable operation and explain the bases for the curve IAW the Primary Containment Control - Drywell Lesson Plan:
a. Drywell Spray Initiation Limit
b. Pressure Suppression Pressure

EOP102E009 (R) Given any step of the procedure, determine the reason for performance of that step and/or predict expected system response to control manipulations prescribed by that step IAW the Primary Containment Control - Suppression Pool Lesson Plan.

Material Required for Examination	EOP Flow chart 102
Question Source:	Facility Exam Bank
Question Modification Method:	Editorially Modified
Question Source Comments:	VISION Bank QID# Q56157

EOP 102 PRIMARY CONTAINMENT CONTROL, has an override that states:

IF: THEN:

Drwl sprays have been initiated Before Drwl press reaches 0 psig. Terminate drwl sprays.

Supp chamber sprays have been initiated Before suppression chamber press reaches 0 psig. Terminate suppression chamber sprays.

Which of the following statements describes the reason for this requirement?

a. 0 psig drywell pressure ensures a drywell temperature below 212F, therefore there is NO need to continue drywell sprays.

b. It makes one more RHR loop available as soon as possible for injection into the reactor pressure vessel.

c. It prevents drawing a negative pressure in the containment, which would open the vacuum breakers and draw air into the containment.

d. This action ensures that the drywell structure will NOT endure excessive thermal stresses due to rapid cooldown.

Answer c **Exam Level** R **Cognitive Level** Memory **Facility:** Hope Creek **Exam Date:** 06/17/2003

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 1 **SRO Group** 1 295024G418

295024 High Drywell Pressure **Record Number** 31

2.4 Emergency Procedures and Plan

2.4.18 Knowledge of the specific bases for EOPs. 2.7 3.6

Explanation of Answer

Justification:
It prevents drawing a negative pressure in the containment, which would open the vacuum breakers and draw air into the containment. Correct. IAW bases for ste p PCC-1, a negative pressure will open the SC to RB vacuum breakers and de-inert containment.
0 psig drywell pressure ensures a drywell temperature below 212F, therefore there is NO need to continue drywell sprays. Incorrect.
It makes one more RHR loop available as soon as possible for injection into the reactor pressure vessel. Incorrect. Concern is de-inerting containment.
This action ensures that the drywell structure will NOT endure excessive thermal stresses due to rapid cooldown. Incorrect. Ensures containment is not de-inerted.

Reference Title

EOP 102 Bases for step PCC-1

Learning Objectives

EO101PE008	(R) Given any step of the procedure, describe the reason for performance of that step and/or expected system response to control manipulations prescribed by that step.

Material Required for Examination None

Question Source: INPO Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: INPO BANK QID # 20450 Quad Cities 08/13/2001 modified for Hope Creek.

Following a reactor scram and Main Steam Isolation Valve closure, reactor pressure reaches 1050 psig.

Which of the following describes the response of the "H" and "P" Safety Relief Valves (SRV) in the Low-Low Set mode of operation for the given conditions?

a. The "P" SRV opens, which actuates low-low set causing the "H" SRV to open and both valves will control pressure at new, lower operating setpoints.

b. The "H" and "P" SRVs both open and both valves will control pressure at the same opening setpoints and new, lower closing setpoints.

c. The "H" SRV opens and the "H" and "P" SRVs control pressure together at new operating setpoints.

d. The "H" and "P" SRVs both open and the "H" SRV will control pressure at new operating setpoints with the "P" SRV operating as needed at slightly higher than "H" operating setpoints.

Answer d **Exam Level** B **Cognitive Level** Comprehension **Facility:** Hope Creek **Exam Date:** 06/17/2003

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 1 **SRO Group** 1 295025A103

295025 High Reactor Pressure **Record Number** 32

EA1. Ability to operate and/or monitor the following as they apply to HIGH REACTOR PRESSURE:

EA1.03 Safety/relief valves: Plant-Specific 4.4 4.4

Explanation of Answer

Justification:
Correct answer: The "H" and "P" SRVs both open and the "H" SRV will control pressure at new operating setpoints with the "P" SRV operating as needed at slightly higher than "H" operating setpoints.
The Low-Low set function arms at 1047 psig. Once armed the H will lift at 1017 for all subsequent lifts, and the P will remain the same at 1047 psig. The H will close at 905 psig and the P will reclose at 935 psig.

The following distractors are incorrect:
The "H" and "P" SRVs both open and both valves will control pressure at the same opening setpoints and new, lower closing setpoints. Incorrect. The Low-Low set function arms at 1047 psig. Once armed the H will lift at 1017 for all subsequent lifts, and the P will remain the same at 1047 psig. The H will close at 905 psig and the P will reclose at 935 psig.

The "H" SRV opens and the "H" and "P" SRVs control pressure together at new operating setpoints. Incorrect. The Low-Low set function arms at 1047 psig. Once armed the H will lift at 1017 for all subsequent lifts, and the P will remain the same at 1047 psig. The H will close at 905 psig and the P will reclose at 935 psig.

The "P" SRV opens, which actuates low-low set causing the "H" SRV to open and both valves will control pressure at new, lower operating setpoints. Incorrect. The Low-Low set function arms at 1047 psig. Once armed the H will lift at 1017 for all subsequent lifts, and the P will remain the same at 1047 psig. The H will close at 905 psig and the P will reclose at 935 psi.

Reference Title

HC.OP-SO.SN-0001

Learning Objectives

MSTEAME003 (R) Concerning the safety relief valves; summarize, list or identify the following.
a. The number and type of SRV's at Hope Creek.
b. Which SRV's have an ADS function.

c. Power supplies to the SRV solenoids.
d. Which SRV's can be operated remotely and the location from which each of these valves can be operated.
e. Purpose of the low-low set function and determine which SRV's are used for this function.
f. Determine the sequence of operation of the low-low set SRV's including arming setpoints, lift points and reclose setpoints.

Material Required for Examination None

Question Source: Facility Exam Bank

Question Modification Method: Direct From Source

Question Source Comments: Vision Bank QID# Q53505

Given the following conditions:

- The plant is in an ATWS condition.
- MSIV's are closed, pressure control band is 800 to 900 psig using SRV's.
- APRM's read 10%.
- Manual rod insertion is in progress.
- Suppression Chamber pressure is 2.8 psig.
- Reactor pressure is 850 psig.
- Suppression pool water temperature is 195 degrees F.

Based on these conditions, which of the following require an immediate reactor pressure reduction?

a. Reactor pressure stabilizes at 900 psig.

b. Suppression Pool level reaches 120 inches and is rising.

c. Suppression Pool water temperature reaches 205 degrees F.

d. Suppression Pool level reaches 70 inches and is lowering.

Answer c **Exam Level** B **Cognitive Level** Application **Facility:** Hope Creek **Exam Date:** 06/17/2003

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 2 **SRO Group** 1 295026K102

295026 Suppression Pool High Water Temperature **Record Number** 33

EK1. Knowledge of the operational implications of the following concepts as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE:

EK1.02 Steam condensation 3.5 3.8

Explanation of Answer

Justification:
Suppression Pool water temperature reaches 205 degrees F. Correct. IAW EOP 102 Curve SPT-P with RPV pressure at the upper end of the band at 900 psig, you would be in the action required area of the curve. The required action is to reduce pressure to get below the curve IAW step SP/T-7.
Suppression Pool level reaches 120 inches and is rising. Incorrect. Still below the action level of 124 inches.
Reactor pressure stabilizes at 900 psig. Incorrect. Pressure reductionnot required at 900 psig and 195 DegF SPT.
Suppression Pool level reaches 70 inches and is lowering. Not required until 38.5 inches SPL

Reference Title

EOP 102 Step SP/T-7

Learning Objectives

EOP102E008 (R) Given plant conditions and access to the Heat Capacity Temperature Limit curve, determine the region of acceptable operation and explain the bases for the curve IAW the Primary Containment Control - Suppression Pool Lesson Plan.

EOP102E009 (R) Given any step of the procedure, determine the reason for performance of that step and/or predict expected system response to control manipulations prescribed by that step IAW the Primary Containment Control - Suppression Pool Lesson Plan.

Material Required for Examination EOP Flowcharts

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following conditions:

- Drywell pressure is 7.3 psig and rising.
- Rx water level is -40 inches.
- Rx pressure 1008 psig and lowering.
- 67 control rods NOT Full-In.
- Suppression pool temperature 129 deg F and increasing.
- MSIVs are closed.
- Suppression pool water level is 84 inches.

Which one of the following EOP actions is required for these conditions and why?

a. Inject SBLC to prevent Suppression Chamber water temperature from exceeding the Heat Capacity Temerature Limit.

b. Spray Drywell to prevent SRVs from exceeding the Suppression Chamber Dynamic Load Limit.

c. Maintain water level +12.5" to +54" to prevent large power swings on the reactor core.

d. Emergency Depressurize to prevent Drywell pressure from exceeding Primary Containment Pressure Limit.

Answer a **Exam Level** B **Cognitive Level** Comprehension **Facility:** Hope Creek **Exam Date:** 06/17/2003

Tier: Emergency and Abnormal Plant Evolutions **RO Group** 2 **SRO Group** 1 295026K304

295026 Suppression Pool High Water Temperature **Record Number** 34

EK3. Knowledge of the reasons for the following responses as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE:

EK3.04 SBLC injection 3.7 4.1

Explanation of Answer

Justification:
Inject SBLC to prevent Suppression Chamber water temperature from exceeding the Heat Capacity Temerature Limit. Correct. Bases of Boron Injection Initiation Temperature limit.
Spray Drywell to prevent SRVs from exceeding the Suppression Chamber Dynamic Load Limit. Incorrect. Wrong limit.
Maintain water level +12.5" to +54" to prevent large power swings on the reactor core. Incorrect. Level must be lowered.
Emergency Depressurize to prevent Drywell pressure from exceeding Primary Containment Pressure Limit. Incorrect. Drywell should be sprayed. ED action to be taken if sprays not effective.

Reference Title

EOP 101A bases BIIT

Learning Objectives

EOP102E009 (R) Given any step of the procedure, determine the reason for performance of that step and/or predict expected system response to control manipulations prescribed by that step IAW the Primary Containment Control - Suppression Pool Lesson Plan.

Material Required for Examination EOP Flowcharts

Question Source: INPO Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: INPO Bank QID # 15341 08/23/1999 Monticello modified for Hope Creek.

Given the following conditions:

- The plant is several hours into a LOCA.
- HPCI automatically initiated and then subsequently tripped on low oil pressure.
- A & B RHR loops are NOT available.
- All other available ECCS are injecting.
- Drywell pressure is 64.4 psig and rising.
- HPCI Pump suction pressure is 73 psig.
- SP level indication is failed.
- SP temperature is 175°F.

What is containment water level and based on that level, which of the following actions are required?

a. 21.2 ft; Vent the Suppression Pool.

b. 22.0 ft; Vent the Suppression Pool.

c. 21.2.ft; Vent the Drywell.

d. 22.0 ft; Vent the Drywell.

DELETED

Answer	b	Exam Level	S	Cognitive Level	Application	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Emergency and Abnormal Plant Evolutions	RO Group	2	SRO Group	2				295029A203
295029	High Suppression Pool Water Level							Record Number	35

EA2. Ability to determine and/or interpret the following as they apply to HIGH SUPPRESSION POOL WATER LEVEL:

EA2.03 Drywell/containment water level 3.4 3.5

Explanation of Answer

Should provide the students with EOP-102
JUSTIFICATION: SRO 55.43(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency conditions.
Containment Level = [(HPCI Press - DW Press) 2.3 ft/psi] + 2.2ft.
CORRECT - 22.0'. [(73.0 - 64.4) 2.3} + 2.2. 22.0 ft containment level (263.8 inches) or equivalent to 169.8 inches indicated torus level. EOP-102 Step DWP-12 requires Venting the Suppression Pool if <180 inches in the torus.
INCORRECT - 21.2'. Vent the Suppression Pool. [(73.0 - 64.4)2.2]+2.3 (2.2 and 2.3 transposed in formula) Wrong value, correct action.
INCORRECT - 21.2'; Vent the Drywell. (2.2 and 2.3 transposed in formula) Wrong value; incorrect action based on 254.6 inches (94 inches must be subtracted to provided equivalent Suppression Pool water level.)
INCORRECT - 22.0 ft; Vent the Drywell. Correct value; incorrect action based on 94 inches must be subtracted to provided equivalent Suppression Pool water level.

Reference Title

HC.OP-EO.ZZ-0102, Step SP/L-13

Learning Objectives

EOP102E011	(R) Given the formula for calculating containment water level and corresponding values, calculate containment water level IAW the Primary Containment Control - Suppression Pool Lesson Plan.
EOP102E009	(R) Given any step of the procedure, determine the reason for performance of that step and/or predict expected system response to control manipulations prescribed by that step IAW the Primary Containment Control - Suppression Pool Lesson Plan.

Material Required for Examination	EOP Flowcharts

Question Source:	New	Question Modification Method:	
Question Source Comments:			

Given the following conditions:

- A LOCA has occurred.
- RPV water level is stabilized above TAF.
- Suppression Chamber water level is 60 inches and lowering.

Which one of the following correctly fills in the blanks describing the ALTERNATE Core Spray Loop to be used for Suppression Chamber Makeup and the prerequisite Suppression Chamber pressure?

Core Spray Loop ____________ is the ALTERNATE Makeup path to be used only if Suppression Chamber pressure is ____________ 20 psig.

a. A; above

b. A; below

c. B; above

d. B; below

Answer	b	Exam Level	B	Cognitive Level	Memory	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Emergency and Abnormal Plant Evolutions		RO Group	2	SRO Group	1			295030A106
295030	Low Suppression Pool Water Level							Record Number	36

EA1. Ability to operate and/or monitor the following as they apply to LOW SUPPRESSION POOL WATER LEVEL:

EA1.06 Condensate storage and transfer (make-up to the suppression pool): Plant-Specific | 3.4 | 3.4

Explanation of Answer

Justification -
A; Below. Correct IAW HC.OP-EO.ZZ-0315, Suppression Chamber pressure must be less than 20 psig to implement M/U from the CST via Core Spray. A CS Loop is the Alternate loop.
A; Above. Incorrect. A is prefered loop.
B; Above. Incorrect. Only used below 20 psig.
B; Below. Incorrect. B is prefered.

Reference Title

HC.OP-EO.ZZ-0315 2.2.2

Learning Objectives

EOP300E004	From memory, describe any/all flow paths established by the performance of each of the 300 series Emergency Operating procedures.
EOP300E002	Explain the basis/reason for all prerequisites, precautions, and limitations of each of the 300 series Emergency Operating Procedures.

Material Required for Examination None

Question Source: New **Question Modification Method:**

Question Source Comments:

Which one of the following correctly describes the Technical Specification bases for the Suppression Pool low water level limit?

a. This limit ensures adequate SRV T-Quencher submergance during Emergency Depressurization.

b. This limit ensure adequate water volume is available based on NPSH and vortex prevention.

c. This limit prevents exceeding the Suppression Pool design temperature limit during a DBA LOCA.

d. This limit prevents exceeding the Suppression Pool design pressure limit during a DBA LOCA.

Answer	d	Exam Level	S	Cognitive Level	Memory	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Emergency and Abnormal Plant Evolutions	RO Group	2	SRO Group	1				295030G225
295030	Low Suppression Pool Water Level							Record Number	37
2.2	Equipment Control								
2.2.25	Knowledge of bases in technical specifications for limiting conditions for operations and safety limits.							2.5	3.7

Explanation of Answer

Justification:
10CFR55.43 (2) Facility operating limitations in the Technical Specifications and their bases.
This limit prevents exceeding the Suppression Pool design pressure limit during a DBA LOCA. Correct. Bases for SP lower water level limit IAW HCGS TS 3.6.
This limit ensures adequate SRV T-Quencher submergance during Emergency Depressurization. Incorrect. Plausible but incorrect bases.
This limit ensure adequate water volume is available based on NPSH and vortex prevention. Incorrect. Bases for EOP Caution 2 Limits at 0 inches of SPL.
This limit prevents exceeding the Suppression Pool design temperature limit during a DBA LOCA. Incorrect. Plausible but incorrect bases.

Reference Title

HCGS TS Bases 3/4.6.2

Learning Objectives

PRICONE009 (R) Given a Scenario of applicable operating conditions and access to technical specifications:
a. Select those sections which are applicable to the Primary Containment Structure IAW HCGS technical specifications.
b. Evaluate Primary Containment Structure operability and determine required actions based upon system operability IAW HCGS technical specifications. (SRO/STA ONLY)
c. Explain the bases for those technical specification items associated with the Primary Containment Structure IAW HCGS technical specifications.

Material Required for Examination None

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following conditions:

- A Station Blackout occurred.
- B EDG is running.
- B RHR is in Suppression Pool Cooling at 10,000 gpm.
- HPCI and RCIC have tripped on Low Steam Inlet pressure.
- Suppression Pool Temperature is 225F rising slowly.
- RPV water Level is –150 inches and lowering slowly.
- Suppression Chamber pressure is 5 psig.
- Drywell temperature is 310F rising slowly.
- Drywell pressure is 5 psig.
- Suppression Pool water level just reached 0 inches.
- NO other ECCS is running.

Which one of the following actions is required?

a. Open 5 ADS SRVs to emergency depressurize the RPV.

b. Lower B RHR SP Cooling flow to 9000 gpm.

c. Trip B RHR pump immediately.

d. Re-align B RHR for LPCI mode at rated flow.

Answer	d	Exam Level	B	Cognitive Level	Application	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Emergency and Abnormal Plant Evolutions		RO Group	1	SRO Group	1			295031A101
295031	Reactor Low Water Level							Record Number	38

EA1. Ability to operate and/or monitor the following as they apply to REACTOR LOW WATER LEVEL:

EA1.01 Low pressure coolant injection (RHR): Plant-Specific 4.4 4.4

Explanation of Answer

Justification:
Re-align B RHR for LPCI mode at rated flow. Correct. EOP - 101 step ALC-2 requires LPCI started and maximize injection flow to the RPV and ignore NPSH limits.
Lower B RHR SP Cooling flow to 5000 gpm. Incorrect. Would be correct if remaining in SPC.
Trip B RHR pump immediately. Incorrect- proper action for non ECCS pump on inadequate NPSH.
Open 5 ADS SRVs to emergency depressurize the RPV. Incorrect. EOP-202 step RF-2 requires SP level above 0 inches, other means are used to depressurize.

Reference Title

HC.OP-EO.ZZ-010 ALC-2

Learning Objectives

EO101LE006	(R) Given any step of the procedure, describe the reason for performance of that step and/or expected system response to control manipulation prescribed by that step.
EO101LE008	(R) Explain the significance of "Minimum Steam Cooling RPV Water Level " and state its value.

Material Required for Examination EOP Flowcharts

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following conditions:

- Drywell pressure is 4.5 psig.
- RPV level is -45 inches and is being intentionally lowered.
- Many control rods remain at their original positions.
- SLC, CRD, and RCIC are injecting.
- Reactor power is 7 percent.
- SRVs are cycling on Low Low Set.
- A and B RHR Loops are in Suppression Pool Cooling.
- Suppression Pool temperature is 112 F.

Which one of the following conditions permits RPV water level to be stabilized between -190 and the current RPV level when that condition is achieved?

a. All SRVs remain closed.

b. Reactor power reaches 3 percent.

c. Suppression Pool Temperature lowers to 108 F.

d. RPV level reaches -50 inches.

Answer	b	Exam Level	S	Cognitive Level	Application	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Emergency and Abnormal Plant Evolutions		RO Group	1	SRO Group	1			295031A202
295031	Reactor Low Water Level							Record Number	39

EA2. Ability to determine and/or interpret the following as they apply to REACTOR LOW WATER LEVEL:

EA2.02 Reactor power 4.0 4.2

Explanation of Answer

Justification:
55.43(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. This item test SRO ability to resolve the question by first choosing the applicable EOP Flowchart and correctly applying the stem conditions to that flowchart.
Correct. Reactor power reaches 3 percent. The conditions provided in the stem require terminate and prevent injection and RPV level reduction to lower power. The conditions require EOP-101A Step LP-14 implemented for level reduction. If reactor power reaches 3 percent from 7percent, EOP-101A Step LP-14 allows level reduction to be stopped. Step LP-15 then allows RPV level to be maintained between that level and -190 inches.
Incorrect- All SRVs remain closed. Would be correct if Drywell Pressure was below 1.68 psig.
Incorrect- Suppression Pool Temperature lowers to 108 F. With Stem conditions of 7 percent power, Step LP-11 is answered YES. This requires lowering level until power is less than 4 percent. You can not back up and change the answer to LP-11 using the retainment step LP-6. Plausible misconception.
Incorrect. RPV level reaches -50 inches. Based on stem conditions, Steps LP-11 and LP-12 must be answered YES. This bypasses lowering level to -50 inches.

Reference Title

HC.OP-EO.ZZ-0101A

Learning Objectives

EO101AE008	(R) Given any step of the procedure, explain the reason for performance of that step and/or evaluate the expected system response to control manipulations prescribed by that step.

Material Required for Examination	EOP 101 Flowchart

Question Source:	New	Question Modification Method:	

Question Source Comments:	

Given the following conditions:

- The plant is operating at rated power.
- Control Room Overhead alarms are received:

  - B1-A4 HPCI TURBINE TRIP
  - D3-A1 HPCI/RHR A LEAK TEMP HI

- When the operator checks the HPCI panel, HPCI inboard and outboard steam line isolation valves are stroking closed.
- HPCI turbine was NOT running at the time.
- HPCI Steam line pressure is 900 psig and lowering.

Which one of the following would cause this isolation?

a. HPCI Pipe Chase High Temperature.

b. HPCI Steam Line Low Supply Pressure.

c. HPCI Pump Room High Temperature.

d. HPCI Steam Line High Differential Pressure.

Answer	c	Exam Level	R	Cognitive Level	Comprehension	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Emergency and Abnormal Plant Evolutions	RO Group	3	SRO Group	2				295032A101
295032	High Secondary Containment Area Temperature							Record Number	40

EA1. Ability to operate and/or monitor the following as they apply to HIGH SECONDARY CONTAINMENT AREA TEMPERATURE:

EA1.01 Area temperature monitoring system 3.6 3.7

Explanation of Answer

HPCI Equipment Room High Temp. Correct. Isolates immediately. Causes D3-A1 annunciator when tripped which in turn causes B1-A4 annunciator.
HPCI Steam Line Low Supply Pressure. Incorrect. Isolates at 100 psig. Pressure is 900 psig.
HPCI Pipe Chase High Temp. Incorrect. Isolates after 15 minute time delay. Would also cause annuciator D3-B1 HPCI STM LK ISLN TIMER INITIATED.
HPCI Steam Line High Differential Pressure. Incorrect. Would cause B1-A5 annunciator, HPCI STEAM LINE DIFF PRESSURE HI.

Reference Title

HC.OP-AR.ZZ-0014 Attachment A1

Learning Objectives

EOP103E003	(R) Define the term "Maximum Safe Operating Temperature".
EOP103E006	(R) Given any step in the procedure, describe the reason for performance of that step and/or expected system response to control manipulations prescribed by the step.

Material Required for Examination EOP Flowcharts

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following conditions:

- The plant is operating at 100 percent power.
- RACS RMS levels have begun to rise.
- Reactor Building backround radiation levels in the vicinity of RACS piping are also rising.

Which of the following would be the cause of rising RACS RMS readings?

a. Tube rupture in a Reactor Recirc Pump Seal Cooler Heat Exchanger.

b. Drywell Equipment Drain Sump Cooler leak.

c. Reactor Building Equipment Drain Sump Cooler leak.

d. Tube rupture in RWCU Regenerative Heat Exchanger.

