Question	1

Hope Creek RO Exam - Nov 2005

☑ RO
☐ SRO

Tier # 1 **Group #** 1

Importance 3.6

HC Obj: IOP006E006

295001	Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4	
AK1.03	Knowledge of the operational implications of the following concepts as they apply to the Partial or Complete Loss of Forced Core Flow Circulation	Thermal Limits :(CFR: 41.8 to 41.10 / 45.3)

Question

Given: Hope Creek was at 100% power when the "B" Recirc pump developed excessive vibration and needed to be tripped.

WHICH ONE of the following actions is REQUIRED to be taken in accordance with HC.OP-IO.ZZ-0006, Power Changes during Operation?

A MAPLHGR limit must be reduced and MCPR safety limit must be reduced.

B MAPLHGR limit must be reduced and MCPR limits must be raised.

C MAPLHGR limit must be raised and MCPR limits must be reduced.

D MAPLHGR limit must be raised and MCPR limits must be raised.

Answer B

References
HC.OP-AB.RPV-0003 (Q), Rev. 9, Recirculation System
HC.OP-IO.ZZ-0006,Rev. 33, Power Changes during Operation
DL-26, attachment 3v

Justification

References during Exam None

A. INCORRECT per IOP-6, step 5.3.7 MAPLHGR limit must be reduced and MCPR safety limit must be raised
B. CORRECT - see "A" Above
C. is INCORRECT per IOP-6, step 5.3.7 which states that the MCPR safety limit must be raised
D. is INCORRECT per IOP-6, step 5.3.7 MAPLHGR limit must be reduced
IOP006E006 - (R) Assess plant conditions and determine if the requirements for entering SINGLE LOOP have been met.

Question Source Mod ☑ **Memory Level** ☐ **Comprehension Level**

Question History:

SXD review - 7/21/05 - LOD 1.75 perhaps re-write to make more difficult, removed "initially" from in front of at 100% power in stem.
AF - 8/23 - NOT fair to ask subsequent actions questions from memory, feels everybody will jump on "C"
MB - 9/26 - SXD to resolve
SXD - OK
RJC - tie to lesson plan objection
MB - added a lesson plan tie in - IOP Lesson Plan - OBJ 6) Assess plant conditions and determine if the requirements for entering SINGLE LOOP have been met.
AF - still doesn't think they will know this from memory, -- Perhaps look for MCPR questions in bank --, maybe give them IOP-6
MB - 10/25 - Re-wrote question to only ask what happens to Thermal Limits on Trip of Recirc pump IAW IOP-6
MB - Minor edit (added a reference)

Question	2

Hope Creek RO Exam - Nov 2005

☑ RO
☐ SRO

Tier # 1 **Group #** 1

Importance 3.9

HC Obj:

295003 Partial or Complete Loss of AC / 6

AA2.05 Ability to determine and interpret the following as they apply to Partial or Complete Loss of AC — Whether a partial or complete loss of A.C. Power has occurred:(CFR: 41.10 /43.5/ 45.13)

Question

Given the following conditions:
- The plant is in Operational Condition 5 with the Electrical Distribution System aligned in the Normal lineup.
- An internal short on Transformer 1BX-501 causes a sudden pressure fault on the transformer.

Which one of the following describes the resulting availability of power for the Safe Shutdown Systems?

A Power is lost permanently to both 4.16KV switchgear 10A401 and 10A403.
13 KV breaker BS 1-2 stays closed.
B and D Diesel Generators start but their output breakers DO NOT CLOSE.

B Power is lost momentarily to both 4.16KV switchgear 10A402 and 10A404.
13 KV breaker BS 1-2 trips open.
Power is restored when the B and D Diesel generators output breakers close.

C Power to both 4.16KV switchgear 10A402 and 10A404 fast transfers to Transformer 1AX501.
13 KV breaker BS 1-2 trips open.
B and D diesel generators START but their output breakers DO NOT CLOSE.

D Power to both 4.16KV switchgear 10A402 and 10A404 fast transfers to Transformer 1AX501.
13 KV breaker BS 1-2 trips open.
B and D diesel generators DO NOT START.