Answer	a	**Exam Level**	B	**Cognitive Level**	Comprehension	**Facility:**	Hope Creek	**Exam Date:**	06/17/2003
Tier:	Emergency and Abnormal Plant Evolutions	**RO Group**	2	**SRO Group**	2				295033A102
295033	High Secondary Containment Area Radiation Levels							**Record Number**	41

EA1. Ability to operate and/or monitor the following as they apply to HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS:

EA1.02 Process radiation monitoring system 3.7 3.8

Explanation of Answer

Justification
Drywell Equipment Drain Sump Cooler leak. Incorrect- DWEDS Cooler cooled by Chilled Water. Can be cooled by RACS if cross-tied but RACS would then leak into the sump.
- Reactor Building Equipment Drain Sump Cooler leak.-Incorrect- Rx Bldg. Equipment Drain Sump Cooler at lower pressure than RACS. RACS would leak into the sump.
Tube rupture in RWCU Regenerative Heat Exchanger.–Incorrect- The RWCU regenerative heat exchanger is not cooled by RACS/only NRHX.
Tube rupture Reactor Recirc Pump Seal Cooler Heat Exchanger. - Correct. Reactor coolant from the seal area will leak into the RACS system and cause RMS to rise.

Reference Title

HC.OP-AP.SP-0001 Attachment 15

Learning Objectives

EOP103E004	(R) Define the term "Maximum Safe Operating Radiation Level".
EOP103E006	(R) Given any step in the procedure, describe the reason for performance of that step and/or expected system response to control manipulations prescribed by the step.

Material Required for Examination None

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following conditions:

- The plant is operating at 100 percent power.
- Overhead annunciator E6-C5 "RBVS & WING AREA HVAC PNL 10C382" alarms.
- The Reactor Operator reports Reactor Building Differential Pressure is negative at 0.25 inches water gauge.

Which one of the following actions is required?

a. Start another Reactor Building Supply fan IAW HC.OP-SO.GR-0001 Reactor Building Ventilation.

b. Place FRVS in service IAW HC.OP-AB.ZZ-0001 Transient Plant Conditions.

c. Isolate RBVS Isolation Dampers IAW HC.OP-SO.SM-0001 Isolation System Operation.

d. Place FRVS in service IAW HC.OP-SO.GU-0001 FRVS Operation.

Answer	d	Exam Level	S	Cognitive Level	Memory	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Emergency and Abnormal Plant Evolutions	RO Group	3	SRO Group	2				295035G132
295035	Secondary Containment High Differential Pressure							Record Number	42
2.1	Conduct of Operations								
2.1.32	Ability to explain and apply system limits and precautions.								3.4 3.8

Explanation of Answer

Justification:
SRO 10CRF 55.43 (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
Place FRVS in service IAW HC.OP-SO.GU-0001 FRVS Operation. Correct. FRVS is required by AB-CONT-0003 because RB DP is less than the required 0.30 inches WC.
Place FRVS in service using HC.OP-AB.ZZ-0001 Transient Conditions. Incorrect. AB-ZZ-0001 does not provide direction for starting FRVS.
Isolate RBVS Isolation Dampers IAW HC.OP-SO.SM-0001 Isolation System Operation. Incorrect. Isolations performed with this procedure are performed to isolate system breaches.
Start another Reactor Building Supply fan IAW HC.OP-SO.GR-0001 Reactor Building Ventilation. Incorrect. Starting a supply fan would aggrevate the problem.

Reference Title
HC.OP-AB.CONT-0003
HC.OP-SO.GU-00001

Learning Objectives	
ABCNT3E001	Recognize abnormal indications/alarms and/or procedural requirements for implementing Reactor Building Integrity.

Material Required for Examination	None		
Question Source:	New	Question Modification Method:	
Question Source Comments:			

Given the following conditions:

- The reactor is operating at 100% power.
- Annunciator B1-B3 (RCIC PUMP ROOM FLOODED) alarms with the following alarm message presented on the CRIDS display: D2887 RCIC PUMP RM 4110 LSH 4151-1 HI.
- An investigation reveals that Reactor Building Floor Drain Sump pumps have been running continuously for 20 minutes.
- The Reactor Building Operator reports the RCIC, B and D RHR Pump rooms have about 6 inches of water on the floor when he checked the elevation.
- CST level is lowering.

In addition to running the sump pumps, which of the following action(s), if any, is required by EOP 103/4?

I --- Isolate RCIC
II -- Immediately commence a normal reactor shutdown
III -- Runback reactor recirculation and manually scram the reactor
IV - Emergency depressurize the reactor

a. I - ONLY

b. II - ONLY * Answer key changed to reflect correct answer is "b"

c. I and II

d. I, III, and IV

*Answer c | Exam Level B | Cognitive Level Application | Facility: Hope Creek | Exam Date: 06/17/2003

Tier: Emergency and Abnormal Plant Evolutions | RO Group 3 | SRO Group 2 | 295036K303

295036 Secondary Containment High Sump/Area Water Level | Record Number 43

EK3. Knowledge of the reasons for the following responses as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL:

EK3.03 Isolating affected systems 3.5 3.6

Explanation of Answer

Justification:
I, III, and IV. Incorrect. Normal shutdown not scram and ED. Would apply if reactor coolant leak.
I and II. Correct. EOP-103/4 Step RB-15 The leak is not reactor coolant discharging into the area. Isolate RCIC and commence a normal shutdown.
II - ONLY. Incorrect. Requires RCIC Isolation.
I - ONLY. Incorrect. Required normal shutdown.

Reference Title

HC.OP-EO.ZZ-0103/4

Learning Objectives

EOP103E006	(R) Given any step in the procedure, describe the reason for performance of that step and/or expected system response to control manipulations prescribed by the step.
EOP103E002	Given a set of plant conditions, analyze and determine if entry conditions into HC.OP-EO.ZZ-0103/4 exists.
EOP103E005	(R) Define the term "Maximum Safe Floor Level".

Material Required for Examination EOP-103/4 flow chart

Question Source:	Facility Exam Bank	Question Modification Method:	Significantly Modified
Question Source Comments:	VISION Bank QID #Q68864 Significantly Modified		

Given the following conditions:

- An ATWS occurred from 100 percent power.
- All immediate operator actions have been completed.
- At 1420 hrs, both SLC pumps are started and the SLC Tank Low Level Computer point alarm is received.
- "B" SLC Pump has tripped immediately after start.

Assuming the remaining SLC pump delivers Tech Spec minimum flow rate for the next 90 minutes, which one of the following actions are required?

a. Continue SLC pump operation and raise reactor water level.

b. Continue SLC pump operation and begin reactor cooldown.

c. Verify the "A" SLC pump is tripped and continue rod insertion.

d. Verify the "A" SLC pump is tripped and exit EOP-101A.

Answer	b	Exam Level	B	Cognitive Level	Application	Facility:	Hope Creek	Exam Date: 06/17/2003
Tier:	Emergency and Abnormal Plant Evolutions		RO Group	1	SRO Group	1		295037A203
295037	SCRAM Condition Present and Reactor Power Above APRM Downscale or Un						Record Number	44

EA2. Ability to determine and/or interpret the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN:

EA2.03	SBLC tank level	4.3	4.4

Explanation of Answer

Justification:
Continue SLC pump operation and begin reactor cooldown. Correct. After 90 minutes operation with only 1 SLC pump running, less then 1100 gallons remain in th SLC Tank, but level is above the 325 gallon Low Level Pump trip setpoint. 4640 Gallons at the low level alarm point with 90 minutes runtime at 41.2 gpm [4640 - (41.2 x 90)] = 932 gallons remaining. Step RC/Q-19 directs continuation at step RC/P-20 for depressurization and cooldown.
Continue SLC pump operation and raise reactor water level. Incorrect. RPV Level cannot be raised until the reactor is shutdown under all conditions without boron.
Verify the 'A' SLC pump is tripped and continue rod insertion. Incorrect. 'A' pump will still be running.
Verify the 'A' SLC pump is tripped and exit EOP-101A. Incorrect. 'A' pump will still be running. EOP 101A is not exited until the reactor is shutdown under all conditions without boron.

Reference Title

HC.OP-EO.ZZ-0101A

Learning Objectives

EO101AE007	Explain the significance of "Cold Shutdown Boron Weight" and state its value.

Material Required for Examination	EOP Flowcharts; HCGS TS section 3.1.5		
Question Source:	New	Question Modification Method:	
Question Source Comments:			

Given the following conditions:

- A turbine trip and hydraulic ATWS occur from 65 percent power.
- EOP-101A and 102 are currently being executed.

Current plant conditions:

RPV Parameters

- Pressure 950 psig with TBVs controlling.
- Level -80" with RCIC and CRD injecting.
- Power IRM Range 8 @ 50 / 125 decreasing.
- SLC is injecting.

Containment Parameters

- Suppression Pool water temperature is 192 F steady
- Suppression Pool level 76.8 inches rising slowly

Which of the following actions will improve the required margin of safety?

a. Reduce Suppression Pool water temperature.

b. Lower Suppression Pool level.

c. Open additional Turbine Bypass Valves.

d. Rapidly Depressurize the RPV.

Answer	a	Exam Level	S	Cognitive Level	Application	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Emergency and Abnormal Plant Evolutions		RO Group	1	SRO Group	1			295037G406
295037	SCRAM Condition Present and Reactor Power Above APRM Downscale or Un						Record Number		45
2.4	Emergency Procedures and Plan								
2.4.6	Knowledge symptom based EOP mitigation strategies.								3.1 4.0

Explanation of Answer

Justification:
10CFR55.43 (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
Reduce Suppression Pool water temperature. Correct. SPT reduction allowed and required.
Lower Suppression Pool level. Incorrect. Level within allowable band.
Open additional Turbine Bypass Valves. Incorrect. Would cause cooldown with ATWS in progress. Not at SPT-P limit.
Rapidly Depressurize the RPV. Incorrect. Not permitted at this time.

Reference Title

EOP-101A and 102

Learning Objectives

EO101AE008	(R) Given any step of the procedure, explain the reason for performance of that step and/or evaluate the expected system response to control manipulations prescribed by that step.

Material Required for Examination	EOP Flowcharts

Question Source:	INPO Exam Bank	**Question Modification Method:**	Editorially Modified
Question Source Comments:	INPO BANK QID# 17079 Susquehanna 09/30/1999		

Given the following conditions:

- An Unusual Event is declared due to a radiological release.
- The Meterological Tower link to Hope Creek is malfunctioning.
- The link to Salem Generating Station is working properly.

Which one of the following sets of data must be requested from Salem Station to be communicated to the States of New Jersey and Delaware with the Initial Contact Message Form (ICMF)?

a. Wind Direction TO; Wind Speed 300 ft elevation.

b. Wind Direction TO; Wind Speed 33 ft elevation.

c. Wind Direction FROM; Wind Speed 300 ft elevation.

d. Wind Direction FROM; Wind Speed 33 ft elevation.

Answer	d	**Exam Level**	B	**Cognitive Level**	Memory	**Facility:**	Hope Creek	**Exam Date:**	06/17/2003
Tier:	Emergency and Abnormal Plant Evolutions	**RO Group**	2	**SRO Group**	1				295038A102
295038	High Off-Site Release Rate							**Record Number**	46

EA1. Ability to operate and/or monitor the following as they apply to HIGH OFF-SITE RELEASE RATE:

EA1.02 Meteorological instrumentation 3.0 3.8

Explanation of Answer

Wind Direction FROM; Wind Speed 33 ft elevation. Correct. IAW ICMF section IV
Wind Direction TO; Wind Speed 33 ft elevation. Incorrect. Wind direction is FROM
Wind Direction TO; Wind Speed 300 ft elevation. Incorrect. Wrong elevation. Wind direction is FROM
Wind Direction FROM; Wind Speed 300 ft elevation. Incorrect. Wrong elevation. Wind speed is 33 ft elev.

Reference Title

ECG Attachment 1

Learning Objectives

Material Required for Examination None

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following conditions:

- A severe accident has occurred.
- A radiological release is in progress.

Which one of the following choice correctly fills in the blanks of this statement?

The Emergency Plan prevents ______________ from receiving radiation doses of ______________ Rem Whole Body and ______________ Rem to the Thyroid.

a. members of the public; 25; 75

b. station personnel; 75; 200

c. station personnel; 25; 75

d. members of the public; 25; 200

Answer	d	Exam Level	B	Cognitive Level	Memory	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Emergency and Abnormal Plant Evolutions	RO Group	2	SRO Group	1				295038K301
295038	High Off-Site Release Rate							Record Number	47
EK3.	Knowledge of the reasons for the following responses as they apply to HIGH OFF-SITE RELEASE RATE:								
EK3.01	Implementation of site emergency plan							3.6	4.5

Explanation of Answer

Justification:
members of the public; 25; 200
members of the public; 25; 75
station personnel; 25; 75
station personnel; 75; 200

Reference Title

UFSAR Chapter 15

Learning Objectives

ACCANLE011	Given various accident scenarios, explain why Iodine, Xenon, and Krypton are elements of concern during these accident conditions IAW the Student Handout.

Material Required for Examination: None

Question Source: New

Question Modification Method:

Question Source Comments:

Which one of the following describes potential consequences of the failure to place the Containment Hydrogen Recombiners in service at the proper hydrogen concentration?

Increases threat to containment integrity caused by ____________________

a. high temperature.

b. high internal pressure.

c. high drywell to suppression chamber differential pressure.

d. low internal pressure.

Answer	b	Exam Level	B	Cognitive Level	Memory	Facility:	Hope Creek	Exam Date:	06/17/2003

Tier:	Emergency and Abnormal Plant Evolutions	RO Group	1	SRO Group	1	500000K101

500000	High Containment Hydrogen Concentration	Record Number	48

EK1. Knowledge of the operational implications of the following concepts as they apply to HIGH CONTAINMENT HYDROGEN CONCENTRATIONS:

EK1.01	Containment integrity	3.3	3.9

Explanation of Answer

high internal pressure. Correct. High H2 concentration s may cause a deflageration or detonation which will lead to containment failure by high internal pressure.
high temperature. Incorrect. Will damage internals but not design basis
high drywell to suppression chamber differential pressure. Incorrect. Function of the vent header and downcomer pipes.
low internal pressure. Incorrect. Internal pressure will increase.

Reference Title

LP NOH01H2RECM-00

EOP 102 PC/H-1

Learning Objectives

EO102PE002	Identify the reason(s) drywell temperature, pressure and hydrogen generation are controlled IAW the Primary Containment Control - Drywell Lesson Plan.
H2RECME001	From memory, explain the purpose of the Containment Hydrogen Recombiner System IAW the CHRS Lesson Plan.

Material Required for Examination None

Question Source: INPO Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments: INPO BANK QID # 21879 Perry 01/01/2001Modified for Hope Creek

Given the following conditions:

- The plant is operating at 25% power performing a startup.
- Control rod 18-23 has been determined to be stuck at position 00.
- While attempting to withdraw the control rod, indicated drive water flow is reading "0" gpm.

Which of the following is the cause of this indication?

a. Hydraulic Control Unit Directional Control Valve (122) has failed to reposition.

b. The 2 gpm Stabilizing Valve has failed to reposition.

c. Both Cooling Water Header to Exhaust Header Pressure Equalizing Valves have failed open.

d. The Drive Water Header Pressure Control Valve has failed closed.

Answer	a	Exam Level	B	Cognitive Level	Comprehension	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Plant Systems	RO Group	1	SRO Group	2				201001K303
201001	Control Rod Drive Hydraulic System							Record Number	49

K3. Knowledge of the effect that a loss or malfunction of the CONTROL ROD DRIVE HYDRAULIC SYSTEM will have on following:

K3.03 Control rod drive mechanisms | 3.1 | 3.2

Explanation of Answer

Justification:
Hydraulic Control Unit Directional Control Valve (122) has failed to reposition.- Correct- IAW M-47-1 SV-122 is the withdrawal solenoid and SV-120 is the exhaust solenoid
The 2 gpm Stabilizing Valve has failed to reposition.-Incorrect- this would effect total system flow not withdrawal flow
Both Cooling Water Header to Exhaust Header Pressure Equalizing Valves have failed open-Incorrect- this would allow the exhaust header pressure to rise above cooling water pressure but not effect flow to the drive
The Drive Water Header Pressure Control Valve has failed closed.-Incorrect- this would remove all flow to the system

Reference Title

M-47.

Learning Objectives

CRDHYDE002 (R) Given P&ID's M-46-1 and M-47-1, determine the flowpath of the Control Rod Drive Hydraulic System, including the following flowpaths:
a. Drive Water
b. Cooling Water
c. Exhaust Water
d. Charging Water
e. Scram
f. Seal Purge for Recirculation Pumps
g. RPV Level Reference Leg Backfill

Material Required for Examination Copy of P&ID M-46 and M-47

Question Source: Facility Exam Bank **Question Modification Method:** Direct From Source

Question Source Comments: Vision Bank QID# Q53363

A LOCA has occurred and reactor vessel water level is -140 inches.

Which of the following describes the steps necessary to restart a CRD pump?

a. Close the non-1E circuit breaker by depressing the CLOSE pushbutton and close the 1E circuit breaker by depressing the CRD pump START pushbutton.

b. Depress the LOCA OVERRIDE pushbutton, close the non-1E circuit breaker by depressing the CLOSE pushbutton, and close the 1E circuit breaker by depressing the CRD pump START pushbutton.

c. Depress the LOCA OVERRIDE pushbutton, close the 1E circuit breaker by depressing the CLOSE pushbutton, and close the non-1E circuit breaker by depressing the CRD pump START pushbutton.

d. Close the 1E circuit breaker by depressing the CLOSE pushbutton and close the non-1E circuit breaker by depressing the CRD pump START pushbutton.

Answer	c	Exam Level	B	Cognitive Level	Comprehension	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Plant Systems			RO Group	1	SRO Group	2		201001K605
201001	Control Rod Drive Hydraulic System							Record Number	50

K6. Knowledge of the effect that a loss or malfunction of the following will have on the CONTROL ROD DRIVE HYDRAULIC System:

K6.05	A.C. power	3.3	3.3

Explanation of Answer

Justification:
Depress the LOCA override push-button, close the 1E circuit breaker by depressing the close push-button, and close the non-1E circuit breaker by depressing the CRD pump START push-button. -Correct- CRD pumps are load shed on a LOCA signal and requires both the 1E and Non 1E breakers to be closed
Close the non-1E circuit breaker by depressing the close push-button and close the 1E circuit breaker by depressing the CRD pump START push-button.-Incorrect- has the power supplies titles backwards, also requires LOCA override
Close the 1E circuit breaker by depressing the close push-button and close the non-1E circuit breaker by depressing the CRD pump START push-button.-Incorrect- LOCA signal is present and will require LOCA override
Depress the LOCA override push-button, close the non-1E circuit breaker by depressing the close push-button, and close the 1E circuit breaker by depressing the CRD pump START push-button.-Incorrect- has titles of breakers backwards

Reference Title

HC.OP-AB.ZZ-0001

HC.OP-SO.BF-0001

Learning Objectives

CRDHYDE015 (R) Given a copy of HC.OP-SO.BF-0001 and P&ID M-46-1, explain the actions necessary to place in service, or shift, the following components, including specific plant locations, IAW HC.OP-SO.BF-0001:

a. CRD Pumps
b. CRD Pump Suction Filter
c. CRD Drive Water Filter
d. CRD Flow Control Station
e. Stabilizing Valves

0AB135E003 (R) Discuss the operational implications of the abnormal indications/alarms for system operating parameters related to Station Blackout/Loss Of Offsite Power Diesel Generator Malfunction, Abnormal Operating Procedure.

Material Required for Examination None

Question Source:	Facility Exam Bank	Question Modification Method:	Direct From Source
Question Source Comments:	Vision Bank Qid# Q54297		

Given the following conditions:

- Reactor power is 85 percent during a plant start-up.
- A control rod is selected for withdrawal.
- An adjacent "C" level LPRM providing signals to an Average Power Range Monitor (APRM) Channel and a Rod Block Monitor (RBM) Channel fails downscale once the rod is in motion.

Which one of the following describes the effect of the failure on the Reactor Manual Control System (RMCS) and the reason why?

RMCS will initiate control rod blocks:

a. at a lower actual local power level because the LPRM will be automatically bypassed and removed from the RBM only. The APRM and the RBM readings will be lower than actual.

b. at a lower actual local power level because the LPRM will be automatically bypassed and removed from both the APRM and RBM. The APRM and RBM readings will NOT be affected.

c. at a higher actual local power level because the LPRM will be automatically bypassed and removed from the APRM only. The APRM reading will NOT be affected and the RBM reading will be lower than actual.

d. at a higher actual local power level because the LPRM will NOT be automatically bypassed to the APRM or the RBM. The APRM and RBM readings will be lower than actual.

Answer	d	Exam Level	B	Cognitive Level	Comprehension	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Plant Systems			RO Group	1	SRO Group	2		201002A105
201002	Reactor Manual Control System							Record Number	51

A1. Ability to predict and/or monitor changes in parameters associated with operating the REACTOR MANUAL CONTROL SYSTEM controls including:

A1.05 Local reactor power 3.4 3.6

Explanation of Answer: The average including the failed LPRM will be lower than actual. This would required a higher local power to cause a rod block. Once the LPRM card is bypassed, then the LPRMs are averaged and will read normally.

Reference Title

HC.OP-SO.SF-0002

Learning Objectives

MANCONE008	(R) From memory, explain the interrelationships between the Reactor Manual Control System and the following: Rod Worth Minimizer Neutron Monitoring System Rod Block Monitor System Mode Switch Refueling System Refueling Bridge Refueling Grapple/Hoists f. 120 VAC Uninterruptible Power Supply
RBMSYSE002	(R) Given a diagram of, or access to, the controls, indications and alarms in the Control Room, determine the status of the Rod Block Monitor (RBM) System by observation of the controls/indications/alarms, IAW control room procedures.

Material Required for Examination: None

Question Source:	INPO Exam Bank	Question Modification Method:	Significantly Modified
Question Source Comments:	INPO Bank QID #12556 Limerick 11/10/1995		

Given the following:

- Reactor power is 83%.
- Neither RBM is bypassed with the joystick.
- Rod 30-31 has just been selected.

Use the attached figure of the 4-Rod Display for LPRM indications
(Ribbon readings are approximates)

Assuming all other LPRMs are operable, which of the following describes the operability status of the RBM CHANNEL A and CHANNEL B?

a. A- Operable; B- Operable * Correct answer changed to #"a".

b. A- Operable; B- Inoperable

c. A- Inoperable; B- Operable

d. A- Inoperable; B- Inoperable

Answer	b	Exam Level	S	Cognitive Level	Application	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Plant Systems		RO Group	2	SRO Group	2			201006G132
201006	Rod Worth Minimizer System (RWM) (Plant Specific)							Record Number	52
2.1	Conduct of Operations								
2.1.32	Ability to explain and apply system limits and precautions.							3.4	3.8

Explanation of Answer

Justification:
SRO 55.43 (2) Facility operating limitations in the Technical specifications and their bases.
Technical Specification interpretation IAW SH.OP-AP.ZZ-0108 Exhibit 3 which requires at least 50% LPRM inputs for each level operable.
A- Operable; B- Inoperable Correct- Only 1 of 4 LPRM for D LPRM Level makes RBM B Administratively inoperable.
A- Operable; B- Operable Incorrect- RBM B is inoperable.
A- Inoperable; B- Operable Incorrect- RBM A is operable; B is inoperable.
A- Inoperable; B- Inoperable Incorrect- RBM A is operable.

Reference Title

SH.OP-AP.ZZ-0108 Exhibit 3

Learning Objectives

RBMSYSE005	(R) Given a scenario of applicable operating conditions and access to the Technical Specifications: Select those sections which are applicable to the Rod Block Monitor (RBM) System. Evaluate Rod Block Monitor (RBM) System operability and determine required actions based upon system inoperability. Explain the bases for those Technical Specification items associated with the Rod Block Monitor (RBM) System. (SRO Only) IAW Technical Specifications.

Material Required for Examination: Need LPRM figure.

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following conditions:

- The plant is operating on the 100% Rod Line at 100% power.
- 'B' Recirculation Pump trip occurs due to inadvertant bump of the Drive Motor Breaker local trip switch.
- Reactor settles at 56% power.
- Recirc Loop A Flow (FI-R611A) is 38 Mlbm/hr.
- Recirc Loop B Flow (FI-R611B) is 3 Mlbm/hr.
- Core Flow (FR-R613) is 35 Mlbm/hr.
- There are NO indications of thermal hydraulic instability.
- HC.OP-AB.RPV-0003, Recirculation System is entered.
- HC.OP-AB.RPV-0002, Reactor Power Oscillations is entered.

Which of the following action(s) is(are) required per HC.OP-AB.RPV-0002, Reactor Power Oscillations?

a. Lock the Mode Switch in the Shutdown position.

b. Raise 'A' Recirculation Pump speed until total core flow is above 45%.

c. Reduce 'A' Recirc flow and insert control rods IAW Stuff Sheet.

d. Insert control rods IAW Stuff Sheet.

Answer a **Exam Level** B **Cognitive Level** Application **Facility:** Hope Creek **Exam Date:** 06/17/2003

Tier: Plant Systems **RO Group** 2 **SRO Group** 2 202001A203

202001 Recirculation System **Record Number** 53

A2. 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 those abnormal conditions or operations:

A2.03 Single recirculation pump trip 3.6 3.7

Explanation of Answer

Justification:
Raise 'A' Recirculation Pump speed until total core flow is above 45%. Inorrect. Would be response if in the Exit region. Response to operating in the exit region is either inserting rods OR raising core flow with the running recirc pump.
Lock the Mode Switch in the Shutdown position. Correct. In the Scram Region of new power to flow map.
Reduce 'A' Recirc flow and insert control rods IAW Stuff Sheet.. Incorrect. Insert rods by scram.
Insert control rods IAW Stuff Sheet. Would be correct if in the exit region.

Reference Title

HC.OP-AB.RPV-0002

Learning Objectives

ABRPV2E007	(R) Explain the bases for Subsequent Actions and the information contained in the Discussion Bases Section of Reactor Power Oscillations.
ABRPV2E001	Recognize abnormal indications/alarms and/or procedural requirements for implementing Reactor Power Oscillations.
ABRPV2E007	(R) Explain the bases for Subsequent Actions and the information contained in the Discussion Bases Section of Reactor Power Oscillations.

Material Required for Examination New Power to Flow Map

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following conditions:

- The plant is operating at 100 percent power with all systems normal.
- LPCI Channel B receives a LOCA Level 1 initiation signal.
- BVH-210 RHR Room Cooler is in Auto Lead.
- FVH-210 RHR Room Cooler is in Auto.
- CRIDS Point D3122 RHR Room Cooler Low Flow alarm occurs 5 minutes later.
- B RHR Room temperature is 110 degF.

Which one of the following describes the status of the RHR Room Coolers in B RHR Room?

a. B running; F running.

b. B running; F NOT running.

c. B NOT running; F running.

d. B NOT running; F NOT running.

Answer	c	Exam Level	B	Cognitive Level	Application	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Plant Systems	RO Group	1	SRO Group	1				203000K110
203000	RHR/LPCI: Injection Mode (Plant Specific)							Record Number	54

K1.	Knowledge of the physical connections and/or cause- effect relationships between RHR/LPCI: INJECTION MODE and the following:		
K1.10	ECCS room coolers	3.2	3.2

Explanation of Answer

B NOT running; F running. Correct- B cooler will start first on temperature. A low flow trip of the B cooler will cause the computer trouble alarm. F cooler will start because the cooler is in Auto.
B running; F running. Incorrect. B cooler would be tripped.
B running; F NOT running. Incorrect. B cooler would be tripped. F cooler would be running.
B NOT running; F NOT running. Incorrect. F cooler would be running.

Reference Title

HC.OP-SO.GR-0001 3.3.5

H-83 Sheets 5 and 11

Learning Objectives

RHRSYSE008	(R) Given a system which physically connects to or is required to support the operation of the RHR System or components therein, explain the function of the supporting system, IAW the RHR System Lesson Plan.
RHRSYSE012	Given a set of conditions and a drawing of the controls, instrumentation and/or alarms located in the main control room, assess the status of the Residual Heat Removal System or its components by evaluation of the controls/instrumentation/alarms IAW the RHR System Lesson Plan.

Material Required for Examination	H-83 sheets 5 and 11		
Question Source:	New	Question Modification Method:	
Question Source Comments:			

Given the following current conditions:

- The plant is operating at 95% power.
- The RWCU system has just been returned to service.
- The "A" RWCU pump is running.
- The "B" RWCU pump is C/T.
- RWCU return to Feedwater temp CRIDS pt A215 is reading 410F.

Based on the above conditions, what flow is the maximum allowable flow for long term operation on RWCU return to Feedwater flow CRIDS pt A2856?

a. 173 gpm

b. 175 gpm

c. 345 gpm

d. 350 gpm

Answer	a	Exam Level	B	Cognitive Level	Application	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Plant Systems	RO Group	2	SRO Group	2				204000K103
204000	Reactor Water Cleanup System							Record Number	55

K1. Knowledge of the physical connections and/or cause- effect relationships between REACTOR WATER CLEANUP SYSTEM and the following:

K1.03 Reactor feedwater system 3.1 3.1

Explanation of Answer: Requires HC.OP-SO.BG-0001 Attachment 1 & 2 for ILOT use
correct answer. One pump operation at 410 degf on attachment 2 just below line for long term operation

Reference Title

HC.OP-SO.BG-0001 Attachment 1 & 2 for ILOT use

Learning Objectives

RWCU00E013	Given any system that interrelates with the RWCU System, explain the purpose of that interface IAW the RWCU System Lesson Plan.
RWCU00E003	(R) Given the necessary sheets of P&ID's M-44-1 and M-45- 1: a. Determine the normal RWCU System flowpath IAW the RWCU System Lesson Plan. b. Determine the blowdown RWCU System flowpath(s) IAW the RWCU System Lesson Plan. c. Determine the recirculation RWCU System flowpath IAW the RWCU System Lesson Plan.

Material Required for Examination: HC.OP-SO.BG-0001 Attachment 1 & 2 for ILOT use

Question Source: Facility Exam Bank **Question Modification Method:** Direct From Source

Question Source Comments: Vision Bank QID# Q56217

Given the following conditions:

- The plant is operating at 100% reactor power.
- HPCI Pump InService test is in progress at rated flow.
- HPCI discharge pressure is 1150 psig.
- While attempting to adjust pump flow, the flow controller setpoint remains stationary at 4000 gpm in AUTO.
- The PO reports the HPCI flow controller works in MANUAL and develops rated flow.

What effect does this have on HPCI Operability at the PRESENT time?

a. HPCI is operable because it can develop rated flow.

b. HPCI is "operable but non-conforming" because it is NOT capable of meeting all surveillance requirements.

c. HPCI is "operable but degraded" because it has lost testing capability.

d. HPCI is inoperable because it is NOT capable of meeting all surveillance requirements.

Answer	d	Exam Level	S	Cognitive Level	Application	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Plant Systems		RO Group	1	SRO Group	1			206000A402
206000	High Pressure Coolant Injection System							Record Number	56
A4.	Ability to manually operate and/or monitor in the control room:								
A4.02	Flow controller: BWR-2, 3, 4							4.0	3.8

Explanation of Answer

Justification:
SRO 55.43 (2) Facility operating limitations in the Technical specifications and their bases.
HPCI is inoperable because it is NOT capable of meeting all surveillance requirements. Correct. HPCI must be in AUTO with a setpoint of 5600 gpm and capable of rated flow and discharge pressure.
HPCI is operable because it can develop rated flow. Incorrect. HPCI must be in AUTO with a setpoint of 5600 gpm.
HPCI is "operable but degraded" because it has lost testing capability. An Operable but degraded case could be made if the setpoint was stuck at 5600 gpm.
HPCI is "operable but non-conforming" because it is NOT capable of meeting all surveillance requirements. Operable but non conforming is not applicable.

Reference Title

HC.OP-SO.BJ-0001

SH.OP-AP.ZZ-0108

Learning Objectives

HPCI00E018	(R) Given plant conditions and access to Technical Specifications: Select those sections which are applicable to the HPCI System IAW HCGS technical specifications. Evaluate HPCI System operability and required actions based upon system operability IAW HCGS technical specifications. (SRO Only) Explain the bases for those technical specification items associated with the HPCI System IAW HCGS technical specifications. (SRO Only)

Material Required for Examination	Tech Spec 3.5.1
Question Source:	Facility Exam Bank
Question Modification Method:	Significantly Modified
Question Source Comments:	VISION QID# Q55949 significantly modified

From the list below, select the choice which is LOWEST in priority for use as reactor pressure control as described in HC.OP-IO.ZZ-0007 OPERATIONS FROM HOT STANDBY section Maintaining Hot Standby (MSIV's Open).

I. Main Steam Line Drains.
II. RCIC or HPCI Steam Line Drains.
III. RCIC or HPCI in Full Flow test.
IV. RFPT's min flow operation.

a. I.

b. II.

c. III.

d. IV.

Answer	c	**Exam Level**	R	**Cognitive Level**	Memory	**Facility:**	Hope Creek	**Exam Date:**	06/17/2003
Tier:	Plant Systems	**RO Group**	1	**SRO Group**	1				206000G123
206000	High Pressure Coolant Injection System							**Record Number**	57

2.1	Conduct of Operations		
2.1.23	Ability to perform specific system and integrated plant procedures during different modes of plant operation.	3.9	4.0

Explanation of Answer

JUSTIFICATION:
IAW HC.OP-IO.ZZ-0007 Note 5.2.5 directs the descending order of priority as follows.
Main Steam Line Drains; RFPT's min flow operation IAW system operating procedure, RCIC or HPCI Steam Line Drains, RCIC or HPCI in Full Flow test.
III - Correct. RCIC or HPCI in Full Flow test is the last of the listed systems.
I - Incorrect.
II - Incorrect.
IV - Incorrect.

Reference Title

HC.OP-IO.ZZ-0007 Note 5.2.5

Learning Objectives

IOP007E003	(R) Explain the basis for all Precautions, Limitations and Notes listed in the OPERATIONS FROM HOT STANDBY Integrated Operating Procedure, IAW this Lesson Plan

Material Required for Examination None

Question Source: Facility Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments: VISION QID #Q57126 Significantly modified.

Given the following conditions:

- A LOCA occurred.
- RPV level has been stabilized above TAF.
- Drywell sprays are in service.
- Core Spray pump 'A' amp indicator begins to fluctuate.

Which one of the following would cause the fluctuation and what action is permitted that would remedy the condition IAW HC.OP-AB.ZZ-0155 Degraded ECCS Performance?

a. Clogging of the suction strainer; reduce loop flow.

b. Partial closure of the injection valve; manually open the injection valve fully.

c. Clogging of the suction strainer; throttle 'A' Core Spray Pump manual discharge valve.

d. Partial closure of the injection valve; stop the 'C' Core Spray pump.

Answer	a	Exam Level	B	Cognitive Level	Application	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Plant Systems			RO Group	1	SRO Group	1		209001A206
209001	Low Pressure Core Spray System							Record Number	58

A2. Ability to (a) predict the impacts of the following on the LOW PRESSURE CORE SPRAY SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:

A2.06 Inadequate system flow 3.2 3.2

Explanation of Answer

Justification:
Clogging of the suction strainer; reduce loop flow. Correct. Fluctuating amps is a symptom of clogged suction strainers. Throttling loop flow directed by Attachment 2
Partial closure of the injection valve; manually open the injection valve fully. Incorrect. Partial closure is the remedy, not the cause.
Clogging of the suction strainer; throttle 'A' Core Spray manual discharge valve. Incorrect. Would reduce pump flow but not in accordance with AB.ZZ-0155.
Partial closure of the injection valve; stop the 'C' Core Spray pump. Incorrect. Stopping C pump would increase flow through A pump and make the conditions worse.

Reference Title

HC.OP-AB.ZZ-0155

Learning Objectives

0AB155E005	(R) Interpret and apply charts, graphs and tables contained within the Degraded ECCS Performance/Loss Of NPSH, Abnormal Operating Procedure.
0AB155E001	Recognize abnormal indications/alarms and/or procedural requirements for implementing, Degraded ECCS Performance/Loss Of NPSH, Abnormal Operating Procedure.
0AB155E003	(R) Discuss the operational implications of the abnormal indications/alarms for system operating parameters related to Degraded ECCS Performance/Loss Of NPSH, Abnormal Operating Procedure.

Material Required for Examination None

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following conditions:

- Drywell pressure increased to 2 psig,
- Off-site power is lost.

Which of the following describes the start sequence for the core spray systems after off-site power was lost?

a. Core Spray pumps "A" and "B" start immediately after the diesel generator output breakers are closed. Core Spray pumps "C" and "D" start six seconds after the diesel generator output breakers are closed.

b. Core Spray pumps "A" and "C" start immediately after the diesel generator output breaker is closed. Core Spray pumps "B" and "D" start six seconds after the diesel generator output breakers are closed.

c. Core Spray pumps "A", "B", "C", and "D" start immediately after the diesel generator output breakers are closed.

d. Core Spray pumps "A", "B", "C", and "D" start six seconds after the diesel generator output breaker is closed.

Answer	d	Exam Level	R	Cognitive Level	Comprehension	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Plant Systems	RO Group	1	SRO Group	1				209001K201
209001	Low Pressure Core Spray System							Record Number	59
K2.	Knowledge of electrical power supplies to the following:								
K2.01	Pump power							3.0	3.1

Explanation of Answer

Justification:
Core Spray pumps "A", "B", "C", and "D" start six seconds after the diesel generator output breaker is closed. Correct. With a LOP, all pumps start 6 seconds after the edg output breaker closed.
Core Spray pumps "A" and "C" start immediately after the diesel generator output breaker is closed. Core Spray pumps "B" and "D" start six seconds after the diesel generator output breaker is closed.
Core Spray pumps "A", "B", "C", and "D" start immediately after the diesel generator output breaker closes.
Core Spray pumps "A" and "B" start immediately after the diesel generator output breaker is closed. Core Spray pumps "C" and "D" start six seconds after the diesel generator output breaker is closed.

Reference Title

HC.OP-SO.BE-0001

Learning Objectives

CSSYS0E007	(R) Given plant problems/industry events associated with the Core Spray System: a. Summarize/Identify the root cause of the plant problem/industry event, IAW the Core Spray System Lesson Plan. b. Summarize/Identify the HCGS design and/or procedural guidelines that mitigate/reduce the likelihood of the plant problem/industry event at HCGS, IAW the Core Spray System Lesson Plan. c. Summarize/Identify the "lessons learned" from the plant problem/industry event, IAW the Core Spray System Lesson Plan.
CSSYS0E005	(R) For a given set of plant conditions, from memory, summarize/identify the interrelationship between the Core Spray System and any of the following, IAW the Core Spray System Lesson Plan: a. Residual Heat Removal (RHR) System b. Torus Compartment c. 4160 VAC Class 1E Distribution System d. 480 VAC Class 1E Distribution System e. 125 VDC Class 1E Distribution System

f. Nuclear Boiler
g. Liquid Radwaste System
h. Condensate Storage and Transfer System
i. Primary Containment Instrument Gas (PCIG) System
j. High Pressure Coolant Injection (HPCI) System
k. Condensate Storage Tank
l. Automatic Depressurization System (ADS)
m. Emergency Diesel Generators (EDGs)
n. Nuclear Boiler Instrumentation System
o. Standby Liquid Control (SLC) System

Material Required for Examination	None

Question Source:	Facility Exam Bank	Question Modification Method:	Significantly Modified
Question Source Comments:	VISION Bank QID# Q53278		

Given the following conditions:

- A Loss of Offsite Power (LOP) concurrent with an ATWS has occurred.
- The "A", "B", & "C" Emergency Diesel Generators are supplying their 4kv buses.
- Emergency Diesel Generator "D" will NOT start.
- The CRS has ordered that the Standby Liquid Control System be initiated.

Which one of the following describes the components of the Standby Liquid Control System that are available for injection?

a. SLC pump A and squib valve F004A ONLY.

b. NEITHER SLC pump NOR associated squib valve.

c. BOTH SLC pumps and both squib valves.

d. SLC pump B and squib valve F004B ONLY.

Answer	c	Exam Level	B	Cognitive Level	Memory	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Plant Systems	RO Group	1	SRO Group	1				211000K202
211000	Standby Liquid Control System							Record Number	60

K2. Knowledge of electrical power supplies to the following:

K2.02 Explosive valves 3.1 3.2

Explanation of Answer

Justification:
BOTH SLC pumps and both squib valves. Correct. The "A" pump and squib valve are powered from "A" EDG. The "B" pump and squib valve are powered from the "B" EDG. The "D" DG supplies power to the SLC isolation valve F006B which is normally open and remains open on a loss of power.
NEITHER SLC pump nor associated squib valve. Incorrect.
SLC pump A and squib valve F004A ONLY. Incorrect.
SLC pump B and squib valve F004B ONLY. Incorrect.

Reference Title

HC.OP-SO.BH-0001

Learning Objectives

SLCSYSE007 From memory identify the power supply (i.e., 1E or Non-1E) to each of the following I.A.W. the Lesson Plan.
a. Standby Liquid Control Pumps.
b. Standby Liquid Control System Squib valves.
c. Standby Liquid Control System Storage Tank Heaters.

SLCSYSE015 (R) From memory, summarize/identify the impact that a loss or malfunction of each of the following would have on the Standby Liquid Control System I.A.W. the Lesson Plan.
a. Standby Liquid Control Squib Valve
b. Standby Liquid Control Storage Tank Level
c. Redundant Reactivity Control System
d. AC Power

Material Required for Examination	None		
Question Source:	Facility Exam Bank	Question Modification Method:	Editorially Modified
Question Source Comments:	Vision Bank QID # Q54183		

Manipulating which one of the following components ensures the Squib valves do NOT fire and the RWCU system remains in operation during testing of the Standby Liquid Control pumps?

a. Closing SLC Injection valves F006A and B.

b. Opening the breakers for F001 and F004.

c. Bezel keyswitches on 10C651C console.

d. Local panel pump start switches.

Answer	d	Exam Level	R	Cognitive Level	Memory	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Plant Systems		RO Group	1	SRO Group	1			211000K402
211000	Standby Liquid Control System							Record Number	61

K4. Knowledge of STANDBY LIQUID CONTROL SYSTEM design feature(s) and/or interlocks which provide for the following:

K4.02 Component and system testing 3.0 3.2

Explanation of Answer

Justification:
Using local panel pump start switches. Correct. Local start switches prevent firing of the squib valves and automatic closure of the RWCU F001 and F004.
Opening the breakers for F001 and F004. Not driven by any procedure. Does not stop firing of the squib valves.
Bezel keyswitches on 10C651C console. Keyswitches function as a permissive for manual SLC initiation.
Closing SLC Injection valves F006A and B. Prevents injection to the RPV. Will not prevent RWCU isolation or Squib valve firing.

Reference Title

HC.OP-SO.BH-0001

Learning Objectives

SLCSYSE009 (R) Given plant conditions, summarize/identify the Standby Liquid Control System local response to an automatic/manual system initiation I.A.W. the Lesson Plan.

Material Required for Examination None

Question Source: New **Question Modification Method:**

Question Source Comments:

During a failure-to-scram condition, which of the following indications would confirm that HC.OP-AB.ZZ-0000(Q), "Reactor Scram", should be exited and HC.OP-EO.ZZ-0101A(Q) entered?

a. All Blue lights are lit on the Full Core Display.

b. All HCU Accumulator Trouble lights are lit on the Full Core Display.

c. Computer Pt in Alarm Overhead Annuciator C6-F5 extinguished.

d. All APRM "downscale" lights are illuminated.

Answer c **Exam Level** R **Cognitive Level** Memory **Facility:** Hope Creek **Exam Date:** 06/17/2003

Tier: Plant Systems **RO Group** 1 **SRO Group** 1 212000A307

212000 Reactor Protection System **Record Number** 62

A3. Ability to monitor automatic operations of the REACTOR PROTECTION SYSTEM including:

A3.07 SCRAM air header pressure 3.6 3.6

Explanation of Answer

Computer Pt in Alarm Overhead Annuciator C6-F5 extinguished. Correct. Indication that CRD Pilot Air Pressure is NOT depressurized as expected following scram.
All Blue lights are lit on the Full Core Display. Indicates all scram valves open. This is the response for a normal scram.
All HCU Accumulator Trouble lights are lit on the Full Core Display. This is the response for a normal scram.
All APRM "downscale" lights are illuminated. This is the response for a normal scram.

Reference Title

HC.OP-AB.ZZ-0001 Attachment 1

HC.OP-AR.ZZ-0011 Attachment F5

Learning Objectives

AB0000E003 State four (4) methods by which the operator can verify a successful scram action.

Material Required for Examination Overhead annunciator window C6 figure from HC.OP-AR.ZZ-0011

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following conditions:

- The reactor has scrammed and the mode switch is in SHUTDOWN.
- The problem that caused the scram has been identified and corrected.
- Annunciator "CRD SCRAM DISCH VOL WTR LVL HI" is sealed in.

Which one of the following describes the reactor protection system (RPS) response when you place the Scram Discharge Volume High Level Keylock switch in BYPASS, followed by taking the scram reset switches to RESET, and one minute later placing the mode switch in STARTUP for NI testing?

a. The RPS will reset and remain reset.

b. The RPS will reset and again scram.

c. Nothing will occur due to the present plant conditions.

d. The RPS will reset when the scram discharge volume drains.

Answer	b	Exam Level	B	Cognitive Level	Comprehension	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Plant Systems	RO Group	1	SRO Group	1				212000A404
212000	Reactor Protection System							Record Number	63

A4. Ability to manually operate and/or monitor in the control room:

A4.04 Bypass SCRAM instrument volume high level SCRAM signal 3.9 3.9

Explanation of Answer

Justification:
The RPS will reset and again scram. Correct. Placing the mode switch to Startup unbypasses the Bypass interlock and a scram occurs.
The RPS will reset and remain reset. Incorrect. Placing the mode switch to Startup unbypasses the Bypass interlock and a scram occurs.
Nothing will occur due to the present plant conditions. Incorrect. A scram occurs.
The RPS will reset when the scram discharge volume drains. Incorrect. The SDV will not drain due to the scram signal pressent

Reference Title

HC.OP-SO.SB-0001

Learning Objectives

RPS000E016	From memory, explain how to reset a scram IAW the RPS System Operating Procedure.
RPS000E017	Given a labeled diagram/drawing of, or access to, the Reactor Protection System controls, and/or alarms located in the Control Room: Explain the function of each indicator. Assess plant conditions that will cause the indications to light or extinguish. Determine the effect of each control switch on the Reactor Protection System.

Material Required for Examination None

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following conditions:

- The unit is at 100% power when the BD483 inverter output momentarily spikes to 137 volts and immediately returns to a normal regulated output of 119.5 VAC.
- The cause of this spike is unknown at this time.

How will this transient affect the unit?

a. ½ scram and rod block from 'D' and 'F' APRM's; NO other effects

b. ½ scram and rod block from 'B' and 'D' APRMs and a rod block from 'A' and 'B' RBM; NO other effects

c. ½ scram from 'B' APRM and a rod block from 'B' RBM; NO other effects

d. ½ scram and rod block from 'B', 'D' and 'F' APRM's and a rod block from 'B' RBM; NO other effects

Answer	d	Exam Level	R	Cognitive Level	Comprehension	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Plant Systems	RO Group	2	SRO Group	2				215002K203
215002	Rod Block Monitor System							Record Number	64

K2.	Knowledge of electrical power supplies to the following:		
K2.03	APRM channels: BWR-3, 4, 5	2.8	2.9

Explanation of Answer

JUSTIFICATION:
Correct answer: " ½ scram and rod block from 'B', 'D' and 'F' APRM's and a rod block from 'B' RBM; no other effects". High input voltage trips the EPA breaker on overvoltage above 132 volts.
The following distractors are incorrect as follows:
·" ½ scram and rod block from 'B' and 'D' APRMs and a rod block from 'A' and 'B' RBM; no other effects"
Incorrect - BD483 UPS powers B, D, and F APRMs, and 'B' RBM, no mention of "F" APRM and mention of "A" RBM
·"½ scram from 'B' APRM and a rod block from 'B' RBM; no other effects"
Incorrect- BD483 UPS powers B, D, and F APRMs, no mention of "D" or "F"
"½ scram and rod block from 'D' and 'F' APRM's; no other effects"
- Incorrect- BD483 UPS powers B, D, and F APRMs, and 'B' RBM, no mention of "B" APRM or "B" RBM

Reference Title

HCGS Tech Spec 3.8.4.6

HC.OP-SO.SE-0001

Learning Objectives

APRM00E004	Given a system which connects to or is required for the support of the APRMS, explain the function of the system interrelationship, IAW the Student Handout.