Answer D

References Hope Creek Question Q76871 - Modified
Drawing E-0001 and 066-01: Class 1E AC Power Distribution
NOH01EAC00-02 - CLASS 1E AC POWER DISTRIBUTION, page 32 of 93

Justification

References during Exam Drawing E-0001

Justification:
Correct answer. 13 Kv Breakers BS 2-3 and BS 1-2 trip open. Bus section 2 is de-energized, Bus section 1 remains energized. The bus infeed breaker swap to the AX501 feed. The loads remain energized. Because one infeed is always available, the Diesels do NOT start.
A - INCORRECT - Power is NOT permanently lost to both 4.16KV switchgears. Power is restored when the bus infeed breaker swaps to the AX501 feed.
B - INCORRECT - Power is NOT restored from the B & D Diesel Generators
C - INCORRECT - The B & D Diesel Generators DO NOT START

Question Source Mod ☐ **Memory Level** ☑ **Comprehension Level**

Question History:

SXD Review 7/21/05 - OK
AF - 8/23 will do more research on - may have problem with "D"
SXD - OK
AF - Possible K/A mismatch, reworded question, SXD to look at
MB - 11/8 - removed BS-2-3 from distractor

Question 3

Hope Creek RO Exam - Nov 2005

☑ **RO**
☐ **SRO**

Tier # 1 **Group #** 1

Importance 2.6

HC Obj:

295004 Partial or Total Loss of DC Pwr / 6

AK3.01 Knowledge of the reasons for the following responses as they apply to Partial or Total Loss of DC Pwr — Load shedding Plant Specific:(CFR: 41.5/41.10 / 45.6 /45.13)

Question

With the plant at 100% power, the plant loses power to 125V DC Class 1E switchgear 10D410.

If the plant were to experience a LOCA, how will Load shedding and control of non-1E loads be affected:

Load shedding of Non-1E loads that get control power from 10D410 …

A will occur and these loads can be be operated from the Control Room (ie. Load shedding and control will NOT be affected)

B will occur, however, these loads can NOT be operated from the Control Room.

C will NOT occur, however, these loads can be operated from the Control Room.

D will NOT occur and these loads can NOT be operated from the Control Room.

Answer D

References
INPO Question 23597 (somewhat)
Hope Creek Lesson Plan NOH01EAC00-02, CLASS 1E AC POWER DISTRIBUTION p34 talks about load shedding of non-1E loads on a LOCA
NOH01DCELEC-00, DC ELECTRICAL DISTRIBUTION p.22 talks about 125V DC supplying breaker control power

Justification

References during Exam None

A. - INCORRECT - Load shedding will occur.
B. - INCORRECT - load will NOT auto trip on a Load Shed Signal and CANNOT be operated from the Control Room.
C. - INCORRECT - load will NOT auto trip on a Load Shed signal
D. - CORRECT - Load shedding will NOT occur and load CANNOT be operated from the Control Room

Question Source Mod ☑ **Memory Level** ☐ **Comprehension Level**

Question History:

SXD Review - 7/21 - Question Stem confusing -
7/27 - Rewrote Question Stem - re-submitted
8/2 JD - Weak Question, doesn't address K/A - K/A Q about DC load manual shedding to conserve battery life
8/3 - rewrote question again.
AF - 8/23 - word search make Not's all caps. Possible K/A mismatch.
MB - 9/26 - SXD to resolve
SXD - OK as is
AF - still thinks it’s a K/A mismatch
MB - OK

Question	4

- [x] RO
- [ ] SRO

Tier # 1 **Group #** 1

Importance 3

HC Obj:

295005 Main Turbine Generator Trip / 3

AG2.1.2 Knowledge of operator responsibilities during all modes of plant operation (CFR: 41.10 / 45.13)

Question

Due to a main turbine vibration problem with a generator load of 110 MWe, a manual turbine trip is performed.

Which of the following describes the Maximum Time Limit permitted to open the generator Output Breakers for the given conditions in accordance with procedure HC.OP-SO.AC-0001, Main Turbine? (Assume they have NOT already tripped on reverse power.)

A 15 seconds after the turbine trip

B 45 seconds after the turbine trip

C 60 seconds after the turbine trip

D 90 seconds after the turbine trip

Answer A

References Hope Creek Question - Q53470
HC.OP-SO.AC-0001(Q) - Rev. 48, MAIN TURBINE OPERATION - P&L 3.1.15

Justification

References during Exam None

Correct Answer:·"15 seconds" -Procedure caution calls for operator actions within 15 seconds of the turbine trip at low power.
The following distractors are incorrect as follows:
"45 seconds" - Procedure caution calls for operator actions within 15 seconds
"60 seconds" - Procedure caution calls for operator actions within 15 seconds of the turbine trip at low power.
"90 seconds"-Procedure caution calls for operator actions within 15 seconds of the turbine trip at low power. Only when above 150 MWe is the time extended to 90 seconds.

Question Source Bank

- [x] Memory Level
- [ ] Comprehension Level

Question History:

SXD Review - 7/21 - Had question about lower power -
7/27 - verified power level ok per IOP-4 p.15
AF 8/23 - A and B essentially the same. Suggests making A longer than 15 seconds.
MB - 9/26 - SXD to resolve
SXD - Minor change to stem
MB -10/3 Made changes as requested
AF - OK
Val - obscure fact
MB - 11/8 - leave as is.
MB - 11/17 deleted immediately and replaced with 45 seconds

Question	5

Hope Creek RO Exam - Nov 2005

☑ RO
☐ SRO

Tier # 1 Group # 1

Importance 3.7

HC Obj: AB0000E004

295006 SCRAM / 1

AK1.03 Knowledge of the operational implications of the fallowing concepts as they apply to the SCRAM — Reactivity Control:(CFR: 41.8 to 41.10 /45.3)

Question

A reactor scram has just occurred and the crew is executing HC.OP-AB.ZZ-0000, REACTOR SCRAM.

Which of the following is the reason that step S-8 directs the operator to RESET the scram (SB) if conditions permit?

A To reduce the potential for CRD pump runout and reduce the amount of time for the HCU accumulators to recharge.

B To restore the CRD hydraulic system to normal for insert and withdrawal capability if rods are found at the 02 or beyond position.

C To reestablish the normal primary vessel boundaries by isolating the CRD HCU from the scram discharge volume (SDV) and closing the SDV vent and drain valves.

D To prevent excessive discharge of hot radioactive water to the Reactor Building Equipment Drain Sump.

Answer B

References Hope Creek Question - Q56128
NOH01AB0000-01, Reactor Scram AB-0000 p.14

Justification

References during Exam AB-0000 with entry conditions blacked out

Justification:
C - INCORRECT - To reestablish the normal primary vessel boundaries by isolating the CRD HCU from the scram discharge volume (SDV) and closing the SDV vent and drain valves. Incorrect – the Scram reset will open the vents and drains
B - CORRECT - To restore the CRD hydraulic system to normal for insert and withdrawal capability if rods are found at the 02 or beyond position. Correct.
A - INCORRECT -To reduce the potential for CRD pump runout and reduce the amount of time for the HCU accumulators to recharge. Incorrect – system flow restricting orifice limit pump runout to 200 gpm
D - INCORRECT - To prevent excessive discharge of hot radioactive water to the Reactor Building Equipment Drain Sump. Incorrect – resetting scram will send water to the Rx Bldg Equipment Drain Sump

Question Source Bank ☑ Memory Level ☐ Comprehension Level

Question History:

Submitted 7/22
SXD Reviewed 7/23 - for Distactor C - asked is this verified?
AF- 8/23 swapped A & C justification.
MB - Made changes as requested
AF - OK
MB - added reference to question (AB-0000), deleted portion of stem to "reset half scram"

Question	6

- [x] RO
- [ ] SRO

Tier # 1 **Group #** 1

Importance 3.9

HC Obj:

295016 Control Room Abandonment / 7

AG2.1.30 Ability to locate and operate components, including local controls. (CFR: 41.7 / 45.7)

Question

Remote Shutdown Panel Transfer Switch "B" has been placed in the EMERGENCY position.