Material Required for Examination	None		
Question Source:	Facility Exam Bank	Question Modification Method:	Editorially Modified
Question Source Comments:	Vision Bank QID# Q56309		

Given the following conditions:

- IRM C reads 5 on Range 6.
- The range switch was placed to Range 5 and then Range 4.

Which of the following describes the resulting IRM C indication?

a. 5 on range 5, and off-scale on range 4

b. Off-scale on range 5, and off-scale on range 4.

c. 5 on range 5, and 50 on range 4.

d. 50 on range 5, and 50 on range 4.

Answer	c	Exam Level	R	Cognitive Level	Comprehension	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Plant Systems	RO Group	1	SRO Group	2				215003A403
215003	Intermediate Range Monitor (IRM) System							Record Number	65
A4.	Ability to manually operate and/or monitor in the control room:								
A4.03	IRM range switches							3.6	3.4

Explanation of Answer

Requires attached figures.
The IRM range switches are arranged with the odd scale ranges are from 0-40, and the even scale range switches are 0-125. Going from the odd ranges to the even ranges expands the scale from 0-40 to 0-125. Then going from the even to the odd ranges will increase the magnitude of the scale by a factor of 10.

Reference Title

NOH01IRMSYS-00

Learning Objectives

IRMSYSE009	(R) Given a set of conditions and a drawing of the controls, instrumentation and/or alarms, or access to, the control room, evaluate the status of the IRM controls/instrumentation/alarms IAW control room procedures.
IRMSYSE004	(R) Given a labeled diagram of, or access to, the IRM controls/indication bezel: Explain the function of each indicator Assess the plant conditions that will cause the indicator to light or extinguish Predict the effect of each control on the IRM System Select the condition or permissives required for the control switches to perform their intended function. IAW control room procedures

Material Required for Examination	None		
Question Source:	Facility Exam Bank	Question Modification Method:	Editorially Modified
Question Source Comments:			

Given the following conditions:

- Reactor startup in progress.
- IRMs are on range 10 and reading 25.
- Reactor Mode switch is in STARTUP/HOT STANDBY.
- IRM C fails downscale.

Which of the following lists rod block status with the present condition AND if the Reactor Mode switch is placed in RUN?

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

b. NO Rod Block exists. Placing Reactor Mode switch in RUN will NOT result in a Rod Block.

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

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

Answer	a	**Exam Level**	R	**Cognitive Level**	Comprehension	**Facility:**	Hope Creek	**Exam Date:**	06/17/2003
Tier:	Plant Systems			**RO Group**	1	**SRO Group**	2		215003K102
215003	Intermediate Range Monitor (IRM) System							**Record Number**	66

K1. Knowledge of the physical connections and/or cause- effect relationships between INTERMEDIATE RANGE MONITOR (IRM) SYSTEM and the following:

K1.02 Reactor manual control 3.6 3.6

Explanation of Answer

Justification:
A Rod Block exists. The Rod Block will clear after placing the Reactor Mode switch in RUN. Correct. Downscale Rod Block clears after Reactor mode Switch is placed in Run.
NO Rod Block exists. Placing Reactor Mode switch in RUN will NOT result in a Rod Block. Incorrect. A rod block will exist based on the IRM downscale.
NO Rod Block exists. Placing the Reactor Mode switch in RUN will result in a Rod Block. Incorrect. A rod block will exist based on the IRM downscale.
A Rod Block exists. The Rod Block will NOT clear after placing the Reactor Mode switch in RUN. The rod block will clear after placing Mode switch to run.

Reference Title

HC.OP-SO.SE-0001

Learning Objectives

IRMSYSE009	(R) Given a set of conditions and a drawing of the controls, instrumentation and/or alarms, or access to, the control room, evaluate the status of the IRM controls/instrumentation/alarms IAW control room procedures.

Material Required for Examination None

Question Source: INPO Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: INPO BANK QID# 6330 09/26/1998 Dresden

Given the following conditions:

- The Mode Switch is in the STARTUP/HOT STANDBY position.
- APRM "D" is indicating 1%.
- Reactor power is approximately midscale on Range 7 of the IRMs.
- Recirc Pump speeds are at minimum.
- APRM "C" begins to fail upscale.

Predict the correct RPS response.

a. Half scram when "C" APRM reads 51%.

b. Half scram when "C" APRM reads 15%.

c. Full scram when "C" APRM reads 15%.

d. Full scram when "C" APRM reads 51%.

Answer	b	Exam Level	B	Cognitive Level	Comprehension	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Plant Systems		RO Group	1	SRO Group	1			215005A102
215005	Average Power Range Monitor/Local Power Range Monitor System							Record Number	67

A1.	Ability to predict and/or monitor changes in parameters associated with operating the APRM/LPRM controls including:	
A1.02	RPS status	3.9 4.0

Explanation of Answer

Justification:
Half scram when "C" APRM reads 15%. Correct. Half scram on RPS A when C APRM reaches 15 percent with the mode switch in Startup.
Half scram when "C" APRM reads 51% Incorrect. 15 percent with the mode switch in Startup.
Full scram when "C" APRM reads 15% Incorrect. Half scram only.
Full scram when "C" APRM reads 51% Incorrect. Half scram only. 15 percent with the mode switch in Startup.

Reference Title

HC.OP-SO.SE-0001

Learning Objectives

APRM00E009	(R) From memory, IAW Technical Specifications, determine the rod blocks and/or scrams initiated by the APRM System, IAW the Student Handout.

Material Required for Examination	None
Question Source:	INPO Exam Bank
Question Modification Method:	Editorially Modified
Question Source Comments:	INPO Exam Bank QID# 17669 05/10/1999 Vermont Yankee 1

Given the following conditions:

- A plant startup is in progress with power at 20%.
- Recirculation flow is 30%.
- The "A" APRM Flow Unit output remains at 30% as recirculation flow is raised.

As the plant startup continues, what will be the FIRST protective action to occur and the reason for that action?

a. A full scram will occur due to flow biased neutron flux upscale.

b. A control rod block will occur due to flow biased neutron flux upscale.

c. A half scram will occur due to a flow unit "inop" signal.

d. A control rod block will occur due to a flow unit comparator trip.

Answer	d	Exam Level	B	Cognitive Level	Comprehension	Facility:	Hope Creek	Exam Date:	06/17/2003

Tier:	Plant Systems	RO Group	1	SRO Group	1	215005K604
215005	Average Power Range Monitor/Local Power Range Monitor System				Record Number	68

K6.	Knowledge of the effect that a loss or malfunction of the following will have on the APRM/LPRM:		
K6.04	Trip units	3.1	3.2

Explanation of Answer

Justification:
A control rod block will occur due to a flow unit comparator trip. Correct. The comparator trip will alarm at 10 percent difference between A and B or A and C flow units.
A control rod block will occur due to flow biased neutron flux upscale.Incorrect. FBNF RB trip is set at 42 percent APRM power with the mode switch in Run.
A half scram will occur due to a flow unit "inop" signal. Incorrect. Flow unit inop trip by itself does not cause half scram.
A full scram will occur due to flow biased neutron flux upscale. Incorrect

Reference Title

HC.OP-SO.SE-0001

Learning Objectives

APRM00E002	(R) Given a labeled diagram of, or access to, the APRMS/Flow Unit controls located on control room panels 10C608/10C651: Explain the function of each indicator, IAW the Student Handout. Assess the plant conditions that cause each indicator to light or extinguish, IAW the Student Handout. Predict the effect of each control switch on the APRMS/Flow Units, IAW the Student Handout. Select the conditions or permissives required for the control switches to perform their intended function, IAW the Student Handout.

Material Required for Examination	None		
Question Source:	INPO Exam Bank	Question Modification Method:	Editorially Modified
Question Source Comments:	INPO BANK QID# 14154 Peach Bottom 03/26/2001		

Given the following conditions:

- The plant has been manually scrammed.
- A normal reactor cooldown is in progress.
- The reference leg backfill system is out of service.

Then, annunciator (A7-C5) "RPV LEVEL 4" is received. The operator investigates and observes that reactor water level "notching" is occurring.

Which of the following is the most accurate indicated water level from the indicator that is experiencing "notching"?

a. An average of the water levels from the top AND bottom of the "notch".

b. The water level at the bottom of the "notch". *Correct answer is "b".

c. The water level at the top of the "notch".

d. An average of the water levels from all indicators that are "notching".

* Answer	c	Exam Level	B	Cognitive Level	Memory	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Plant Systems	RO Group	1	SRO Group	1				216000A104
216000	Nuclear Boiler Instrumentation							Record Number	69

A1. Ability to predict and/or monitor changes in parameters associated with operating the NUCLEAR BOILER INSTRUMENTATION controls including:

A1.04	System venting	2.6	2.8

Explanation of Answer

Reference Title

NOH01RXINST-01, Industry events.

Learning Objectives

RXINSTE011	(R) Given plant problems/industry events associated with the Nuclear Boiler Instrumentation System: Discuss the root cause of the plant problem/industry event IAW the associated plant problem/industry event document. Discuss the HCGS design and/or procedural guidelines that mitigate/reduce the likelihood of the problem/industry event at HCGS IAW the associated plant problem/industry event document. Discuss the "lessons learned" from the problem/event IAW the associated plant problem/industry event document.

Material Required for Examination	None		
Question Source:	Facility Exam Bank	Question Modification Method:	Direct From Source
Question Source Comments:	VISION BANK QID# Q55159		

Given the following conditions:

- The plant is operating at 30 percent power.
- The I&C department reports that reactor pressure transmitter SA-PT-N403B on instrument rack C027 has failed it's sensor calibration.
- I&C also states that the pressure transmitter must be replaced.

Based on these conditions, declare that the associated:

a. channel must be placed in the tripped condition within one hour.

b. channel must be placed in the tripped condition within 24 hours.

c. system is inoperable and returned to operable status within 72 hours.

d. channel must be placed in the tripped condition within twelve hours.

Answer	a	Exam Level	S	Cognitive Level	Application	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Plant Systems	RO Group	1	SRO Group	1				216000G222
216000	Nuclear Boiler Instrumentation							Record Number	70
2.2	Equipment Control								
2.2.22	Knowledge of limiting conditions for operations and safety limits.							3.4	4.1

Explanation of Answer

Justification:
55.43(2) Facility operating limitations in the Technical Specification and their bases.
channel must be placed in the tripped condition within one hour. CORRECT. Operator must determine the transmitter feeds RRCS Logic from M-42-1 Sht 1 & 2. The operator then determines LCO 3.3.4.1 ATWS RPT action b. is applicable.
channel must be placed in the tripped condition within 24 hours. INCORRECT. Action for one NSSSS transmitter not common to RPS.
system inoperable and returned to operable status within 72 hours. INCORRECT. Action for one trip system inoperable.
channel must be placed in the tripped condition within twelve hours. INCORRECT. Action for RPS pressure transmitter.

Requires TS section 3.3.4
M-42-1 SHT 2 REV 14 TECH SPEC 3.3.4.1.B

Reference Title

Tech Spec 3.3.4

Learning Objectives

RRCS00E006 (R) Given a scenario of applicable conditions and access to technical specifications:
a. Choose those sections which are applicable to the Redundant Reactivity Control System (ATWS Circuitry), IAW HCGS Technical Specifications.
b. Evaluate Redundant Reactivity Control System operability and determine required actions based on system inoperability (ATWS Circuitry), IAW HCGS Technical Specifications. SRO/STA ONLY
c. Explain the bases for those technical specifications associated with the Redundant Reactivity Control System (ATWS Circuitry), IAW HCGS Technical Specifications.

Material Required for Examination Tech Spec sections 3.3.1 through 3.3.4; P&ID M-42-1 Sheet 1

Question Source: Facility Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments:	Vision Bank QID# Q54823 Significantly modified.

Given the following conditions:

- The plant has scrammed following extended operation at 100% power.
- The MSIVs are closed.
- The RCIC system is being operated to maintain RPV water level constant.
- RCIC flow controller BD-FIC-R600 is in AUTO with flow set at 600 gpm.
- Reactor water level is steady when the control oil supply line to RCIC Turbine Governor Valve BD-HV-4283 ruptures.

Which one of the following describes the initial plant/system response to this line break?

a. RCIC pump discharge flow indication (BD-FIC-R600) decreases, RPV water level decreases.

b. RCIC turbine speed indication (BD-SI-4280-1) increases, RPV water level increases.

c. RCIC turbine steam inlet pressure indication (BD-PI-R602) increases, RPV water level increases.

d. RCIC pump discharge pressure indication (BD-PI-R601) decreases, RPV water level decreases.

Answer	b	Exam Level	B	Cognitive Level	Comprehension	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Plant Systems			RO Group	1	SRO Group	1		217000K301
217000	Reactor Core Isolation Cooling System (RCIC)							Record Number	71

K3. Knowledge of the effect that a loss or malfunction of the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) will have on following:

K3.01 Reactor water level 3.7 3.7

Explanation of Answer

Justification:
RCIC turbine speed indication (BD-SI-4280-1) increases, RPV water level increases. Correct. Loss of oil pressure causes the governor valve to fail full open. Turbine speed rises, flow increases, RPV level rises.
RCIC pump discharge flow indication (BD-FIC-R600) decreases, RPV water level decreases. Incorrect. Flow and RPV level rises.
RCIC turbine steam inlet pressure indication (BD-PI-R602) increases, RPV water level increases. Incorrect. Steam pressure will stay the same or lower due to increased steam flow.
RCIC pump discharge pressure indication (BD-PI-R601) decreases, RPV water level decreases. Incorrect. Discharge pressure increases, Level increases.

Reference Title

NOH01RCIC00-00

Learning Objectives

RCIC00E023 (R) Given any of the following and appropriate control room reference material, evaluate and determine the effect on the RCIC system, IAW the RCIC System Lesson Plan:
a. A given valve opening or closure
b. Loss of DC or AC power supply
c. Inadequate system flow
d. An oil system malfunction
e. Failure of the RCIC Gland Seal Condenser Vacuum Pump
f. Loss of room cooling
g. Rupture disc failure on the RCIC exhaust
h. Steam line break
i. Low condensate storage tank level

Material Required for Examination	None		
Question Source:	Facility Exam Bank	Question Modification Method:	Editorially Modified
Question Source Comments:	Vision Bank QID# Q54875		

Given the following conditions:

- The Reactor Core Isolation Cooling (RCIC) system flow controller has failed full downscale demanding a "0" gpm flowrate.
- The controller is in AUTO.

Which of the following is the RCIC turbine response upon receipt of a valid initiation signal for the given conditions?

RCIC will start, accelerate...

a. to and run continuously at approximately 4000 rpm.

b. and trip on mechanical overspeed.

c. then slow to a stop.

d. then will slow to and run at a low speed.

Answer	d	Exam Level	B	Cognitive Level	Comprehension	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Plant Systems	RO Group	1	SRO Group	1		217000K502		
217000	Reactor Core Isolation Cooling System (RCIC)							Record Number	72

K5. Knowledge of the operational implications of the following concepts as they apply to REACTOR CORE ISOLATION COOLING SYSTEM (RCIC):

K5.02 Flow indication 3.1 3.1

Explanation of Answer: Justification:
The ramp generator runs RCIC to about 4000 rpm until the flow signal comes out of saturation at which time the low signal will control -> RCIC running at min speed.

Reference Title

NOH01RCIC00-00

Learning Objectives

RCIC00E022	(R) Given RCIC turbine control system failures, evaluate and determine the effect on the RCIC system, IAW the RCIC System Lesson Plan.

Material Required for Examination: None

Question Source: Facility Exam Bank | **Question Modification Method:** Editorially Modified

Question Source Comments: Vision QID# Q53389

A plant transient is in progress with current plant conditions as follows:

- Drywell Pressure is 3.6 psig and rising at 0.2 psi/min.
- Reactor Level is -35" and lowering at 1.5 in./min.
- Reactor Pressure is 810 psig and lowering at 10 psi/min.
- HPCI Pump is tagged for maintenance.
- All other ECCS systems have performed as expected.

Assuming NO operator action, ADS SRVs will open immediately when:

a. Level 1 is reached.

b. Level 1 is reached and the 105 second timer times out.

c. Top of Active Fuel (TAF) is reached.

d. Top of Active Fuel (TAF) is reached and the 105 second timer times out.

Answer	b	Exam Level	R	Cognitive Level	Comprehension	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Plant Systems			RO Group	1	SRO Group	1		218000A402
218000	Automatic Depressurization System							Record Number	73
A4.	Ability to manually operate and/or monitor in the control room:								
A4.02	ADS logic initiation								4.2 4.2

Explanation of Answer

Justification:
Level 1 is reached and the 105 second timer times out. Correct. High DW pressure is present. ADS logic will wait until Level 1 is reached to start the 105 second timer. After 105 seconds, ADS will open the ADS SRVs
Level 1 is reached. Incorrect. Starts the 105 sec timer.
Top of Active Fuel (TAF) is reached. Incorrect. Level at which ADS is manually actuated if ECCS is available and running.
Top of Active Fuel (TAF) is reached and the 105 second timer times out. Incorrect. Wrong setpoint.

Reference Title

HC.OP-SO.SN-0001

Learning Objectives

ADSSYSE007 (R) Given a set of conditions and a drawing of the controls, instrumentation and/or alarms located in the Control Room, identify the status of the Automatic Depressurization System by evaluation of the controls/instrumentation/alarms, IAW the Automatic Depressurization System Lesson Plan.

Material Required for Examination None

Question Source: INPO Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: INPO Bank QID # 20341 Clinton 07/23/2001 Modified for Hope Creek.

Given the following conditions:

- The plant has scrammed due to a loss of offsite power.
- HPCI and RCIC fail to start both automatically and manually.
- RPV water level lowers below -129".
- The "ADS CHANNEL INITIATION PENDING" annunciators for both logic channels are received.
- The RO is directed to "Inhibit ADS".
- The operator inadvertently arms and depresses the "LOGIC B MAN INIT" and "LOGIC F MAN INIT" pushbuttons.

Select the statement below which describes the response of ADS.

a. ADS will initiate immediately, regardless of Core Spray and RHR status.

b. ADS will initiate in 105 seconds, only if Core Spray pumps A & C or RHR pump A or C are running.

c. ADS will initiate immediately, only if Core Spray pumps B & D or RHR pump B or D are running.

d. ADS will initiate in 105 seconds, regardless of Core Spray and RHR status.

Answer	a	Exam Level	B	Cognitive Level	Comprehension	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Plant Systems	RO Group	1	SRO Group	1				218000K402
218000	Automatic Depressurization System							Record Number	74

K4.	Knowledge of AUTOMATIC DEPRESSURIZATION SYSTEM design feature(s) and/or interlocks which provide for the following:		
K4.02	Allows manual initiation of ADS logic	3.8	4.0

Explanation of Answer

ustification: IAW
· ADS will initiate immediately, regardless of core spray and RHR status.- Correct
· ADS will initiate immediately, if Core Spray pumps A & C or RHR pump A or C are running - Incorrect, the status of core spray/RHR does not effect manual initiation
· ADS will initiate in 105 seconds, regardless of core spray and RHR status.- Incorrect, Manual initiation bypasses the 105 second timer
· ADS will initiate in 105 seconds, if Core Spray pumps A & C - or RHR pump A or C are running - Incorrect, Manual initiation bypasses the 105 second timer

Reference Title

HC.OP-SO.SN-0001 section 3.3.1

Learning Objectives

ADSSYSE006 (R) Given a labeled diagram/drawing of, or access to, the Automatic Depressurization System controls/indication bezel, IAW the Automatic Depressurization System Lesson Plan:
a. Explain the function of each indicator.
b. Assess plant conditions which will cause the indicator to light or extinguish.
c. Determine the effect of each control on the Automatic Depressurization System .
d. Assess plant conditions or permissives required for the control switches to perform their intended function.

Material Required for Examination	Figure of ADS logic pushbuttons.		
Question Source:	Facility Exam Bank	Question Modification Method:	Editorially Modified

Question Source Comments:	Vision Bank QID# Q54168 editiorially modified.

Given the following conditions:

- B RHR pump is running in Suppression Pool Cooling at rated flow.
- B RHR pump trips on an electrical fault in the motor.

Which one of the following describes the response of BC-HV-F007B Minimum Flow valve?

a. Automatically closes after a 10 second delay.

b. Automatically opens immediately when the pump trips.

c. Remains closed after the pump trip.

d. Remains open after the pump trip.

Answer	c	Exam Level	R	Cognitive Level	Comprehension	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Plant Systems		RO Group	2	SRO Group	2			219000A404
219000	RHR/LPCI: Torus/Suppression Pool Cooling Mode							Record Number	75
A4.	Ability to manually operate and/or monitor in the control room:								
A4.04	Minimum flow valves							3.0	2.9

Explanation of Answer

Justification:
Remains closed after the pump trip. Correct. With the RHR pump breaker open, the Min Flow Valve HV F007B will remain closed. The RHR pump must be running for the valve to open on low flow.
Automatically closes after a 10 second delay. Incorrect. Sequence on pump startup with flow above 1250 gpm.
Automatically opens immediately when the pump trips. Incorrect. The RHR pump must be running for the valve to open on low flow.
Remains open after the pump trip. Incorrect. Valve will be initially closed due to rated flow. Valve will remain closed.

Reference Title

HC.OP-SO.BC-0001 3.3.6

Learning Objectives

RHRSYSE011 Given a labeled drawing of, or access to the Residual Heat Removal System controls/indication on 10C650:

a. Explain the function of each indicator IAW the RHR System Lesson Plan.
b. Assess plant conditions which will cause the indicators to light or extinguish IAW the RHR System Lesson Plan.
c. Determine the effect of each control on the RHR System IAW the RHR System Lesson Plan.
d. Assess plant conditions or permissives required for the control switches/pushbuttons to perform their intended functions IAW the RHR System Lesson Plan.

Material Required for Examination None

Question Source: New **Question Modification Method:**

Question Source Comments:

While performing a RHR system test, the breaker for Torus Spray Isolation Valve BC-HV-F027B trips.

Which one of the following Motor Control Centers (MCC) powers this valve?

a. 10B222

b. 10B323

c. 10B242

d. 10B563

Answer	a	**Exam Level**	B	**Cognitive Level**	Memory	**Facility:**	Hope Creek	**Exam Date:**	06/17/2003
Tier:	Plant Systems	**RO Group**	2	**SRO Group**	2				219000K201

219000	RHR/LPCI: Torus/Suppression Pool Cooling Mode	**Record Number**	76
K2.	Knowledge of electrical power supplies to the following:		
K2.01	Valves		2.5 2.9

Explanation of Answer

Justification:
10B222 Correct. Powered from 52-222083 which is a B channel 1E MCC.
10B323 Incorrect. B channel Non 1E MCC powered from 1E power which is shed on a LOCA Level 1 signal.
10B242 Incorrect. D Channel 1E MCC with similar valve load to 10B222.
10B563 Incorrect. B channel 1E MCC which powers Station Service Water components.

Reference Title

E-0012 Sheet 2

Learning Objectives

RHRSYSE008	(R) Given a system which physically connects to or is required to support the operation of the RHR System or components therein, explain the function of the supporting system, IAW the RHR System Lesson Plan.

Material Required for Examination None

Question Source: New **Question Modification Method:**

Question Source Comments:

The Containment Hydrogen Recombiners are directed to be placed in service following a LOCA in the drywell. LOCA conditions still are present.

Which one of the following actions is the MINIMUM required to accomplish this task?

a. Override NSSSS isolation and reset PCIS isolation

b. Override PCIS isolation only.

c. Reset NSSSS isolation only.

d. Reset NSSSS isolation and override PCIS isolation.

Answer	b	Exam Level	B	Cognitive Level	Memory	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Plant Systems	RO Group	1	SRO Group	1				223002K408
223002	Primary Containment Isolation System/Nuclear Steam Supply Shut-Off							Record Number	77

K4.	Knowledge of PCIS/NSSSS design feature(s) and/or interlocks which provide for the following:		
K4.08	Manual defeating of selected isolations during specified emergency conditions	3.3	3.7

Explanation of Answer

Justification:
Override PCIS isolation only. Correct. PCIS isolation override is necessary to open H2 Recombiner flowpath
Override NSSSS isolation and reset PCIS isolation. Incorrect. NSSSS does not have override capability. NSSSS provides isolation input to PCIS isolation
Reset NSSSS isolation only. Incorrect. Only half of input required to open valves.
Reset NSSSS isolation and override PCIS isolation. Not minimum. NSSSs isolation reset is not necessary.