Which of the following lists the SRVs that can be operated at the Remote Shutdown Panel (10C399) AND describes the status of their controls in the Control Room (CR)?

A A, B, C, D, & E.
CR controls still function normally.

B A, B, C, D, & E.
CR controls are disabled.

C F, H, & M.
CR controls still function normally.

D F, H & M.
CR controls are disabled.

Answer D

References Hope Creek Question - Q62205, HC.OP-IO.ZZ-0008, Section 5.1, Attachment #1, Step B.2.9

NOH01MSTEAMC-02, MAIN STEAM SYSTEM, Obj R3d

Justification

References during Exam None

D - CORRECT - F, H & M. CR controls are disabled. Only SRVs M, F & H can be controlled from the RSP and when the transfer switches are in EMERGENCY, the CR functions are disabled.

A - INCORRECT - A, B, C, D & E are the ADS valves NOT the valves that can be controlled from the RSP.
B - INCORRECT - A, B, C, D & E CANNOT be controlled from the RSP.
C - INCORRECT - CR controls are disabled when RSP transfer switch B has been placed in the EMERGENCY position.

Question Source Bank

- [x] Memory Level
- [ ] Comprehension Level

Question History:

SXD Review - 7/21 - LOD 1.75 evaluate Revising
AF - OK
MB - spacing on distractors 11/8

Question	7

Hope Creek RO Exam - Nov 2005

☑ RO
☐ SRO

Tier # 1 **Group #** 1

HC Obj: NOH01STACS0-02 - Obj 20

Importance 2.9

295018 Partial or Total Loss of CCW / 8

AA2.04 Ability to determine and interpret the following as they apply to Partial or Total Loss of CCW :(CFR: 41.10/43.5/ 45.13) System Flow

Question

Hope Creek is at 100% power with the following SACS lineup:

- "A" & "C" SACS pumps running supplying TACS and the "A" SACS loads.
- "B" SACS pump is in "Auto" and NOT running.
- "D" SACS pump running supplying the "B" SACS loads.

When a Small Break LOCA occurs outside the drywell causing a SCRAM. Reactor Water level drops to minus 50" as HPCI auto starts and recovers level.

Which of the following correctly describes the SACS lineup and how flow is being supplied to the TACS loads?

A "A" & "C" SACS pumps running supplying TACS and the "A" SACS loads
"D" SACS pumps running supplying the "B" SACS loads

B "A" & "C" SACS pumps running supplying TACS and the "A" SACS loads
"B" & "D" SACS pumps running supplying the "B" SACS loads

C "A" & "C" SACS pumps running supplying the "A" SACS loads
"D" SACS pump running supplying the "B" SACS loads
TACS loads are isolated.

D "A" & "C" SACS pumps running supplying the "A" SACS loads
"B" & "D" SACS pumps running supplying the "B" SACS loads
TACS loads are isolated.

Answer B

References Hope Creek Procedure HC.OP-AB.COOL-0002, SAFETY/TURBINE AUXILIARIES COOLING SYSTEM, p. 9-13
NOH01STACS0-02, SAFETY AND TURBINE AUXILIARY COOLING WATER SYSTEM, p.49-50

Justification

References during Exam None

A - INCORRECT - "B" SACS pump will AUTO START on a Level 2 LOCA
B - CORRECT - "B" SACS pump will AUTO Start on LOCA Level 2, TACS does NOT isolate
C - INCORRECT - TACS does NOT isolate on a Level 2 LOCA only on a Level 1 LOCA
D - INCORRECT - See "C"

Question Source New ☐ **Memory Level** ☑ **Comprehension Level**

Question History:

SXD Review 7/21 - Maybe SRO level question, maybe a direct lookup
7/27 - I don't think it's a direct lookup - Look up Lesson Plan Objective
AF - 8/23 normally operating with 3 SAC's pumps, NOT an RO Question, replace.
MB - 9/27 - re-wrote question
SXD - OK
AF - minor editorial change - added outside the drywell
MB - Incorporated changes
Val - OK
MB - 12/2 - added "B" SACS pump in AUTO in stem