Reference Title

HC.OP-SO.SM-0001

Learning Objectives

NSSSS0E003	(R) Provided access to control room references: a. Determine the source of electrical power for the NSSSS logic channels IAW the NSSSS Lesson Plan. b. Predict plant response to a loss of power to the NSSSS power supplies IAW the NSSSS Lesson Plan.

Material Required for Examination	None		
Question Source:	New	Question Modification Method:	
Question Source Comments:			

Given the following conditions:

- The plant is operating normally at 100 percent power.
- Overhead annunciation "RHR LOOP A TROUBLE A6-B1" alarms.

Which one of the following conditions would cause the alarm?

a. The AP228 ECCS Jockey Pump tripped on overload.

b. The CP228 ECCS Jockey Pump suction strainer clogged.

c. The A RHR Pump room has a high water level.

d. The A RHR Loop Test Return Line manual isolation valve closed.

Answer	b	Exam Level	R	Cognitive Level	Memory	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Plant Systems		RO Group	2	SRO Group	1			226001A404
226001	RHR/LPCI: Containment Spray System Mode							Record Number	78

A4. Ability to manually operate and/or monitor in the control room:

A4.04 Keep fill system 2.8 2.7

Explanation of Answer

Justification:
The CP228 ECCS Jockey Pump suction strainer clogged. Correct. A clogged suction strainer will cause jockey pump discharge pressure to drop. Low discharge pressure causes low alarm on PSL - N654A Low Discharge Pressure which in turn causes the A6-B1 alarm.
The AP228 ECCS Jockey Pump tripped on overload. Incorrect. AP228 feeds the HPCI system.
The A RHR Pump room is flooded. Incorrect. Causes RHR Pump Room Flooded alarm.
The A RHR Loop Test Return Line manual isolation valve closed. Incorrect. Would cause a high discharge pressure of the Jockey Pump but below the high pressure setpoint of 380 psig for detection of leakage past Loop Isolation valves.

Reference Title

HC.OP-AR.ZZ-0004 Attachment B1

Learning Objectives

RHRSYSE012	Given a set of conditions and a drawing of the controls, instrumentation and/or alarms located in the main control room, assess the status of the Residual Heat Removal System or its components by evaluation of the controls/instrumentation/alarms IAW the RHR System Lesson Plan.

Material Required for Examination Figure of alarm window A6-B1

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following conditions:

- A LOCA occurred in the drywell.
- Drywell pressure is 10 psig.
- One of two Drywell pressure transmitters associated with this loop subsequently failed to zero psig.
- B RHR Injection Valve F017B is closed by the operator in preparation to spray.

Which one of the following describes the Drywell Spray Isolation Valve response when the operator is directed to place Drywell Spray in service?

a. Drywell spray inboard isolation valve only opens.

b. Drywell spray outboard isolation valve only opens.

c. NEITHER Drywell spray valves will open.

d. Both Inboard and Outboard Drywell spray isolation valves open.

Answer	d	Exam Level	B	Cognitive Level	Memory	Facility:	Hope Creek	Exam Date:	06/17/2003

Tier:	Plant Systems	RO Group	2	SRO Group	1	226001K108
226001	RHR/LPCI: Containment Spray System Mode			Record Number		79

K1. Knowledge of the physical connections and/or cause- effect relationships between RHR/LPCI: CONTAINMENT SPRAY SYSTEM MODE and the following:

K1.08	Nuclear boiler instrumentation	3.2	3.4

Explanation of Answer

Justification:
Both Inboard and Outboard Drywell spray isolation valves open. Correct. Only one DW pressure transmitter above setpoint is required to open both valves once the High Drywell pressure initially sealed in.
NEITHER Drywell spray valves will open. Incorrect. Both will open.
Drywell spray outboard isolation valve only opens. Incorrect. Both valves will open.
Drywell spray inboard isolation valve only opens. Incorrect. Both valves will open.

Reference Title

HC.OP-SO.BC-0001 3.3.5

Learning Objectives

RHRSYSE010	(R) Given procedure HC.OP-SO.BC-0001, Residual Heat Removal System Operation, explain the listed prerequisites, precautions, and/or limitations during operation IAW HC.OP-SO.BC-0001.

Material Required for Examination: None

Question Source: New | **Question Modification Method:**

Question Source Comments:

Given the following conditions:

- Fuel Pool level is at the normal water level.

Which of the following describes the change in skimmer surge tank level if the first fuel pool cooling pump is started with the discharge valve full open AND the weir gate set at its lowest position?

Skimmer surge tank level . . .

a. increases and then returns to the level that existed prior to starting the pump.

b. increases to a level higher than existed prior to starting the pump.

c. decreases until the pump trips on low tank level.

d. decreases and then increases to ~~the~~ A level lower than existed prior to starting the pump.

Answer	d	Exam Level	B	Cognitive Level	Memory	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Plant Systems			RO Group	3	SRO Group	3		233000A101
233000	Fuel Pool Cooling and Clean-up							Record Number	80

A1. Ability to predict and/or monitor changes in parameters associated with operating the FUEL POOL COOLING AND CLEAN-UP controls including:

A1.01 Surge tank level 2.6 2.9

Explanation of Answer

Justification:
decreases until the pump trips on low tank level. Incorrect. The pump will not trip on low SST level Because the level reduction will be slight as long as pool level is at the normal level.
increases to a level higher than existed prior to starting the pump. Incorrect. SST level will lower.
increases and then returns to the level that existed prior to starting the pump. Incorrect. Decreases.
decreases and then increases to the level lower than existed prior to starting the pump. Correct..Lowers slightly then stabilizes slightly lower due to the restriction at the overflow pipe..

Reference Title

HC.OP-AR.ZZ-0013 Attachment D5

Learning Objectives

FPCC00E008 (R) Concerning spent fuel storage pool water level, summarize, from memory, the following IAW the Fuel Pool Cooling and Cleanup (FPCCS) System Lesson Plan:
a. How normal level is controlled
b. Sources of makeup to the spent fuel storage pool

Material Required for Examination None

Question Source: INPO Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: INPO Bank QID# 19106 11/20/2000 Lasalle.

Which one of the following describes the interfaces of Station Auxiliaries Cooling System (SACS) and Station Service Water (SSW) for the Fuel Pool Cooling and Cleanup System?

I - SSW A Loop as a level makeup source.
II - SACS A Loop as a level makeup source.
III - SACS B Loop as a cooling water source.
IV - SSW B Loop as a level makeup source.

a. I and III only.

b. III and IV only.

c. I, II and III only.

d. I, III, and IV only.

Answer	d	Exam Level	B	Cognitive Level	Memory	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Plant Systems	RO Group	3	SRO Group	3				233000K109
233000	Fuel Pool Cooling and Clean-up							Record Number	81

K1. Knowledge of the physical connections and/or cause- effect relationships between FUEL POOL COOLING AND CLEAN-UP and the following:

K1.09 Component cooling water systems 2.6 2.6

Explanation of Answer

Justification: Both Loops of SSW can be valved in for makeup source in an emergency. SACS is a cooling water source. SACS is NOT a makeup source.
I, III, and IV only. Correct.
I and III only. Incorrect. SSW B also MU source
III and IV only. Correct. SSW Loop A also MU source.
I, II and III only. Correct. Sacs is not a MU Source.

Reference Title

M-10 and M-11

Learning Objectives

FPCC00E006	From memory, list/identify the cooling medium used in the FPCCS Heat Exchangers, IAW the Fuel Pool Cooling and Cleanup (FPCCS) System Lesson Plan.
FPCC00E008	(R) Concerning spent fuel storage pool water level, summarize, from memory, the following IAW the Fuel Pool Cooling and Cleanup (FPCCS) System Lesson Plan: a. How normal level is controlled b. Sources of makeup to the spent fuel storage pool

Material Required for Examination	None		
Question Source:	New	Question Modification Method:	
Question Source Comments:			

Given the following conditions:

- A new fuel bundle is grappled and lifted in the Spent Fuel Pool for placement in the RPV.
- The "Slack Cable" light remains lit.
- The refueling platform main hoist load cell indicates 0 pounds.
- The bundle is lowered, reseated, and released.

Which of the following operations can continue?

a. Control rod removal from the RPV using the refueling bridge main hoist.

b. Control rod removal from the RPV using the frame mounted auxiliary hoist.

c. Fuel transfer within the Spent Fuel Pool using the monorail auxiliary hoist.

d. Fuel transfer within the Spent Fuel Pool using the frame mounted auxiliary hoist.

Answer	b	**Exam Level**	S	**Cognitive Level**	Comprehension	**Facility:**	Hope Creek	**Exam Date:**	06/17/2003
Tier:	Plant Systems	**RO Group**	3	**SRO Group**	2		234000G222		
234000	Fuel Handling Equipment							**Record Number**	82
2.2	Equipment Control								
2.2.22	Knowledge of limiting conditions for operations and safety limits.							3.4	4.1

Explanation of Answer

Justification:
SRO 10CFR55.43 (7) Fuel handling facilities and procedures.
SRO 10CFR55.43 (2) Facility operating limitations in the Technical Specifications and their bases.
Justification:
Control rod removal from the RPV using the frame mounted auxiliary hoist. -Correct. The auxiliary hoist can be used for control rod removal IAW AP-108. SH.OP 108 action requires suspension of inoperable equipment use involving control rod or fuel assembly movement . Continued control rod movement with aux hoist is allowed.
Control rod removal from the RPV using the refueling bridge main hoist.. -Incorrect-The main mast cannot be used for core alts. See SH.OP 108 required action
Fuel transfer within the Spent Fuel Pool using the monorail auxiliary hoist. -Incorrect- Not allowed to move fuel with the Aux hoists. Fuel bundle is too heavy for aux hoists.
Fuel transfer within the Spent Fuel Pool using the frame mounted auxiliary hoist. -Incorrect-Not allowed to move fuel with the aux hoists. Fuel bundle is too heavy for aux hoists.

Reference Title

SH.OP-AP.ZZ-0108 Exhibit 3

HCGS T.S. 3.9.1

Learning Objectives

REFUELE006	From memory, discuss the indications available in the main Trolley Operator Cab IAW the Student Handout.
REFUELE012	(R) Given a scenario of applicable operating conditions and access to Technical Specifications: a. Choose those sections which are applicable to the refueling platform and associated equipment IAW HCGS Technical Specifications. b. Evaluate Refuel Platform operability and determine required actions based upon system operability IAW HCGS Technical Specifications. c. Explain the basis for those Tech Spec items associated with the refuel platform IAW HCGS Technical Specifications. (SRO only)

Material Required for Examination	HC Tech Spec section 3.9.1		
Question Source:	Facility Exam Bank	**Question Modification Method:**	Significantly Modified

Question Source Comments:	Vision Bank QID# Q56966 Significantly modified.

Given the following conditions:

- The plant scrammed from an inadvertent MSIV closure.
- EOP-101 has been entered.
- LO-LO Set is controlling pressure between 905 and 1017 psig.
- The STA reports Drywell pressure has risen from 0.5 psig to 1.4 psig.

Based on these indications, what is causing the drywell pressure to rise and what action is required to mitigate the event?

a. H SRV is stuck closed. Cycle H SRV.

b. H SRV tailpipe vacuum breaker is stuck open. Place H SRV handswitch to closed position.

c. P SRV is stuck open. Cycle P SRV.

d. P SRV tailpipe vacuum breaker is stuck closed. Place P SRV handswitch to open position.

Answer	b	Exam Level	R	Cognitive Level	Comprehension	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Plant Systems		RO Group	1	SRO Group	1			239002A201
239002	Relief/Safety Valves							Record Number	83

A2. Ability to (a) predict the impacts of the following on the RELIEF/SAFETY VALVES; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:

A2.01 Stuck open vacuum breakers 3.0 3.3

Explanation of Answer

Justification:
H SRV tailpipe vacuum breaker is stuck open. Place H SRV handswitch to closed position. Correct. RPV Pressure is within the range for H SRV operation. Rising DW pressure is caused by the H SRV tailpipe vacuum break open and discharging to the drywell airspace.
H SRV is stuck closed. Cycle H SRV. Incorrect. Pressure range indicates H SRV is operating on LO-LO Set setpoints. If SRV Instrument gas supply line ruptured, DW pressure would rise. Wrong action.
P SRV is stuck open. Cycle P SRV. Incorrect. RPV pressure would continue to lower. Wrong action.
P SRV tailpipe vacuum breaker is stuck closed. Place P SRV handswitch to open position. Incorrect. A stuck closed SRV tailpipe vacuum breaker can not be diagnosed from the info given. Wrong action.

Reference Title

EOP-101

Learning Objectives

EO101PE008 (R) Given any step of the procedure, describe the reason for performance of that step and/or expected system response to control manipulations prescribed by that step.

Material Required for Examination EOP- 101 flowchart

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following conditions:

- T = 0 sec LOCA occurs.
- T = 2 sec High Drywell pressure signal is generated and all equipment responds as required.
- T = 20 sec ADS CH D INITIATION PENDING (RPV Level 1) annunciators alarm.
- T = 24 sec ADS CH B INITIATION PENDING (RPV Level 1) annunciators alarm.
- T = 48 sec Drywell pressure drops to 0 psig due to a large pipe break at a penetration.
- All control rods are full in.
- EOP execution is in progress.

When will ADS be initiated manually or automatically?

a. Before RPV level reaches -190 inches; manually.

b. When T = 425 seconds; automatically.

c. When T = 302 seconds; automatically.

d. After RPV level reaches -200 inches; manually.

Answer	a	Exam Level	B	Cognitive Level	Comprehension	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Plant Systems		RO Group	1	SRO Group	1			239002A204
239002	Relief/Safety Valves							Record Number	84

A2. Ability to (a) predict the impacts of the following on the RELIEF/SAFETY VALVES; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:

A2.04 ADS actuation 4.1 4.2

Explanation of Answer

Justification:
Before RPV level reaches -190 inches; manually. Correct. EOP 101 Step ALC-9.ADS is manually inhibited at -129 inches, ADS blowdown before -190" by manual operator action.
When T = 425 seconds; automatically. Incorrect. ADS will be inhibited at -129 inches.
When T = 302 seconds; automatically. Incorrect. ADS will be inhibited at -129 inches.
After RPV level reaches -200 inches; manually. Incorrect. All equipment responds as required. Therefore steam cooling is not required.

Reference Title

HC.OP-SO.SN-0001

HC.OP-EO.ZZ-0101

Learning Objectives

ADSSYSE007	(R) Given a set of conditions and a drawing of the controls, instrumentation and/or alarms located in the Control Room, identify the status of the Automatic Depressurization System by evaluation of the controls/instrumentation/alarms, IAW the Automatic Depressurization System Lesson Plan.

Material Required for Examination: EOP Flowcharts

Question Source: Facility Exam Bank **Question Modification Method:** Direct From Source

Question Source Comments: VISION Bank QID # Q56192

Given the following conditions:

- Reactor power is 50%.
- ALL Turbine Control Valves fail OPEN.
- The MSIVs fail to automatically close.
- The reactor was scrammed and MSIVs are closed manually.

Determine which of the following combinations of reactor power and reactor pressure would indicate that a Safety Limit violation had occurred?

	Reactor Power	RPV Pressure
a.	15%	750 psig
b.	24%	770 psig
c.	28%	775 psig
d.	32%	810 psig

Answer	c	Exam Level	S	Cognitive Level	Comprehension	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Plant Systems	RO Group	1	SRO Group	1				241000G222
241000	Reactor/Turbine Pressure Regulating System							Record Number:	85
2.2	Equipment Control								
2.2.22	Knowledge of limiting conditions for operations and safety limits.								3.4 4.1

Explanation of Answer

JUSTIFICATION:
55.43(2) Facility operating limitations in the Technical Specifications and their bases.
Correct Answer: 28% 775 psig - Power is greater than 25% with pressure less than 785 psig.
The following distractors are incorrect as follows:
32% 810 psig - Pressure is greater than 785 psig so no indication of a safety limit violation.
24% 770 psig - Power is less than 25% with pressure less than 785 psig.
15% 750 psig - Power is less than 25% with pressure less than 785 psig.

Reference Title

HCGS Tech Specs - 2.1.1

Learning Objectives

TECSPCE010	(R) Given specific plant operating conditions and a copy of the Hope Creek Generating Station Technical Specifications, evaluate plant/system operability and determine required actions (if any) to be taken. (SRO/STA Only)

Material Required for Examination None

Question Source: Facility Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments: VISION BANK QID# Q56697 Significantly modified.

Which one of the following decribes the pressure rise at the Main Turbine inlet pressure and reactor steam dome pressure as power is increased from syncronization to rated thermal power? Assume reactor power change at a constant ramp rate.

Main Turbine inlet pressure rise is ____________ and reactor steam dome pressure rise is ____________ .

a. Linear; Linear.

b. Linear; Non-Linear.

c. Non-Linear; Linear.

d. Non-Linear; Non-Linear.

Answer	b	**Exam Level**	B	**Cognitive Level**	Comprehension	**Facility:**	Hope Creek	**Exam Date:**	06/17/2003
Tier:	Plant Systems	**RO Group**	1	**SRO Group**	1				241000K504
241000	Reactor/Turbine Pressure Regulating System							**Record Number**	86

K5. Knowledge of the operational Implications of the following concepts as they apply to REACTOR/TURBINE PRESSURE REGULATING SYSTEM:

K5.04 Turbine inlet pressure vs. reactor pressure 3.3 3.3

Explanation of Answer

Justification:
Linear; Non-Linear. Correct. Main turb inlet presssure rises from 920 to 950 psig at 3.33 percent steam flow to 1 psig rise. Reactor pressure rises from 920 to 1005 psig. Reactor pressure rises higher due to the differential pressure caused by steam line flow increases with increased flow.
Linear; Linear. Incorrect. Reactor pressure rise is non-linear
Non-Linear; Linear. Incorrect. MT inlet pressure rise is linear. Reactor pressure rise is non-linear
Non-Linear; Non-Linear. Incorrect. MT inlet pressure rise is linear.

Reference Title

NOH01EHCLOG-00 figure 2

Learning Objectives

EHCLOGE002 (R) Given plant conditions evaluate the cause-effect relationship between the pressure regulating system and the following IAW the Lesson Plan:
Reactor Power
Reactor Pressure
Steam Flow
Reactor Water Level

Material Required for Examination None

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following conditions:

- The plant is operating at 100% power.
- Main Turbine testing is in progress.
- The LOCKED OUT pushbutton on 10C650E has been depressed, energizing the lockout valve.

Which of the following describes the effect of energizing the Lockout Valve?

a. The Master Trip Solenoid is bypassed to prevent depressurizing the Emergency Trip System.

b. All Turbine trips are bypassed to allow for testing.

c. All Turbine trips EXCEPT for the mechanical overspeed trip are bypassed.

d. Only the Turbine mechanical overspeed trip is bypassed.

Answer	d	Exam Level	R	Cognitive Level	Memory	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Plant Systems			RO Group	2	SRO Group	2		245000K503
245000	Main Turbine Generator and Auxiliary Systems							Record Number	87

K5. Knowledge of the operational implications of the following concepts as they apply to MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS:

K5.03 Hydraulically operated valve operation 2.6 2.6

Explanation of Answer

Justification: Only the Turbine mechanical overspeed trip is bypassed. Correct. The lockout valve shifts the upper lockout valve spool such that high pressure fluid from the MTV is blocked and high pressure fluid from the FAS header is supplied to the steam admission valve disc dump valves. Does not bypass the Master Trip Solenoid.
The Master Trip Solenoid is bypassed to prevent depressurizing the Emergency Trip System. Incorrect. MTS is not bypasses.It remains active.
All Turbine trips are bypassed to allow for testing. Incorrect. Only Mechanical Overspeed trip is bypassed to allow testing.
All Turbine trips EXCEPT for the mechanical overspeed trip are bypassed. Incorrect. Reverse of actual.

Reference Title

HC.OP-SO.CH-0001

Learning Objectives

EHCOILE009 (R) Concerning the hydraulic trip system, from memory summarize/identify the purpose of the following components.
Mechanical Trip Valve
Lockout Valve
Master Trip Solenoid Valve

Material Required for Examination	None
Question Source:	Facility Exam Bank
Question Modification Method:	Direct From Source
Question Source Comments:	Vision Bank QID# Q54744

Given the following conditions:

- The plant is operating at 100 percent power.
- The Hotwell level control is selected to 'A' on 10C651A.
- A large pipe break occurs on the tube side of the in service SJAE Condenser.

Which one of the following describes the plant response and what operator action(s) will be required?

a. 'A' Primary Condensate Pump trips; Reduce Reactor power using the 'Stuff Sheet'.

b. All Primary Condensate Pumps trip; Trip all Secondary Condensate and Feedwater Pumps.

c. All Primary Condensate Pumps trip; Lock the reactor mode switch in Shutdown.

d. 'A' Primary Condensate Pump trips; Verify Full Reactor Recirc and Feedwater pump runbacks.

Answer:	c	Exam Level	R	Cognitive Level	Comprehension	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Plant Systems		RO Group	2	SRO Group	3			256000A206
256000	Reactor Condensate System							Record Number	88

A2.	Ability to (a) predict the impacts of the following on the REACTOR CONDENSATE SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:		
A2.06	Low hotwell level	3.2	3.2

Explanation of Answer

Justification:
All Primary Condensate Pumps trip; Lock the reactor mode switch in Shutdown. Correct. Lowering Hotwell level will result from a condensate system pipe break. A hotwell level channel will trip all three PCPs. At 100 percent power with a loss of condensate and feedwater, RPV will drop rapidly. HC.OP-AB.RPV-0004 Immediate operator action of locking the mode switch to shutdown is required.
All Primary Condensate Pumps trip; Trip all Secondary Condensate and Feedwater Pumps. Incorrect. Wrong action per AB.
'A' Primary Condensate Pump trips; Reduce Reactor power using the 'Stuff Sheet'. Incorrect. Wrong action per AB.
'A' Primary Condensate Pump trips; Verify Full Reactor Recirc and Feedwater pump runbacks. Incorrect. Wrong action per AB.

Reference Title

HC.OP-AB.RPV-0004

Learning Objectives

MNCONDE015	(R) Given initial conditions and the loss of one or more condensate pumps, explain the interlocks and automatic actuations associated with the runback and/or trip logic of the condensate, feedwater and reactor recirculation systems.
ABRPV4E003	(R) From memory, recall the Immediate Operator Actions for Reactor Level Control.

Material Required for Examination	None		
Question Source:	New	Question Modification Method:	
Question Source Comments:			

Given the following conditions:

- Plant power level is 90 percent.
- # 2B Feedwater Heater (FWHTR) water level is rising.

Which one of the following actions occur if the FWHTR level reaches the HI-HI level setpoint?

a. The Main Turbine trips.

b. # 2B FWHTR Bleeder Trip Valves trip closed.

c. # 1, DC, & 2B FWHTR Condensate inlet and outlet valves auto close.

d. # 1, DC, & 2B FWHTR Startup and Operating Vents auto close.

Answer	c	Exam Level	B	Cognitive Level	Memory	Facility:	Hope Creek	Exam Date:	06/17/2003

Tier:	Plant Systems	RO Group	2	SRO Group	3	256000K406
256000	Reactor Condensate System				Record Number	89

K4. Knowledge of REACTOR CONDENSATE SYSTEM design feature(s) and/or interlocks which provide for the following:

K4.06 Control of extraction steam 2.8 2.8

Explanation of Answer

Justification:
1, DC, & 2B FWHTR Condensate inlet valve auto closure. Correct. High level in the #1, #2 or Drain Cooler automatically close the condensate inlet valves AE-HV-1633 and outlet valves 1600.
Main turbine trip. Incorrect. High RPV level trips turbine
Bleeder trip valves trip closed. Incorrect. Neck heaters have internal piping within the condenser shell and do not have BTVs. Correct for other heaters except 1,2,&DC.
1, DC, & 2B FWHTR Startup and Operating Vents auto close. Incorrect. Vents auto open.