Question 8

☑ **RO** ☐ **SRO**

Tier # 1 **Group #** 1

Importance 3

HC Obj:

295019 Partial or Total Loss of Inst. Air / 8

AA1.03 Ability to operate and/or monitor the following as they apply to Partial or Total Loss of Inst. Air — Instrument Air Compressor Power supplies:(CFR: 41.7145.5/45.6)

Question

Given the following conditions:

Hope Creek is starting up from a Refueling outage, the plant is currently in OPCON 3 with temperature at 240°F and with the Instrument Air pressure at 105 psig and the Instrument/Service Air Compressors are aligned as follows:

Compressor	Control Mode	Status
00K107	MAN	Running
10K107	MAN	OFF
10K100	AUTO	OFF

A Maintenance Worker accidentally bumps into 7.2KV Bus 10A120 causing it's input breaker to open and the bus to de-energize.

Assuming NO operator actions, which of the following correctly states the expected response of the Instrument/ Service Air systems?

A Service Air compressor 10K107 de-energizes, Instrument Air header pressure remains at 105 psig.

B Service Air compressor 00K107 de-energizes, Instrument Air header pressure drops to 92 psig, when Service Air Compressor 10K107 starts and returns pressure to ~95 psig.

C Service Air compressor 00K107 de-energizes, Instrument Air header pressure drops to 70 psig when Emergency Air Compressor 10K100 starts and returns pressure to ~105 psig.

D Service Air compressor 00K107 de-energizes, Instrument Air header pressure drops to 85 psig when Emergency Air Compressor 10K100 starts and returns pressure to ~95 psig.

Answer D

References NOH01SERAIR-01, SERVICE AIR SYSTEM, p.47-48
NOH01INSAIR-01, INSTRUMENT AIR SYSTEM, p15, 42

Justification

References during Exam None

A. INCORRECT - Power to SAC 10K107 is from 7.2 KV bus 10A110, NOT 10A120
B. INCORRECT - SAC 10K107 will NOT start at 92 psig because it's in MAN control.
C. INCORRECT - EIAC 10K100 will auto start at 85 psig, however, it unloads at 100 psig, thereby making in NOT capable of raise pressure to 105 psig.
D. CORRECT - Loss of Power to 10A120 causes a loss of Power to SAC 00K107, Instrument Air header pressure drops to 85 psig, when EIAC 10K100 starts and brings pressure back to some value < 100 psig.

Question Source New ☐ **Memory Level** ☑ **Comprehension Level**

Question History:

SXD reviewed 7/25 - minor editorial changes to stem and distractor B - changed 105 psig to 95 psig.
AF - 8/23 - 3 Instrument air questions - 1 question contradics this one. 105 psig isn't normal for instrument air. Changed to OK
AF - 10/13 minor change
MB - incorporated change
Val - want to give students print
SXD - can't give print, makes it a direct lookup and K/A tests power supply
MB - 11/17 - changed "C" from 85 psig to 70 psig

Question 9

Hope Creek RO Exam - Nov 2005

☑ **RO**
☐ **SRO**

Tier # 1 **Group #** 1

Importance 3.4

HC Obj:

295021 Loss of Shutdown Cooling / 4

AA2.05 Ability to determine and interpret the following as they apply to Loss of Shutdown Cooling — Reactor Vessel Metal Temperature (CFR: 41.10 /43.5/45.13)

Question

Given the following conditions and using the provided figure:

- The reactor has been shutdown for 90 hours following 1000 EFPD of operation.
- The plant is in Cold Shutdown with RPV metal and RCS temperature of 140°F.
- A total loss of Shutdown Cooling occurred at 1200 hours.
- All efforts to restore heat removal from the RPV have failed.
- Both Recirculation pumps have been secured.

Assuming NO additional operator action, when will the plant reach OPCON 3?