Reference Title

HC.OP-SO.AE-0001

Learning Objectives

FWHEATE005	(R) From memory, describe the effects of too high or too low of a feedwater heater water level.

Material Required for Examination: None

Question Source: New | Question Modification Method:

Question Source Comments:

Given the following conditions:

- Reactor power is 14 percent.
- Reactor pressure is 923 psig.
- 'A' Reactor Feed Pump is feeding the vessel in Manual control.
- Start Up Level Control Valve (SULCV) demand is 22 percent.

Which one of the following choices describes the approximate positions of the SULCV's LV-1754 and 1785?

SULCV 1754 is __________ percent open and SULCV 1785 is __________ percent open.

a. 22; 0

b. 0; 22

c. 100; 2

d. 2; 100

Answer	c	Exam Level	R	Cognitive Level	Comprehension	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Plant Systems	RO Group	1	SRO Group	2				259001A307
259001	Reactor Feedwater System							Record Number	90
A3.	Ability to monitor automatic operations of the REACTOR FEEDWATER SYSTEM including:								
A3.07	FWRV position								3.2 3.2

Explanation of Answer

100; 2 Correct. 1754 is the 3 inch globe valve. At 20 % demand, this valve is full open. 1785 is the 12 inch globe valve. It is open 2 %.
22; 0 Incorrect - 1754 is 100 percent open. 1785 is 2 percent open.
0; 22 Incorrect - 1754 is 100 percent open. 1785 is only 2 percent open..
2; 100 Incorrect. Reverse of actual positions.

Reference Title

Digital Feed Drawing 1H-AE-ECS-0128-03F

LP NOH01FWCONT-00

Learning Objectives

FWCONTE005	(R) Given a labeled diagram/drawing of, or access to the Feedwater Control System control/indication bezel (10C651B,C), IAW Feedwater Control System Lesson Plan: Explain the function of each indicator. Assess plant conditions that will cause the indicators to light or extinguish. Determine the effect of each control switch on the Feedwater Control System. Assess plant conditions or permissives required for the control switches to perform their intended functions.

Material Required for Examination	Figure of SULCV controls on 10C651; P&ID M-06		
Question Source:	New	Question Modification Method:	
Question Source Comments:			

Given the following conditions:

- The plant is operating at 16 percent power during a startup.
- An I&C Techinician has completed channel functional testing on RPV Level 8 instrumentation to the Digital Feedwater System.

The values recorded for the trip setpoints were as follows:

A - 52.3 inches
B - 54.9 inches
C - 55.7 inches

Based on this data, which one of the following actions are required?

a. Restore the affected channel(s) to operable status within 72 hours or be in at least STARTUP within the next 6 hours.

b. Restore the affected channel(s) to operable status within 7 days or be in at least STARTUP within the next 6 hours.

c. Place the affected channel(s) in the tripped condition AND restore one channel to operable status within 72 hours or be in at least STARTUP within the next 6 hours.

d. Place the affected channel(s) in the tripped condition AND restore one channel to operable status within 7 days or be in at least STARTUP within the next 6 hours.

Answer	d	Exam Level	S	Cognitive Level	Application	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Plant Systems	RO Group	1	SRO Group	1				259002G133
259002	Reactor Water Level Control System							Record Number	91

2.1 Conduct of Operations

2.1.33 Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications. 3.4 4.0

Explanation of Answer

Justification:
SRO 55.43(2) Facility operating limitations in the Technical Specifications and their bases.
Correct: Place the affected channel(s) in the tripped condition AND restore one channel to operable status within 7 days or be in at least STARTUP within the next 6 hours. Only C channel is above the TS Allowable value of 55.5". No action is required for B channel. TS 3.3.9 action a. and b. are applicable.
Incorrect: Restore the affected channel(s) to operable status within 72 hours or be in at least STARTUP within the next 6 hours. Action for 2 inoperable channels. Channel C must also be placed in trip condition.
Incorrect: Restore the affected channel(s) to operable status within 7 days or be in at least STARTUP within the next 6 hours. Channel C must also be placed in trip condition.
Incorrect: Place the affected channel(s) in the tripped condition AND restore one channel to operable status within 72 hours or be in at least STARTUP within the next 6 hours. Action for 2 inoperable channels.

Reference Title

HCTS 3.3.9.

Learning Objectives

FWCONTE018 (R) Given a scenario of applicable operating conditions and access to Technical Specifications.
a. Identify those sections which are applicable to the Feedwater Control System.

b. Evaluate Feedwater Control System Operability and determine required actions based upon system operability.
c. Explain the bases for those Technical Specifications sections associated with the Feedwater Control System. (SRO Only)

Material Required for Examination	Tech Specs section 3.3.9.

Question Source:	New	Question Modification Method:	
Question Source Comments:			

Given the following conditions:

- The plant is at 100% power.
- The "C" steam flow detector for the feedwater level control system fails low (its output indicates 0 lbm/hr steam flow).

SELECT the statement which describes the automatic plant response with NO operator action.

a. Reactor water level will remain the same. The feedwater level control system will shift to single element control.

b. Reactor water level will decrease and stabilize at a lower than normal value. The feedwater level control system will remain in three element control.

c. Reactor water level will decrease and stabilize at a lower than normal value. The feedwater level control system will shift to single element control.

d. Reactor water level will increase and stabilize at a higher than normal value. The feedwater level control system will remain in three element control.

Answer	a	Exam Level	B	Cognitive Level	Comprehension	Facility:	Hope Creek	Exam Date:	06/17/2003

Tier:	Plant Systems	RO Group	1	SRO Group	1	259002K307
259002	Reactor Water Level Control System			Record Number		92

K3. Knowledge of the effect that a loss or malfunction of the REACTOR WATER LEVEL CONTROL SYSTEM will have on following:

K3.07 Reactor water level indication 3.4 3.4

Explanation of Answer

JUSTIFICATION:
Reactor water level will remain the same. The feedwater level control system will shift to single element control.
Correct - the system transfers to Single Element control. With stable conditions single and three element should control at the same setpoint.

Reactor water level will decrease and stabilize at a lower than normal value. The feedwater level control system will remain in three element control.
Incorrect - level should not decrease; the system will shift to single element control.

Reactor water level will decrease and stabilize at a lower than normal value. The feedwater level control system will shift to single element control.
Incorrect - level should not decrease.

Reactor water level will increase and stabilize at a higher than normal value. The feedwater level control system will remain in three element control.
Incorrect - level should not increase; the system will shift to single element control.

Reference Title

Engineering Drawing H-1-AE-ECS-0128-0

Learning Objectives

FWCONTE013 (R) From memory, describe the response of the FWLC System if the total steam flow signal were to be lost, IAW the Feedwater Control System Lesson Plan.

Material Required for Examination	None		
Question Source:	Facility Exam Bank	**Question Modification Method:**	Editorially Modified
Question Source Comments:	Vision Bank QID #Q53792 modified		

Given the following conditions:

- The plant is operating at 100 percent power.
- AD483 Inverter output power is lost.

What effect does the loss have on the 'A' RFPT?

a. 'A' RFPT will NOT trip on RPV Level 8.

b. 'A' RFPT trips due to Loss of Control Oil pressure.

c. 'A' RFPT trips due to Loss of Speed Signal.

d. 'A' RFPT will NOT trip on Overspeed.

Answer	c	Exam Level	B	Cognitive Level	Memory	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Plant Systems			RO Group	2	SRO Group	2		262002K101
262002	Uninterruptable Power Supply (A.C./D.C.)							Record Number	93

K1. Knowledge of the physical connections and/or cause- effect relationships between UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) and the following:

K1.01 Feedwater level control: Plant-Specific 2.8 3.0

Explanation of Answer

Justification:
'A' RFPT trips due to Loss of Speed Signal. Correct. (<100 rpm with 60 sec time delay with the turbine control valves open.)
'A' RFPT trips due to Loss of Control Oil Pressure. Incorrect. Supplied from DC power source.
'A' RFPT will NOT trip on RPV Level 8. Incorrect. 2 of 3 channels remain which would satisfy logic to trip all 3 pumps.
'A' RFPT will NOT trip on Overspeed. Incorrect. Overspeed is mechcanical and will trip if needed.

Reference Title

HC.OP-AB.ZZ-0136

Learning Objectives

FWCONTE016 (R) Given any of the following systems, state the interrelationship between the FWLC System and that System, IAW the Feedwater Control System Lesson Plan:
a. 120 VAC Non-1E Electrical Distribution
b. 125 VDC Non-1E Electrical Distribution
c. Main Turbine
d. Recirculation System
e. Rod Worth Minimizer (RWM)
f. Main Steam
g. Redundant Reactivity Control (RRCS)
h. Hydrogen Water Chemical Injection System

Material Required for Examination None

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following conditions:

- The 1AD413 125 Volt Battery Charger is in service and providing a normal charge on its battery.
- The 1AD414 125 Volt Battery Charger is tagged for maintenance.
- While in this lineup, AC power to the charger is lost.
- The bus supplying the charger is reenergized after 20 minutes by its associated diesel generator.

Which of the following describes the response of this battery charger?

The 1AD413 Battery Charger will:

a. return to the "float" mode to recharge the battery.

b. trip and is interlocked "off " with the diesel generator powering the bus.

c. reset to the "equalize" mode to recharge the battery.

d. trip and must be manually restored as permitted by diesel generator loading.

Answer	a	Exam Level	B	Cognitive Level	Comprehension	Facility:	Hope Creek	Exam Date:	06/17/2003

Tier:	Plant Systems	RO Group	2	SRO Group	2	263000A101
263000	D.C. Electrical Distribution				Record Number	94

A1.	Ability to predict and/or monitor changes in parameters associated with operating the D.C. ELECTRICAL DISTRIBUTION controls including:		
A1.01	Battery charging/discharging rate	2.5	2.8

Explanation of Answer

Justification:
return to the "float" mode to recharge the battery. Correct. Although the charging rate will be higher than prior to the charger loss, the charger will remain in the Float mode.
trip and is interlocked "off " with the diesel generator powering the bus. Incorrect. The charger does not trip. The charger is restored when the bus power is restored.
reset to the "equalize" mode to recharge the battery. Incorrect. Equalize mode must be manually initiated using the timer control on the charger.
trip and must be manually restored as permitted by diesel generator loading. Incorrect. Returns when the AC bus is repowered.

Reference Title

HC.OP-SO.PK-0001

Learning Objectives

DCELECE010	Explain the purpose of and briefly describe how to initiate a given battery charger equalizing mode IAW the DC Electrical Distribution Lesson Plan.
DCELECE005	(R) Summarize the interrelationship(s) between 125VDC 1E/N1E Power Systems and the following IAW the DC Electrical Distribution Lesson Plan. a. 480VDC 1E/N1E Power Supply b. Auxiliary Building Ventilation System

Material Required for Examination	None		
Question Source:	INPO Exam Bank	Question Modification Method:	Editorially Modified
Question Source Comments:	INPO Bank QID # 14168 03/26/2001 Peach Bottom modified for Hope Creek.		

Given the following conditions:

- The unit tripped from 100% due to a loss of offsite power.
- HC.OP-AB.ZZ-0135 implementation is in progress.
- EDG B, C, & D are loaded onto their respective busses.
- EDG A is running.
- EDG A voltage is 3500 V.
- EDG A frequency is 56 Hz.
- EDG A output breaker is open.

Which one of the following actions must be taken to mitigate this event?

a. Raise speed and voltage from 10C651 panel to within limits and verify output breaker automatically closes.

b. Dispatch an operator to the EDG Remote Panel to raise speed and voltage to within limits and verify output breaker automatically closes.

c. Shutdown EDG A with the local/remote panel Emergency Shutdown pushbuttons.

d. Turn on the Synchroscope key and manually close EDG A output breaker from the Control Room.

Answer c | **Exam Level** B | **Cognitive Level** Comprehension | **Facility:** Hope Creek | **Exam Date:** 06/17/2003

Tier: Plant Systems | **RO Group** 1 | **SRO Group** 1 | 264000A304

264000 Emergency Generators (Diesel/Jet) | **Record Number** 95

A3. Ability to monitor automatic operations of the EMERGENCY GENERATORS (DIESEL/JET) including:

A3.04 Operation of the governor control system on frequency and voltage control 3.1 3.1

Explanation of Answer

Shutdown EDG A with the local/remote panel Emergency Shutdown pushbuttons. Correct. Voltage and frequency cannot be adjusted with a LOP signal present. Normal stop controls in the Control Room and local/remote panels are disabled. The only option is to secure the EDG with emergency stop PBs locally or leave it run unloaded. There is a time limitation for running unloaded of 4 hours.

Raise speed and voltage from 10C651 panel to within limits and verify output breaker automatically closes. Incorrect. Speed and voltage controls are disabled with LOP signal present.

Dispatch an operator to the EDG Remote Panel to raise speed and voltage to within limits and verify output breaker automatically closes. Incorrect. Speed and voltage controls are disabled with LOP signal present.

Turn on the Synchroscope key and manually close EDG A output breaker from the Control Room. Incorrect. Turning on sync scope key enable the snyc check monitor. With no infeed voltage to match, the breaker will not close.

Reference Title
HC.OP-SO.KJ-0001
LP NOH01EDG000-00

Learning Objectives	
EDG000E011	(R) Given plant conditions, predict the response of Diesel Generator governor and voltage regulator control circuitry to an Emergency start (LOP/LOCA).
EDG000E017	(R) Given plant conditions, predict the response of the Diesel Generator to an emergency start signal if: a. A normal shutdown is in progress. b. An emergency shutdown is in progress.

Material Required for Examination	None

Question Source:	INPO Exam Bank	Question Modification Method:	Significantly Modified

Question Source Comments:	INPO BANK QID # 13832 08/29/1999 Palo Verde. Modified for Hope Creek.

Given the following conditions:

- The Starting Air Compressor 1DK402 is Safety tagged for maintenance.
- The 'D' EDG Starting Air Receivers are crosstied to the 'B' EDG. (See attached figure)
- While pressurizing the 'D' EDG receivers, the crosstie air hose splits open.

Assuming NO operator action, which one of the following describes the effect on the associated EDGs to respond to a LOP?

a. Only B EDG will respond.

b. Only D EDG will respond.

c. B and D EDGs will respond.

d. Neither B NOR D EDG will respond.

Answer	c	Exam Level	B	Cognitive Level	Comprehension	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Plant Systems	RO Group	1	SRO Group	1				264000K601
264000	Emergency Generators (Diesel/Jet)							Record Number	96

K6. Knowledge of the effect that a loss or malfunction of the following will have on the EMERGENCY GENERATORS (DIESEL/JET):

K6.01 Starting air 3.8 3.9

Explanation of Answer

Justification:
B and D EDGs will respond. Correct. Check valves in the line maintain receiver pressure to allow both diesels to start on the LOP
Only B EDG will respond. Incorrect. Both diesels will start.
Only D EDG will respond. Incorrect. Both diesels will start.
Neither B NOR D EDG will respond. Incorrect. Both diesels will start.

Reference Title

HC.OP-SO.KJ-0001

Learning Objectives

EDG000E017 (R) Given plant conditions, predict the response of the Diesel Generator to an emergency start signal if:
a. A normal shutdown is in progress.
b. An emergency shutdown is in progress.

Material Required for Examination: Figure from Section 5.11 of HC.OP-SO.KJ-0001

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following conditions:

- The reactor core has been operating with one or more known fuel pin leaks.
- A reactor scram occurred from 100 percent power.
- Both Scram Discharge Volume Drain Valves did NOT go full closed.

Which one of the following rooms would become the most significant radiological hazard?

a. Reactor Building North Equipment Sump Room.

b. HPCI Pump and Turbine Room.

c. Reactor Building South Equipment Sump Room.

d. RCIC Pump and Turbine Room.

Answer	c	Exam Level	S	Cognitive Level	Comprehension	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Plant Systems			RO Group	3	SRO Group	3		268000G404
268000	Radwaste							Record Number	97

2.4 Emergency Procedures and Plan

2.4.4 Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures. 4.0 4.3

Explanation of Answer

Justification:
SRO 55.43(4) Radiation hazards that may arise during normal and abnormal situations.
Correct: Reactor Building South Equipment Sump Room. The North and South Scram discharge volumes drain through a common line to the Reactor Building Equipment Drain Sump 1BT266 located in the South Reactor Building Sump Room on 54' elevation. If the drain valves did not close as stated in the stem, a LOCA would exist discharging into this room. The leaking fuel would severely raise radiation levels in that room as well. Entry into 103/4 for any room Rad monitor alarm. It is not limited to only rooms of Table 1 and 2.
Incorrect: Reactor Building North Equipment Sump Room. North SDV does not drain to this sump. Common misconception.
Incorrect: HPCI Pump and Turbine Room. Rad levels would increase slightly from steam line drains unless HPCI was placed I/S. Stem does not support HPCI operation.
Incorrect: RCIC Pump and Turbine Room. Rad levels would increase slightly from steam line drains unless RCIC was placed I/S. Stem does not support RCIC operation.

Reference Title

EOP-103/4

EOP Conversion documents

Learning Objectives

EOP103E002	Given a set of plant conditions, analyze and determine if entry conditions into HC.OP-EO.ZZ-0103/4 exists.

Material Required for Examination	None		
Question Source:	New	Question Modification Method:	
Question Source Comments:			

Given the following conditions:

- A discharge of the Equipment Drain Sample Tank is in progress to the river.
- The Liquid Radwaste Discharge Isolation Valve to the Cooling Tower Blowdown automatically closes.

Which one of the following conditions would cause this termination?
(Assume no operator action)

a. Liquid Radwaste Effluent radiation element fails low.

b. Cooling Tower Blowdown weir flow rate fails low.

c. Liquid Radwaste Effluent sample flow rate fails high.

d. Cooling Tower Blowdown RMS radiation element fails high.

Answer	b	**Exam Level**	R	**Cognitive Level**	Comprehension	**Facility:** Hope Creek	**Exam Date:** 06/17/2003
Tier:	Plant Systems	**RO Group**	2	**SRO Group**	2		272000K301
272000	Radiation Monitoring System					**Record Number**	98

K3.	Knowledge of the effect that a loss or malfunction of the RADIATION MONITORING System will have on following:	
K3.01	Station liquid effluent release monitoring	3.2 3.8

Explanation of Answer

JUSTIFICATION:
Cooling Tower Blowdown weir flow rate fails low. Correct. Of choices given, only Cooling Tower weir flow (Dilution Flow) low will cause a release isolation and termination.
Liquid Radwaste Effluent radiation element fails low. Incorrect. Cause alarms but not isolation.
Liquid Radwaste Effluent sample flow rate fails high. Incorrect. Cause alarms but not isolation.
Cooling Tower Blowdown RMS radiation element fails high. Incorrect. Cause alarms but not isolation.

Reference Title

HC.OP-AR.SP-0001 Attachment 5

Learning Objectives

RWOVERE005	(R) From memory list/identify the five conditions that will cause a liquid release to be automatically terminated.

Material Required for Examination: None

Question Source: Facility Exam Bank | **Question Modification Method:** Direct From Source

Question Source Comments: Vision Bank QID# Q68906

Given the following conditions:

- The plant is operating at 100 percent power during hot summer conditions.
- CRIDS page 105 indicates Reactor Building Backdraft Damper PD-9438C1 is closed.
- All room temperature points are reading NHI.
- NO other Backdraft Dampers are closed.

What impact will this closure have on the plant?

a. MSIV closure is imminent. A Reactor scram is necessary.

b. Reactor Water Clean Up will isolate.

c. FRVS must be placed in service to maintain room temperatures.

d. Associated room temps will rise. Reset the damper.

Answer	d	Exam Level	R	Cognitive Level	Application	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Plant Systems			RO Group	3	SRO Group	3		288000G450
288000	Plant Ventilation Systems							Record Number	99
2.4	Emergency Procedures and Plan								
2.4.50	Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.							3.3	3.3

Explanation of Answer

Justification:
Associated room temps will rise. Reset the damper. Correct. IAW Subequent action F2, re-open the backdraft damper. If a high temperature isolation does not exist.
MSIV closure is imminent. A Reactor scram is necessary. Incorrect. Misinterpretation of table of Attachment 2.
Reactor Water cleanup will isolate. Incorrect. Damper is one of several that feed into RWCU pipechase room.
FRVS must be placed in service to maintain room temperatures. Incorrect. FRVS uses the same ductwork. FRVS uses SACS instead of Chilled water. It also uses the same flowpath as RBVS.

Reference Title

HC.OP-AB.CONT-0003

Learning Objectives

Material Required for Examination: HC.OP-AB.CONT-0003 Attachment 2

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following conditions:

- "B" Channel Reactor Building Refuel Floor Exhaust Radiation monitor is in the trip condition for I&C surveillance testing.
- Power is lost to the "B" Channel Reactor Building Refuel Floor Exhaust Radiation monitor.

Which one of the following describes the plant response, if any?

a. Neither Reactor Building Ventilation Inboard and Outboard Dampers HD-9414A & B or HD-9370A & B close.

b. Reactor Building Ventilation Inboard Dampers HD-9414A and HD-9370A only close.

c. Reactor Building Ventilation Outboard Dampers HD-9414B and HD-9370B only close.

d. Both Reactor Building Ventilation Inboard and Outboard Dampers HD-9414A & B and HD-9370A & B close.

Answer	a	Exam Level	B	Cognitive Level	Memory	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Plant Systems			RO Group	2	SRO Group	1		290001A301
290001	Secondary Containment							Record Number	100

A3. Ability to monitor automatic operations of the SECONDARY CONTAINMENT including:

A3.01 Secondary containment isolation 3.9 4.0

Explanation of Answer

Justification:
Neither Reactor Building Ventilation Inboard and Outboard Dampers HD-9414A & B or HD-9370A & B close. Correct - Loss of power to the same channel that is tripped only results in 1/3 trip and no dampers change position.
Reactor Building Ventilation Inboard Dampers HD-9414A and HD-9370A only close. Incorrect. 2 of 3 logic.
Reactor Building Ventilation Outboard Dampers HD-9414B and HD-9370B only close. Incorrect. 2 of 3 logic.
Both Reactor Building Ventilation Inboard and Outboard Dampers HD-9414A & B and HD-9370A & B close. Incorrect. 2 of 3 logic.

Reference Title

HC.OP-AR.ZZ-0019(Q) Attachment A3

Learning Objectives

SECCONE008 (R) Given a list of plant conditions, select the four automatic signals which will shutdown and isolate normal Reactor Building Ventilation and start the Filtration Recirculation and Ventilation System (FRVS) IAW the Secondary Containment Lesson Plan.

Material Required for Examination None.

Question Source: Facility Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: Vision Bank QID# Q76880 Modified for different channel.

Given the following conditions:

- The plant is operating at 100 percent power.
- The Main Steam Tunnel (MST) Ventilation Barrier (Panel) 10S203 indicates open on the RM-11.

Which one of the following describes the operational impact?

a. Turbine Building Exhaust RMS levels will rise to alert levels.

b. Degraded cooling capability for the MST Coolers.

c. Main Steam Line RMS detectors will read non-conservative.

d. Loss of secondary containment integrity.

Answer	d	**Exam Level**	B	**Cognitive Level**	Memory	**Facility:**	Hope Creek	**Exam Date:**	06/17/2003
Tier:	Plant Systems			**RO Group**	2	**SRO Group**	1		290001K107
290001	Secondary Containment							**Record Number**	101

K1. Knowledge of the physical connections and/or cause- effect relationships between SECONDARY CONTAINMENT and the following:

K1.07 Turbine building ventilation (steam tunnel): Plant- Specific 3.0 3.1

Explanation of Answer

Justification:
Loss of secondary containment integrity. Correct. The MST Ventilation Barrier is part of secondary containment and are verifed in place using the RM-11 and surveilance HC.OP-ST.ZZ-0003
Degraded cooling capability for the MST Coolers. Incorrect. Plausible misconception.
Main Steam Line RMS detectors will read non-conservative. MSL RMS are ion chambers and are not affected by ventialtion changes.
Turbine Building Exhaust RMS levels will rise to alert levels. Incorrect. Opening a ventilation path to the Turbine Building wil not drive RMS values up unless there is a system breach in the MST.