A 1245

B 1307

C 1330

D 1352

Answer B

References Hope Creek Question - Q61328, HC.OP-AB.RPV-0009, Figure 1and Technical Specification Table 1.2

Justification

References during Exam Figure 1 of HC.OP-AB.RPV-0009

Justification
·1307- correct- Operational Condition 3 is achieved when the Reactor temperature reaches 200°F. The 140°F curve of Figure 1 intersects the 90-hour line between the 1.000 and 1.250 hour lines. 1307 is the only option that is between 1 hour and 1 hour and fifteen minutes following the loss of SDC.
·1245. incorrect- Value obtained by using the 160°F curve.
·1330. -incorrect- Value obtained by using the 120°F curve.
·1352. -incorrect- Value obtained by using the 100°F curve.

Question Source Bank ☐ **Memory Level** ☑ **Comprehension Level**

Question History:

SXD Review 7/21 - OK
AF - 8/23 - K/A mismatch - make NO caps
SXD - put metal temp's in stem
MB - made changes as requested.
AF - OK
Val - add using "figure provided"
MB - made change as requested.
MB - 11/17 added and RCS temp

Question 10

Hope Creek RO Exam - Nov 2005

☑ **RO**
☐ **SRO**

Tier # 1 **Group #** 1

Importance 3.4

HC Obj:

295023 Refueling Acc / 8

AK2.03 Knowledge of the interrelations between Refueling Accidents and the following — Radiation Monitoring equipment (CFR41.7 /45.7/ 45.8)

Question

Given the following conditions:

- The plant is in a refueling outage with a fuel move in progress.
- The 'A' Refuel Floor Exhaust Radiation Monitor has failed to a background reading.
- NO actions have been taken to address this failure.

- At time 0000 a fuel bundle is dropped and radiation levels on the refuel floor start to slowly rise.
- At time 0005 the B Refuel Floor Exhaust Radiation Monitor reaches its Hi Trip Setpoint.
- At time 0010 the C Refuel Floor Exhaust Radiation Monitor reaches its Hi Trip Setpoint.

Under these conditions, an automatic trip of the Reactor Building Ventilation Exhaust (RBVE) fans due to Hi Refuel Floor Exhaust Radiation levels:

A will occur at time 0010.

B will NOT occur due to the 'A' Refuel Floor Exhaust Radiation Monitor being failed to a background reading.

C will occur at time 0005.

D will NOT occur until at least 1 Reactor Building Exhaust radiation monitor senses high radiation.

Answer A

References
INPO Question 25978
NOH04000221C-01, RADIATION MONITORING SYSTEM p. 29
HC.OP-SO.SM-0001

Justification

References during Exam None

A. _ CORRECT - Per lesson plan p.29 item g. Automatic actions on a Refuel Floor Exhaust RM-23A HIGH radiation intensity level (any two of the three) - RBVE fans trip.
B - INCORRECT - still have 2/3 monitors available
C - INCORRECT - since A channel is failed downscale, need 2/3 to get actuation. Therefore won't get actuation when B channel gets high signal.
D - INCORRECT - will get a trip of RBVE fans on either Hi Refuel Floor Rad levels and RBVE rad levels.

Question Source Mod ☑ **Memory Level** ☐ **Comprehension Level**

Question History:

SXD Review 7/21 - Changed Distractor D to make it clearer
AF - 8/23 - all caps NO in 2nd bullet, minor editorial changes
MB - 8/ 24 Made changes as requested
AF - OK
Val - answer incorrect - obscure fact only known by I&C
MB - changed stem to fails to a background reading
MB - 11/17 added Exhaust
MB - 12/2 - "B" changed to failed to a background reading.

Question	11

Hope Creek RO Exam - Nov 2005

☑ RO
☐ SRO

Tier # 1 Group # 1

Importance 4

HC Obj:

295024 High Drywell Pressure / 5

EA1.03 Ability to operate and/ or monitor the following as they apply to High Drywell Pressure LPCS

Question

The A Core Spray pump is in full flow test mode in accordance with HC-OP.IS.BE-0001, Core Spray Pumps A and C Inservice Test. A steam leak in the drywell has caused the following conditions:

- Reactor scrammed and all rods inserted.
- RPV level lowered to -60 inches and is now rising with HPCI.
- Drywell pressure is 3.0 psig rising.
- RPV pressure is 800 psig lowering.
- Offsite power remains available to the 4KV buses.