Reference Title

HC.OP-ST.ZZ-0003

Learning Objectives

SECCONE002	From memory describe the components (structures) that make up the Secondary Containment IAW the Secondary Containment Lesson Plan.
SECCONE005	(R) From memory describe the purpose and locations of the steam vent to atmosphere blowout panels, and the differential pressure that will actuate them IAW the Secondary Containment Lesson Plan.

Material Required for Examination None

Question Source: New **Question Modification Method:**

Question Source Comments:

Following sustained steady state operation at 100% power for 103 hours, indicated reactor power drops to 97% without operator action. Recirculation flow and rod positions have NOT changed.

Which of the following is the explanation for this change in power?

a. Core shroud cracking has occurred.

b. Feedwater flow to the reactor has risen.

c. Steam quality exiting the steam dryers has been reduced.

d. An Electro-Hydraulic Control (EHC) system change has caused reactor pressure to rise.

Answer	a	**Exam Level**	R	**Cognitive Level**	Memory	**Facility:**	Hope Creek	**Exam Date:**	06/17/2003
Tier:	Plant Systems	**RO Group**	3	**SRO Group**	3				290002K303
290002	Reactor Vessel Internals							**Record Number**	102

K3. Knowledge of the effect that a loss or malfunction of the REACTOR VESSEL INTERNALS will have on following:

K3.03 Reactor power 3.3 3.4

Explanation of Answer

JUSTIFICATION:
Correct answer: "Core shroud cracking has occurred." BWR OE . Power drops due to steam bypassing the moisture seperator and heating the feedwater in the annulus area.
Feedwater flow to the reactor has risen. Incorrect. A rise in feedwater flow will lower the inlet enthalpy and temperature, increasing coolant density and lowering void fraction and thereby causing power to rise.
Steam quality exiting the steam dryers has been reduced. Incorrect. Lowered steam quality has no direct bearing on reactor power.
An Electro-Hydraulic Control (EHC) system change has caused reactor pressure to rise. Incorrect. A higher reactor pressure suppresses the void fraction, causing power to rise.

Reference Title

GE-NE-523-148-1093 DRF 137-0010 GE BWR Core Shroud Evaluation

BWROG Letter dated 7/13/94

NRC Generic letter dated 7/25/94

Learning Objectives

RXVESSE009 (R) Given plant problems/industry events associated with the Reactor Vessel and Internals:
a. Discuss the root cause of the plant problem/industry event IAW the plant/industry event.
b. Discuss the HCGS design and/or procedural guidelines that mitigate/reduce the likelihood of the problem/industry event at HCGS IAW the plant/ industry event.
c. Discuss the "lessons learned" from this problem/event IAW the plant/industry event.

Material Required for Examination None

Question Source: Facility Exam Bank **Question Modification Method:** Direct From Source

Question Source Comments: VISION BANK QID# Q56259

Given the following conditions:

- The Control Room Ventilation System "A" is operating normally.
- The "B" Train is in a normal, standby lineup.

SELECT the system flow response to a Control Room Ventilation High Radiation Isolation at the points marked on the attached Figure.

a. A-3000 scfm; B-1000 scfm; C-4000 scfm; D-0 scfm; E-18500 scfm

b. A-4000 scfm; B-0 scfm; C-0 scfm; D-4000 scfm; E-18500 scfm

c. A-0 scfm; B-4000 scfm; C-4000 scfm; D-0 scfm; E-14500 scfm

d. A-0 scfm; B-1000 scfm; C-4000 scfm; D-3000 scfm; E-14500 scfm

Answer	d	Exam Level	B	Cognitive Level	Comprehension	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Plant Systems	RO Group	2	SRO Group	2				290003A301
290003	Control Room HVAC							Record Number	103

A3. Ability to monitor automatic operations of the CONTROL ROOM HVAC including:

A3.01 Initiation/reconfiguration 3.3 3.5

Explanation of Answer

JUSTIFICATION:
In the OA Mode following a high Rad signal:
- The "A" flowpath isolates; The "B" flowpath opens supplying 1000 scfm to the CREF fan; The "C" flowpath is always 4000 scfm when the CREF fan is running; The "D" flowpath supplies 3000 scfm to the CFEF fans 4000 scfm total; The "E" flowpath combines 14,500 scfm with the CREF fans 4000 scfm for a total of 18,500 scfm through the CRS fan.
CORRECT - A-0 scfm, B-1000 scfm, C-4000 scfm, D-3000 scfm, E-14500 scfm.
INCORRECT - A-4000 scfm, B-0 scfm, C-0 scfm, D-4000 scfm, E-18500 scfm. "A" is never 4000. "B" & "C" are 0 during normal operation. "E" is 18,500 during normal operation.
INCORRECT - A-0 scfm, B-4000 scfm, C-4000 scfm, D-0 scfm, E-14500 scfm. "B" is either 0 or 1000. "D" is never 0.
INCORRECT - A-3000 scfm, B-1000 scfm, C-4000 scfm, D-0 scfm, E-18500 scfm. "A" is only 3000 during normal operation. "D" is never 0. "E" is only 18,500 during normal operation.

Reference Title

M-78 & M-89

Learning Objectives

CAVENTE003	Given a drawing of the Control Room Ventilation Supply System (Fig. 1), trace the air flowpath for the following conditions IAW the lesson plan: Normal operation Isolate: Outside Air Mode Isolate: Recirc Mode

Material Required for Examination Simplied Control Room HVAC figure

Question Source: Facility Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: VISION BANK QID# Q53946.

Given the following conditions:

- A loss of coolant accident has occurred.
- The Reactor Auxiliaries Cooling System (RACS) has been restored.

Which of the following describes the availability/response of the Emergency Instrument Air Compressor (EIAC) for these conditions should instrument air header pressure begin lowering?

a. The EIAC will automatically start on instrument air header pressure less than 85 psig if the LOCA signal is cleared.

b. The EIAC will NOT automatically start but can be started locally after relieving intercooler pressure.

c. The EIAC is NOT available until the LOCA signal is cleared, PCIS reset, and the 1E breaker is closed.

d. The EIAC is NOT available until the Non-1E breaker is closed and instrument air pressure is less than 85 psig.

Answer	c	Exam Level	B	Cognitive Level	Comprehension	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Plant Systems	RO Group	2	SRO Group	2				300000K501
300000	Instrument Air System (IAS)							Record Number	104

K5. Knowledge of the operational implications of the following concepts as they apply to the INSTRUMENT AIR SYSTEM:

K5.01 Air compressors 2.5 2.5

Explanation of Answer

CORRECT: The EIAC is not available until the LOCA signal is cleared, PCIS reset, and the 1E breaker is closed.
INCORRECT: The EIAC will automatically start on instrument air header pressure less than 85 psig if the LOCA signal is cleared. Breaker not reset
INCORRECT: The EIAC is not available until the Non-1E breaker is closed and instrument air pressure is less than 85 psig. 1E breaker that needs resseting
INCORRECT: The EIAC will not automatically start but may be started manually from the Control Room or locally. Not until 1E breaker is reset.

Reference Title

HC.OP-SO.KB-0001

Learning Objectives

INSAIRE015 (R) From memory, determine the response of the emergency instrument air compressor to the following conditions, IAW the Instrument Air System Lesson Plan:
a. Loss of Offsite Power (LOP)
b. Loss of Coolant Accident (LOCA)
c. Compressor intercooler pressure > 5 psig and a start signal received

Material Required for Examination: None

Question Source: Facility Exam Bank **Question Modification Method:** Direct From Source

Question Source Comments: Vision Bank QID# Q56532

Given the following conditions:

- The plant is operating at 100 percent power when a Loss of Offsite Power (LOP) occurs.
- B Emergency Diesel Generator (EDG) trips due to electrical fault.
- D SACS Pump trips on overload.
- All other equipment functions properly.

Which of the following actions is required?

a. Open SSW to RACS crosstie valves.

b. Attempt one restart of B EDG.

c. Emergency stop D EDG.

d. Start B SACS Pump.

Answer	c	Exam Level	R	Cognitive Level	Comprehension	Facility:	Hope Creek	Exam Date:	06/17/2003

Tier:	Plant Systems	RO Group	2	SRO Group	2	400000K301

400000 Component Cooling Water System (CCWS) — Record Number 105

K3. Knowledge of the effect that a loss or malfunction of the CCWS will have on the following:

K3.01 Loads cooled by CCWS — 2.9 3.3

Explanation of Answer

Emergency stop D EDG. Correct. D EDG is running without cooling. It must be emergency stopped because the LOP signal disables the normal stop controls.
Attempt one restart of B EDG. Incorrect. Incorrect application of AP-0109 provision for breaker re-closure. Conditions given require an inspection of the electrical equipment.
Open Service Water to RACS crosstie valves. Incorrect. SSW to RACS valves remain open on a LOP.
Start B SACS Pump. Incorrect. B SACS Pump does not have power due to the B EDG trip.

Reference Title
Hc.OP-SO.KJ-0001
HC.OP-AB.ZZ-0135

Learning Objectives	
STACS0E020	(R) Given plant conditions, determine the STACS response to a valid LOCA and/or LOP signal. IAW available control room references
EDG000E010	Given a system that is required for the support of Emergency Diesel Generator operation, explain the purpose (support function) of that system.

Material Required for Examination: None

Question Source: New — Question Modification Method:

Question Source Comments:

Given the following conditions:

Station Service Water (SSW) pump status:
- 'A' SSW pump I/S in AUTO.
- 'B' SSW pump I/S in AUTO.
- 'C' SSW pump O/S in AUTO.
- 'D' SSW pump O/S in AUTO.

Which one of the following will result in the automatic start of the 'D' SSW Pump?

a. 'A' SSW Loop low flow.

b. 'A' SSW Pump low flow.

c. 'B' SSW Loop low flow. * Both "c" and "d" are correct.

d. 'B' SSW Pump low flow.

* Answer	d	Exam Level	B	Cognitive Level	Memory	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Plant Systems	RO Group	2	SRO Group	2				400000K401
400000	Component Cooling Water System (CCWS)							Record Number	106
K4.	Knowledge of CCWS design feature(s) and or interlocks which provide for the following:								
K4.01	Automatic start of standby pump								3.4 3.9

Explanation of Answer

Justification:
'B' SSW Pump low flow. Correct. Low flow of the opposite pump (B) within the affected loop (B) starts the standby pump (D).
'B' SSW Loop low flow. Incorrect. Low pump flow.
'A' SSW Pump low flow. Incorrect. Low flow of the opposite pump within the affected loop starts the standby pump.
'A' SSW Loop low flow. Incorrect. Low pump flow of the opposite pump..

Reference Title

HC.OP-SO.EA-0001

Learning Objectives

SERWATE005	(R) Identify/describe the signals that auto start the Station Service Water System. IAW available control room references

Material Required for Examination	None		
Question Source:	New	Question Modification Method:	
Question Source Comments:			

Given the following conditions:

- The plant is at 80 percent power with a power ascension in progress.
- A Reactor Recirc Pump scoop tube is tripped.
- Local adjustment of Reactor Recirculation pump A speed is required.

Which of the following describes the MINIMUM requirements to perform this evolution?

a. Communications via page announcements to the operator on the scoop tube; the operator must be RO licensed.

b. Communications via radio the operator on the scoop tube; the operator must be RO licensed.

c. Communications via telephone to the operator on the scoop tube; the operator must be SRO licensed.

d. Communications via sound powered phone to the operator on the scoop tube; the operator must be SRO licensed.

Answer	b	Exam Level	B	Cognitive Level	Memory	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Generic Knowledge and Abilities			RO Group	1	SRO Group	1		294001G108
GENERIC								Record Number	107

2.1 Conduct of Operations

2.1.8 Ability to coordinate personnel activities outside the control room. 3.8 3.6

Explanation of Answer

Justification:
Communications via page announcements to the operator on the scoop tube; the operator must be RO licensed. Incorrect. One way page anouncement is permitted only durin emergencies.
Communications via radio to the operator on the scoop tube; the operator must be RO licensed. Correct. The radio allows 3 way communications from the scoop tube positioner. RO license is required since moving the scoop tube directly changes reactivity.
Communications via page to the operator on the scoop tube; the operator must be SRO licensed. Incorrect. Only RO license required.
Communications via radio to the operator on the scoop tube; the operator must be SRO licensed. Incorrect. Only RO license required.

Reference Title

NC.NA-AP.ZZ-0005

HC.OP-SO.BB-0002 3.1.6

Learning Objectives

ADMPROE020	From Memory Determine who is permitted (including conditions) to manipulate controls which directly or indirectly affect reactivity or power level, IAW NC.NA-AP.ZZ-0005, and HC.OP-AP.ZZ-0005
RECIRCE016	(R) Given procedure HC.OP-SO.BB-0002, Reactor Recirculation System Operation, explain the bases for listed precautions and limitations.

Material Required for Examination None

Question Source: INPO Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments: INPO Bank QID # 21193 Dresden 06/14/2002

Given the following conditions:

- The plant is operating at 29 percent power.
- Overhead Annunciator C5-C2 TCV FAST CLOSURE & MSV TRIP BYP is ILLUMINATED.

Then the Main Turbine Generator trips.

- All Turbine Bypass valves responded full open.
- Overhead Annunciator B3-E5 RPV PRESSURE HI is ILLUMINATED.
- RPV pressure stabilizes at 1030 psig.

Which one of the following correctly describes the time limit required by Tech Specs to clear the high pressure alarm?

a. 2 minutes.

b. 15 minutes.

c. 30 minutes.

d. One hour.

Answer	b	Exam Level	B	Cognitive Level	Memory	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Generic Knowledge and Abilities	RO Group	1	SRO Group	1				294001G111
GENERIC								Record Number	108

2.1	Conduct of Operations		
2.1.11	Knowledge of less than one hour technical specification action statements for systems.	3.0	3.8

Explanation of Answer

Justification:
15 minutes. Correct TS 3.4.6.2; The high pressure alarm comes in at the LCO limit of 1020 psig. LCO action time limit is 15 minutes.
2 minutes. Incorrect. LCO for Stuck open SRV.
30 minutes. Incorrect. Plausible but wrong.
One hour. Incorrect. Plausible but wrong.

Reference Title

HCGS Tech Spec 3.4.6.2

HC.OP-AB.RPV-0005 E.5

Learning Objectives

ABRPV5E007	(R) Explain the bases for Subsequent Actions and the information contained in the Discussion Bases Section of Reactor Pressure.

Material Required for Examination	None		
Question Source:	Facility Exam Bank	Question Modification Method:	Significantly Modified
Question Source Comments:	Vision Bank QID# Q68851 Significantly modified.		

Given the following conditions:

- HPCI is removed from standby to perform HC.OP.IS.BJ-0101 HPCI System Valves Inservice Test.
- Valve BJ-HV-F042 Suppression Pool Suction Valve stoke time is 2 seconds longer than the "TECH SPECS OR DESIGN LIMITS" value.
- Valve BJ-HV-F004 CST Suction Valve strokes satisfactory and was returned to open position.

Which one of the following actions is required?

a. Deactivate F004 open; HPCI remains operable.

b. Deactivate F004 open; declare HPCI inoperable.

c. Deactivate F042 closed; HPCI remains operable.

d. Deactivate F042 closed; declare HPCI inoperable.

Answer	d	Exam Level	S	Cognitive Level	Application	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Generic Knowledge and Abilities	RO Group	1	SRO Group	1				294001G112
GENERIC								Record Number	109

2.1	Conduct of Operations		
2.1.12	Ability to apply technical specifications for a system.	2.9	4.0

Explanation of Answer

Justification:
SRO 55.43 (2) Facility operating limitations in the Technical specifications and their bases.
Correct: Deactivate F042 closed; declare HPCI inoperable. Tech Spec 3.6.3 requires inoperable Primary containment Isolation Valves deactivated closed with 4 hours. HPCI operability requires suction source from the Suppression Pool, therefore HPCI is inoperable.
Incorrect: Deactivate F004 open; HPCI remains operable. F042 must be deactivated closed. HPCI operability requires suction source from the Suppression Pool, therefore HPCI is inoperable.
Incorrect: Deactivate F004 open; declare HPCI inoperable. F042 must be deactivated closed. Lining up HPCI to the CST will not meet operability requirements.
Incorrect: Deactivate F042 closed; HPCI remains operable. HPCI operability requires suction source from the Suppression Pool, therefore HPCI is inoperable.

Reference Title

HCTS 3.6.3 and 3.5.1.c.2

Learning Objectives

HPCI00E018	(R) Given plant conditions and access to Technical Specifications: Select those sections which are applicable to the HPCI System IAW HCGS technical specifications. Evaluate HPCI System operability and required actions based upon system operability IAW HCGS technical specifications. (SRO Only) Explain the bases for those technical specification items associated with the HPCI System IAW HCGS technical specifications. (SRO Only)

Material Required for Examination HCTS section 3.5.1 and 3.6.3

Question Source: New **Question Modification Method:**

Question Source Comments:

Select the statement that satisfies 10CFR50.46 Acceptance Criteria for ECCS.

a. Long - Term Cooling - after any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long lived radioactivity remaining in the core.

b. Peak Cladding Temperature -calculated maximum fuel element cladding temperature shall NOT exceed 2100°F.

c. Maximum Cladding Oxidation - calculated total oxidation of the cladding shall nowhere exceed 21% times the total cladding thickness before oxidation.

d. Maximum Hydrogen Generation - calculated total amount of H2 generated from the chemical reaction of the cladding with water or steam shall NOT exceed 17% times the hypothetical amount that would be generated if all the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

Answer	a	Exam Level	R	Cognitive Level	Memory	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Generic Knowledge and Abilities	RO Group	1	SRO Group	1				294001G128
GENERIC								Record Number	110

2.1	Conduct of Operations	
2.1.28	Knowledge of the purpose and function of major system components and controls.	3.2 3.3

Explanation of Answer

Justification
Incorrect- Peak Cladding Temperature shall not exceed 2200 degrees F
Incorrect- Maximum Cladding Oxidation shall nowhere exceed 17% times the total cladding thickness before oxidation.
Incorrect- Maximum Hydrogen Generation - shall not exceed 1% times the hypothetical amount that would be generated if all the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.
Correct- Long - Term Cooling -after any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long lived radioactivity remaining in the core.

Reference Title

10CFR50.46 Acceptance Criteria

Learning Objectives

INECCSE002	From memory, describe the 5 NRC ECCS acceptance criteria as they apply to the design of the Emergency Core Cooling Systems, IAW the Introduction to ECCS Student Handout.

Material Required for Examination None

Question Source: Facility Exam Bank **Question Modification Method:** Direct From Source

Question Source Comments: Vision Bank QID# Q56873

Given the following conditions:

- A reactor shutdown is in progress.
- Power is currently 20%.
- Hydrogen Water Chemistry Injection (HWCI) is out of service.
- Main Steam Line RMS Setpoints are set High.
- 2 Condensate Demineralizers are in service at 3000 gpm each.
- Plant chemistry parameters are as follows:
  - Condensate demin influent conductivity - 0.21 umho/cm
  - Condensate demin effluent conductivity - 0.08 umho/cm
  - Reactor Water Cleanup conductivity - 0.07 umho/cm
  - Reactor coolant sample conductivity - 0.07 umho/cm

Based on these conditions, which one of the following would cause these indications and what procedure actions must be taken?

a. Condensate Demineralizer channeling; remove one demineralizer from service.

b. Crud burst from removing HWCI from service; restore HWCI to service.

c. Main Condenser tube leak; isolate the affected condenser waterbox.

d. Reactor fuel pin cladding leak; continue power reduction at normal rate.

Answer	c	Exam Level	S	Cognitive Level	Application	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Generic Knowledge and Abilities			RO Group	1	SRO Group	1		294001G134
GENERIC								Record Number	111

2.1	Conduct of Operations		
2.1.34	Ability to maintain primary and secondary plant chemistry within allowable limits.	2.3	2.9

Explanation of Answer

JUSTIFICATION:
10CFR55.43 (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
Correct. Main Condenser tube leak; isolate the affected condenser waterbox. Conductivity into the Cond Demins is high. This is a symptom of a Condenser tube Leak. Required action would be to remove the waterbox IAW AB-RPV-0008.
Incorrect. Condensate Demineralizer channeling due to low flow; remove one demineralizer from service. Demineralizer outlet conductivity is normal. Would have low inlet and high outlet conductivity.
Incorrect. Crud burst from removing HWCI from service; restore HWCI to service. RWCU and Reactor coolant conductivity levels are normal.
Incorrect. Reactor fuel pin cladding leak; continue power reduction at normal rate. Power reduction at normal rate not permitted due to MSL RMS setpoints are set high. Indications are not cause for emergency power reduction.

Reference Title

HC.OP-AB.RPV-0007

HC.OP-IO.ZZ-0004

Tech Specs Table 3.3.2-1,

Learning Objectives

HWCI00E006	(R) Explain the plant operating restrictions when a power reduction event occurs that results in reactor power below 20% of rated thermal power without the required Main Steam Line Radiation Monitor setpoint change, IAW HC.OP-IO.ZZ-0004.

Material Required for Examination	None		
Question Source:	New	Question Modification Method:	
Question Source Comments:			

Given the following conditions:

- The plant is in Operational Condition 3.
- A new system engineer has requested that the B Core Spray Pump be started with the discharge valve throttled to 75% open to determine starting current.

The Operations Superintendent . . .

a. may allow the test if the STA or another SRO with an engineering degree concurs.

b. may conduct the evolution without restrictions.

c. must withhold conducting the test until a IPTE package has been approved.

d. must NOT allow the test under any conditions.

Answer	c	Exam Level	S	Cognitive Level	Memory	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Generic Knowledge and Abilities		RO Group	1	SRO Group	1			294001G207
GENERIC								Record Number	112

2.2	Equipment Control		
2.2.7	Knowledge of the process for conducting tests or experiments not described in the safety analysis report.	2.0	3.2

Explanation of Answer

Justification:
SRO 10CFR55.43 (3) Facility license procedures required to obtain authority for the design and operating changes in the facility.
must withhold conducting the test until a IPTE package has been approved. Correct. The evolution is an IPTE and requires a package with Test Engineer and Test Managers designated.
may conduct the evolution without restrictions. Incorrect. Requires IPTE package.
may allow the test if the STA or another SRO with an engineering degree concurs. Needs Test engineer and Test Manager approval.
must NOT allow the test under any conditions. Incorrect. May be performed if IPTE package approved.

Reference Title

NC.NA-AP.ZZ-0084

Learning Objectives

ADMPROE082	Given access to Control Room References Determine if an activity meets the criteria for an Infrequently Performed Test or Evolution. IAW NC.NA-AP.ZZ-0084.

Material Required for Examination	None

Question Source:	INPO Exam Bank	Question Modification Method:	Editorially Modified
Question Source Comments:	INPO Bank QID# 19840 Braidwood. Modified to meet Hope Creek Admin procedures.		

Given the following conditions:

- The plant experienced a Failure to Scram from a turbine trip at 98% power.
- The operator successfully initiated a manual scram approximately 4 seconds following the turbine trip.
- Post trip analysis revealed the following data for the event:
  - Peak Reactor Power: 116% Thermal
  - Peak Reactor Pressure: 1205 psig
  - Minimum Vessel Level: -151 inches (Fuel Zone A and B)
  - Most limiting MCPR: 1.09

Which one of the following states which Safety Limit was violated?

a. Thermal Power Low Pressure - Low Flow.

b. Thermal Power High Pressure - High Flow.

c. RPV Level Safety Limit.

d. RPV Pressure Safety Limit.

Answer	b	Exam Level	B	Cognitive Level	Comprehension	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Generic Knowledge and Abilities			RO Group	1	SRO Group	1		294001G222
GENERIC								Record Number	113

2.2 Equipment Control

2.2.22 Knowledge of limiting conditions for operations and safety limits. 3.4 4.1

Explanation of Answer

Justification:
Thermal Power High Pressure - High Flow Correct. Minimum MCPR is 1.10
Thermal Power Low Pressure - Low Flow Incorrect. Initial power pre transient was above 25 percent with pressure above 785psig.
RPV Level Safety Limit. Incorrect. Level maintained above -161
RPV Pressure Safety Limit. Incorrect. Pressure below 1325 psig.

Reference Title

HCGS Tech Specs Safety Limits

Learning Objectives

TECSPCE001	(R) State the four (4) Safety Limits in terms of conditions.