Based on the above conditions, which one of the following is the correct response of the Core Spray system?

A "A" Core Spray pump continues to run in full flow test, all others are operating in min flow.

B ALL Core Spray pumps are operating on min flow.

C ALL Core Spray pumps are tripped and ALL pumps will start when RPV pressure lowers to 461 psig.

D ALL Core Spray pumps are injecting.

Answer B

References INPO Question 24762
NOH01CSSYS0-01, CORE SPRAY SYSTEM

Justification

References during Exam None

A - INCORRECT - Core spray full flow test valve closes upon Receipt of a CSS initiation signal.

B - CORRECT - Core Spray received a start signal at DW pressure > 1.68 psig. This caused all Core Spray pumps to start, however, RPV pressure is > 461 psig so upstream injection valves are closed and pumps are operating on their mini-flow valves. Core Spray test valve auto closed upon receipt of a CSS initiation signal.

C - INCORRECT - Core Spray pumps receive a start signal with pressure > 1.68 psig.

D - INCORRECT - Core Spray pumps upstream injection valves don't open until RPV pressure is < 461 psig.

"initiation" pump start signal is reached., A Core Spray running, NO trip signal to any CS pumps and NO loss of power., NO Core Spray "initiation" pump start signal is reached., Correct, > 2 psig signal closes full flow test valve

Question Source Mod ☐ Memory Level ☑ Comprehension Level

Question History:

SXD review 7/21 - OK
8/2 JD - Minor editorial change to "A" distractor - Incorporated
AF - 8/23 bulletize laundry list
MB - 8/ 24 Made changes as requested
AF - Minor changes
MB - Incorporated changes
Val - OK

Question 12

Hope Creek RO Exam - Nov 2005

☑ **RO**
☐ **SRO**

Tier # 1 **Group #** 1

Importance 3.8

HC Obj: EHCLOGE002

295025 High Reactor Pressure / 3

EA1.02 Ability to operate and / or monitor the following as they apply to High Reactor Pressure — Reactor/Turbine pressure regulating system :(CFR: 41.7/45.5/ 45.6)

Question

Hope Creek was operating at 75% power when a loss of feedwater heating occurs.

Assuming NO operator action, which of the following describes the effect on Reactor pressure and Main Turbine Pressure regulating system response:

Reactor Pressure will ___I___, which will cause the Main Turbine Pressure regulating system to send a signal to the Control valves to ____II____.

A I. Increase
II. Close

B I. Increase
II. Open

C I. Decrease
II. Close

D I. Decrease
II. Open

Answer B

References NOH01EHCLOG-02, EHC CONTROL LOGIC, p.8

Justification

References during Exam None

A - INCORRECT - when a Loss of FW heating occurs, colder FW will be sent to the reactor. This colder feedwater will cause a collapse in voids and a decrease in moderation, causing Reactor Power to increase. As reactor power increases, Reactor Pressure will increase. The increase in Reactor Pressure will cause the EHC system to OPEN the control valves to stabilize Reactor pressure.
B - CORRECT
C - INCORRECT - see "A" above
D - INCORRECT - see "A"above

Question Source New ☐ **Memory Level** ☑ **Comprehension Level**

Question History:

New Question 9/7
SXD - OK
AF - Possible K/A mismatch, doesn't address K/A abnormal, SXD to resolve
SXD - Leave as is.
Val - close to Q58
SXD - Not asking same thing, leave as is.

Question 13

Hope Creek RO Exam - Nov 2005

☑ **RO**
☐ **SRO**

Tier # 1 **Group #** 1

Importance 3.9

HC Obj: EOP102E009

295026 Suppression Pool High Water Temp. / 5

EG2.1.23 Ability to perform specific system and integrated plant procedures during all modes of plant operation. (CFR: 45.2 / 45.6)