Material Required for Examination: None

Question Source: INPO Exam Bank

Question Modification Method: Editorially Modified

Question Source Comments: INPO BANK QID# 21118 09/17/2001 Browns Ferry modified for Hope Creek

The core has been off-loaded to the fuel pool. Per HC.RE-AP.ZZ-0049, Hope Creek Conduct of Fuel Handling, what is the MINIMUM permissible complement of personnel in the crew involved in fuel movement NOT involving core alterations?

a. Fuel Handling Operator
Radiation Protection Technician
Reactor Engineer, acting as spotter

b. Fuel Handling Operator
Refueling Bridge Operator as spotter
Radiation Protection Technician
Reactor Engineer

c. Fuel Handling Operator
Refueling Bridge Operator
SRO acting as spotter
Radiation Protection Technician

d. Fuel Handling Operator
Refueling Bridge Operator
Radiation Protection Technician
Reactor Engineer
Control Room Refuel Monitor

DELETED

Answer	a	Exam Level	B	Cognitive Level	Memory	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Generic Knowledge and Abilities	RO Group	1	SRO Group	1				294001G226
GENERIC								Record Number	114

2.2 Equipment Control

2.2.26 Knowledge of refueling administrative requirements. 2.5 3.7

Explanation of Answer

Justification : IAW Technical Specifications 1.7, and HC.RE-AP.ZZ-0049 sections 5.3.2.C.7.
The procedure stipulates that the minimum crew for non-core alteration fuel handling activities in the spent fuel pool includes the Fuel Handling Operator, Radiation Protection Technician, Reactor Engineer and Spotter. The Reactor Engineer may fulfill the duties of the spotter; hence the minimum permissible complement of people is three.

Reference Title

HC.RE-AP.ZZ-0049 sections 5.3.2.C.7.

Learning Objectives

ADMPROE073	From Memory State the minimum fuel handling crew requirement for non-core alteration non irradiated fuel handling. IAW NC.NA-AP.ZZ-0049.

Material Required for Examination None

Question Source: Facility Exam Bank **Question Modification Method:** Direct From Source

Question Source Comments: VISION QID# Q58936

Given the following conditions:

- The plant is in Operational Condition 4.
- The Reactor Head detensioning machine is being lowered inplace to detension the reactor head.

Which one of the following personnel must be notified prior to the beginning the detensioning process IAW HC.OP-IO.ZZ-0005 Cold Shutdown to Refueling?

a. Refueling Floor SRO.

b. Reactor Engineer.

c. Control Room Supervisor.

d. Refueling Outage Manager.

Answer	c	Exam Level	S	Cognitive Level	Memory	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Generic Knowledge and Abilities		RO Group	1	SRO Group	1			294001G227
GENERIC								Record Number	115

2.2	Equipment Control	
2.2.27	Knowledge of the refueling process.	2.6 3.5

Explanation of Answer

Justification:
SRO 55.43 (2) Facility operating limitations in the Technical specifications and their bases.
Correct. Control Room Supervisor. Required signoff for HC.OP-IO.ZZ-0005. Changes Operational Condition to OC 5.
Incorrect. Reactor Engineer. Not required.
Incorrect. Refueling Floor SRO. Not required.
Incorrect. Refueling Outage Manager. Not required.

Reference Title

HC.OP-IO.ZZ-0005 step 5.2.33

Learning Objectives

IOP005E007	(R) Assess plant conditions and determine if the requirements for entering OPERATIONAL CONDITION 5 - REFUELING have been met.

Material Required for Examination	None
Question Source:	INPO Exam Bank
Question Modification Method:	Editorially Modified
Question Source Comments:	INPO BANK QID #908 03/27/1998 Grand Gulf Modified for Hope Creek job titles.

Given the following conditions:

- The plant is in Operational Condition 5.
- There are 16 fuel bundles remaining in the Reactor vessel.

Which of the following evolutions would be considered a "Core Alteration" by Technical Specifications?

a. Transferring a control rod from the Reactor vessel to the Spent Fuel Pool.

b. Removing an LPRM string from the Reactor vessel.

c. Removing an IRM detector from undervessel.

d. Transfering a control rod blade guide from the Spent Fuel Pool to the Reactor vessel.

Answer	a	Exam Level	R	Cognitive Level	Memory	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Generic Knowledge and Abilities		RO Group	1	SRO Group	1			294001G227
GENERIC								Record Number	116

2.2 Equipment Control

2.2.27 Knowledge of the refueling process. 2.6 3.5

Explanation of Answer: Justification: TS Definitons 1.7 Core Alteration shall be the movement of any fuel, sources, or reactivity control components within the reactor vessel with fuel in the vessel. Movement of SRMs, IRMs, LPRMs, TIP, or special movable detectors (including undervessel replacement) are not considered to be core alterations.

Reference Title

HCGS Tech Specs 1.0 Definitions

Learning Objectives

TECSPCE005 Define or discuss the terms contained in Section 1.0 of Hope Creek Generating Station Technical Specifications.

Material Required for Examination: None

Question Source: INPO Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: INPO BANK QID# 21022 12/21/2001 Palisades Modified for Hope Creek

Given the following conditions:

- An initial startup is in progress.
- Threshold power is 11 KW/ft.
- A 20 MWe/hr ramp is established during the last rod adjustment.
- The 20 MWe/hr ramp continues until nodal power reaches 12.5 KW/Ft.
- Then PCIOMR preconditioning is begun by ramping power at 10 MWe/hr.

Which of the following is the result of these actions?

a. Fuel cladding failure is probable.

b. Fuel cladding stress is maintained within the vendor specifications.

c. The helium gas volume between the fuel pellets and fuel rod cladding will be larger than expected.

d. Fuel pellet densification will cause high fuel temperature.

Answer	a	Exam Level	R	Cognitive Level	Comprehension	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Generic Knowledge and Abilities	RO Group	1	SRO Group	1				294001G235
GENERIC								Record Number	117

2.2 Equipment Control

2.2.35 Knowledge of control rod programming. 2.5 2.9

Explanation of Answer

JUSTIFICATION:
Correct answer: "Fuel cladding failure is probable." PCIMOR Ramp rate of .11KW/Ft/Hr or approximately 10 Mwe/hr exceeded above Threshold power level of 11 KW/ft.

Fuel pellet densification will cause high fuel temperature. Incorrect. Fuel Densification is not of concern.
"The helium gas volume between the fuel pellets and fuel rod claddingwill be larger than expected." Incorrect. The helium gas volume will be reduced by normal power operations, violating PCIOMR rules will further reduce the gap between the fuel pellets and cladding.
"Fuel cladding stress is maintained within the vendor specifications." Incorrect. By violating PCIOMR rules, the fuel cladding will be stressed beyond vendor recommended limits.

Reference Title

LP NOH01RXFUEL

Learning Objectives

RXFUELE013 Explain the following:
a. The acronym and purpose of PCIOMR.
b. The definition of threshold power.
c. The definition of envelope power.

Material Required for Examination	None		
Question Source:	Facility Exam Bank	Question Modification Method:	Editorially Modified
Question Source Comments:	Vision Bank QID# Q57174		

Per NC.NA-AP.ZZ-0024, Radiation Protection Program, a 21 year old worker with 11 Rem Lifetime dose from the previous 3 years working at Hope Creek will have an administrative exposure control level of (1)________ mrem TEDE per year. This can be raised to a maximum of (2)_______ mrem TEDE by the Radiation Protection Manager.
(Assume NO delegation of authority)

a. (1) 2000
(2) 3000 * Correct answer changed to "a".

b. (1) 2000
(2) 4000

c. (1) 3000
(2) 4500

d. (1) 3000
(2) 4750

* Answer	b	Exam Level	B	Cognitive Level	Memory	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Generic Knowledge and Abilities	RO Group	1	SRO Group	1				294001G304

GENERIC — Record Number 118

2.3 Radiological Controls

2.3.4 Knowledge of radiation exposure limits and contamination control, including permissible levels in excess of those authorized. 2.5 3.1

Explanation of Answer

Justification:
(1) 2000 (2) 3000 Incorrect. RP Supervisor approves extension to 3000
(1) 2000 (2) 4000 Correct per NC.NA-AP.ZZ-0024 Attachment 1. Worker does not meet 5(N-17) threshold. 5(21-17) =20 Rem.
(1) 3000 (2) 4500 Incorrect. Automatic authorization during Emergency Plan implementation.
(1) 3000 (2) 4750 Incorrect. VP Operations approves extension to 4750.

Reference Title

NC.NA-AP.ZZ-0024 Attachment 1

Learning Objectives

ADMPROE059 Given a set of exposure conditions Identify the personnel responsible for approval of the following dose extension:
a. Yearly Dose Extension
b. Declared Pregnant Women Dose Extension
c. Lifetime Dose Extension IAW NC.NA-AP.ZZ-0024:

Material Required for Examination: None

Question Source: INPO Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: INPO Bank QID# 21202 06/14/2002 Dresden modified for Hope Creek.

Given the following conditions:

- The plant is operating at 100 percent power.
- A steam leak is present on a manual valve packing in Main Steam Tunnel room.
- The work will take approximately 30 minutes.
- The RWP for the area is NOT current.
- The general area dose rates are estimated at 1.5 R/hr.

Which one of the following is required to allow the maintenance work to be authorized in accordance with Technical Specifications?

a. One of the maintenance personnel is self monitor qualified.

b. All personnel involved in performing the work are volunteers and have been fully briefed on the hazards involved.

c. All work is documented in the Control Room OS/CRS Narrative log with the total dose received.

d. The job is provided with continuous Radiation Protection coverage.

Answer	d	Exam Level	S	Cognitive Level	Memory	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Generic Knowledge and Abilities	RO Group	1	SRO Group	1				294001G310
GENERIC								Record Number	119

2.3	Radiological Controls		
2.3.10	Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure.	2.9	3.3

Explanation of Answer

JUSTIFICATION:
SRO 55.43(4) Radiation hazards that may arise during normal and abnormal situations.
Correct: The job is provided with continuous Radiation Protection coverage. HCGS TS 6.12.1.c. allows use of continous RP coverage in place of RWP and specific dose rate info.
Incorrect: All personnel involved in performing the work are volunteers and have been fully briefed on the hazards involved. Requirement for Personnel Emergency Exposure Limit.
Incorrect: All work is documented in the Control Room OS/CRS Narrative log with the total dose received. Not required.
Incorrect: One of the maintenance personnel is self monitor qualified. Self monitor can not replace RP coverage.

Reference Title

HCGS TS 6.12.1.c.

Learning Objectives

ADMPROE056	From Memory Describe what the worker is acknowledging when signing a RWP prior to use. IAW NC.NA-AP.ZZ-0024, Radiation Protection Program

Material Required for Examination	None		
Question Source:	INPO Exam Bank	Question Modification Method:	Editorially Modified
Question Source Comments:	INPO BANK QID# 9070		

The moisture content of charcoal adsorber bed of the Gaseous Radwaste System (GRW) is rising.

Which of the following parameter changes will occur and what actions would mitigate the effects?

a. Rising GRW post-treatment radiation levels due to an increase in Krypton. Lower Cooler Condenser temperature.

b. Rising GRW post-treatment radiation levels due to an increase in Iodine. Raise Cooler Condenser temperature.

c. Lowering GRW charcoal adsorber bed temperature. Lower offgas Dilution flow.

d. Lowering GRW charcoal adsorber bed hydrogen concentration. Raise offgas Dilution flow.

Answer	a	Exam Level	B	Cognitive Level	Comprehension	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Generic Knowledge and Abilities	RO Group	1	SRO Group	1				294001G311
GENERIC								Record Number	120

2.3	Radiological Controls		
2.3.11	Ability to control radiation releases.	2.7	3.2

Explanation of Answer

JUSTIFICATION
Rising GRW post-treatment radiation levels due to an increase in Krypton. Correct, Water on charcoal reduces adsorption process -> rad levels increase.
Rising GRW post-treatment radiation levels due to an increase in Iodine. Incorrect, iodine is soluble and should remain in the main condenser.
Lowering GRW charcoal adsorber bed temperature. Incorrect, water makes bed temperature rise due to the decay of already captured radioactive gases.
Lowering GRW charcoal adsorber bed hydrogen concentration. Incorrect, adsorber bed does not adsorb hydrogen.

Reference Title

LP NOHO1GASRWO-01

Learning Objectives

GASRW0E008	(R) Explain/identify the effect of moisture in the process gas stream on the following components IAW available control room references: a. Recombiner b. Charcoal Beds

Material Required for Examination None

Question Source: Facility Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments: VISION BANK ID# Q53490

Plant conditions are as follows:

- Reactor Power is at 70%.
- Condenser Vacuum is 5.5" Hg absolute and degrading.

Which one of the following states immediate operator actions required?

a. Ensure turbine sealing steam pressure is normal.

b. Trip the Main Turbine when 350 Mwe is reached.

c. Place the standby SJAE in-service.

d. Reduce reactor power.

Answer	d	Exam Level	B	Cognitive Level	Memory	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Generic Knowledge and Abilities	RO Group	1	SRO Group	1				294001G404
GENERIC								Record Number	121

2.4 Emergency Procedures and Plan

2.4.4 Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures. 4.0 4.3

Explanation of Answer

Justification:
Reference:HC.OP-AB.BOP-0006 Main Condenser Vacuum Immediate Action
Condition: Degraded Main Condenser Vacuum
Action: Reduce Reactor Power as necessary to maintain Condenser vacuum < 5.0 " Hg Abs

Justification:
Reduce reactor power.-Correct- See HC.OP-AB.BOP-0006
Ensure turbine sealing steam pressure is normal.-Incorrect- subsequent action A.2
Trip the Main Turbine when 350 Mwe is reached. -Incorrect- Retainment Override states < 300 Mwe vice 350 Mwe
Place the standby SJAE in-service.-Incorrect- Subsequent action B

Modified from 29320 Closed Reference
Last used LOR 0006-05
Question Topic:Immediate Actions for Loss of Vacuum
KA: 295002K3.09 [3.2/3.2] LOK F, LOD 2
Material Required for Examination: None

Reference Title

HC.OP-AB.BOP-0006

Learning Objectives

ABBOP6E003	(R) From memory, recall the Immediate Operator Actions for Main Condenser Vacuum.

Material Required for Examination None

Question Source: Facility Exam Bank **Question Modification Method:** Direct From Source

Question Source Comments:

Select the definition of the term, “Minimum Alternate RPV Flooding Pressure”.

a. The lowest differential pressure between the RPV and the suppression chamber at which steam flow through the minimum number of SRVs required for Emergency Depressurization is sufficient to remove all decay heat from the core.

b. The lowest RPV pressure at which steam flow through open SRVs is sufficient to preclude any clad temperature from exceeding 1500°F even if the reactor core is not completely covered.

c. The lowest differential pressure between the RPV and the suppression chamber at which the least number of SRVs can be opened, and will remove all decay heat from the core at a pressure sufficiently low that the ECCS with the lowest head will be capable of making up the SRV steam flow.

d. The lowest RPV pressure if at which Emergency Depressurization is commenced, the covered portion of the reactor core will generate sufficient steam flow through the specified number of open SRVs to prevent any clad temperature in the uncovered part of the core from exceeding 1800 F.

Answer	b	Exam Level	B	Cognitive Level	Memory	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Generic Knowledge and Abilities	RO Group	1	SRO Group	1				294001G417
GENERIC								Record Number	122

2.4	Emergency Procedures and Plan		
2.4.17	Knowledge of EOP terms and definitions.	3.1	3.8

Explanation of Answer

JUSTIFICATION:
CORRECT - The lowest RPV pressure at which steam flow through open SRVs is sufficient to preclude any clad temperature from exceeding 1500°F even if the reactor core is not completely covered. HC.OP-EO.ZZ-LIMITS-CONV.

INCORRECT - The lowest differential pressure between the RPV and the suppression chamber at which steam flow through the minimum number of SRVs required for Emergency Depressurization is sufficient to remove all decay heat from the core. Definition for Minimum RPV Flodding Pressure.

INCORRECT - The lowest differential pressure between the RPV and the suppression chamber at which the least number of SRVs can be opened, and will remove all decay heat from the core at a pressure sufficiently low that the ECCS with the lowest head will be capable of making up the SRV steam flow. Definition for Minimum number of SRVs required for emergency de-pressurization.

INCORRECT - The lowest RPV pressure if at which Emergency Depressurization is commenced, the covered portion of the reactor core will generate sufficient steam flow through the specified number of open SRVs to prevent any clad temperature in the uncovered part of the core from exceeding 1800 F. Definition for Minimum Zero Injection RPV Water Level.

Reference Title
HC.OP-EO.ZZ-0206A
HC.OP-EO.ZZ-LIMITS-CONV Pg 27 of 60

Learning Objectives	
EOP206E004	Define the term "Minimum Alternate RPV Flooding Pressure".

Material Required for Examination	None

Question Source:	Facility Exam Bank	**Question Modification Method:**	Direct From Source
Question Source Comments:	VISION Bank QID# Q56150		

Given the following conditions:

- A reactor scram occurred due to a level transient where RPV level reached -60 inches.
- HPCI Aux Oil Pump failed to start for an unknown reason.
- RCIC automatically initiated and restored level.

Which one of the following describes the reporting requirement, if any, via ENS line?

a. No report.

b. One Hour report.

c. Four Hour report.

d. Eight Hour report.

Answer	c	Exam Level	S	Cognitive Level	Application	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Generic Knowledge and Abilities			RO Group	1	SRO Group	1		294001G430
GENERIC								Record Number	123

2.4 Emergency Procedures and Plan

2.4.30 Knowledge of which events related to system operations/status should be reported to outside agencies. 2.2 3.6

Explanation of Answer

Justification:
SRO 55.43 (1) Conditions and limitations in the facility license.
10CFR55.43 (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations..
Four Hour report. Correct R.A.L 11.3.1 HPCI should have actuated and injected to the vessel but did not. Also 4 hour report on scram.
No report. Incorrect. Would be correct if this was preplanned sequence or test or if HPCI was out for scheduled maintenance.
One Hour report. Incorrect. Would be correct if RPV Level Safety Limit reached or Emergency Classification of UE, Alert, SAE, or GE reached.
Eight Hour report. Would be correct if malfunctioning Aux Oil Pump was found prior to the event.

Reference Title

HCGS ECG RAL 11.3.1

Learning Objectives

Material Required for Examination	HCGS ECG Section 11 RALs
Question Source:	New
Question Modification Method:	
Question Source Comments:	

Given the following:

- A severe accident has occurred.
- You have declared a General Emergency at 0337 hrs for loss of all three fission product barriers.
- The weather conditions are as follows:
  - Clear skies
  - Ambient temp = 35 degrees F

Which of the following is the correct Protective Action Recomendation for the above conditions?

a. Evacuate all sectors 0 - 5 miles and Evacuate downwind sector +/- 1 sector 5 - 10 miles. Shelter all remaining sectors 5 - 10 miles.

b. Shelter all sectors.

c. Evacuate all sectors 0 - 5 miles and Shelter downwind sector +/- 1 sector 5 - 10 miles. Shelter all remaining sectors 5 - 10 miles.

d. Evacuate all sectors 0 - 5 miles.

Answer	a	Exam Level	S	Cognitive Level	Application	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Generic Knowledge and Abilities	RO Group	1	SRO Group	1				294001G444
GENERIC								Record Number	124

2.4 Emergency Procedures and Plan

2.4.44 Knowledge of emergency plan protective action recommendations. 2.1 4.0

Explanation of Answer

Justification:
SRO 55.43 (1) Conditions and limitations in the facility license.
SRO 55.43(4) Radiation hazards that may arise during normal and abnormal situations.
LP NEPECDTYSC rev 00 Obj 5.0
ECG Attachment 4 Appendix 1
Evacuate all sectors 0 - 5 miles and Evacuate downwind sector +/- 1 sector 5 - 10 miles. Shelter all remaining sectors 5 - 10 miles. Correct answer. Loss of all barriers = 10 pts. Question of Appendix 1 is answered yes . Weather conditions are not severe enough to warrant shelter instead of evacuation.
Shelter all sectors. Incorrect. Weather conditions are not severe enough to warrant shelter instead of evacuation.
Evacuate all sectors 0 - 5 miles and Shelter downwind sector +/- 1 sector 5 - 10 miles. Shelter all remaining sectors 5 - 10 miles. Incorrect. Evacuate downwind sectors.
Evacuate all sectors 0 - 5 miles. Incorrect. Would be correct if only 9 point GE.

Reference Title

ECG Attachment 4 Appendix 1

Learning Objectives

Material Required for Examination Appendix 1 from ECG Attachment 4

Question Source: Facility Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments: Vision Bank QID# Q56439

Following a reactor scram and loss of feedwater, the plant is being cooled down using HPCI in full flow recirculation. A review of the operating logs indicates that reactor pressure for the past two hours is as follows:

Time	Reactor Pressure (psig)
0000	950
0015	925
0030	900
0045	850
0100	700
0115	650
0130	550
0145	300
0200	250

Based on these conditions, the cool down rate is _______________ administrative limits and _______________ Technical Specification limits.

a. within; within.

b. within; outside.

c. outside; within.

d. outside; outside.

Answer	d	Exam Level	B	Cognitive Level	Application	Facility:	Hope Creek	Exam Date:	06/17/2003
Tier:	Generic Knowledge and Abilities	RO Group	1	SRO Group	1				294001G447
GENERIC								Record Number	125

2.4 Emergency Procedures and Plan

2.4.47 Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material. 3.4 3.7

Explanation of Answer

Justification:
outside; outside. Correct. Between 0045 and 0145, cooldown reached 105 degrees within a one hour period.
outside; within. Incorrect. Exceeds both TC and Admin limits.
within; outside. Incorrect. Exceeds both TC and Admin limits.
within; within. Incorrect. Exceeds both TC and Admin limits.

Reference Title

HC.OP-IO.ZZ-0004

HCGS TS 3.4.6.1

Learning Objectives

IOP004E005 (R) Analyze plant conditions and parameters to determine if plant operation is in accordance with the SHUTDOWN FROM RATED POWER TO COLD SHUTDOWN Integrated Operating Procedure, supporting System Operating Procedures and Technical Specifications.

Material Required for Examination HC.OP-IO.ZZ-0004 Attachment 4. Steam Table form HC.OP-IO.ZZ-0008

Question Source:	INPO Exam Bank	**Question Modification Method:**	Significantly Modified
Question Source Comments:	9557 12/18/1995 Hope Creek		

Given the following conditions:

- Suppression Pool Narrow Range Level instrument is removed from service for calibration.
- Wide Range Level instrument Channel C is reading 75 inches.
- Wide Range Level instrument Channel A is reading 73 inches.

Which one of the following describes the action required, if any, and bases for your answer?

a. NO action is required because the level would have been within limits at the time of removal.

b. NO action is required because RCIC Suction swap would occur on an actual low level.

c. Makeup to Suppression Pool level is required because the average level is below the limit.

d. Makeup to Suppression Pool level is required because level is outside allowable limits.

Answer: d | Exam Level: B | Cognitive Level: Memory | Facility: Hope Creek | Exam Date: 06/17/2003

Tier: Generic Knowledge and Abilities | RO Group: 1 | SRO Group: 1 | 294001G448

GENERIC | Record Number: 126

2.4 Emergency Procedures and Plan

2.4.48 Ability to interpret control room indications to verify the status and operation of system, and understand how operator action s and directives affect plant and system conditions. 3.5 3.8

Explanation of Answer

Justification:
Makeup to Suppression Pool level is required because level is outside allowable limits. Correct. IAW NC.NA-AP.ZZ-0005 states "Station technicians and operators shall believe instrument readings and treat them as accurate unless proven otherwise."
NO action is required because the level would have been within limits at the time of failure. Incorrect. Plausible misconception.
NO action is required because RCIC Suction swap would occur on an actual low level. Incorrect. RCIC does not swap on SP level.
Makeup to Suppression Pool level is required because the average level is below the limit. Incorrect. Levels are not averaged to obtain actual level.

Reference Title

NC.NA-AP.ZZ-0005 5.12.1

Learning Objectives

ADMPROE023	From memory Choose the correct operator response to instrument indications. IAW NC.NA-AP.ZZ-0005 and HC.OP-AP.ZZ-0005.

Material Required for Examination: None

Question Source: New | Question Modification Method:

Question Source Comments: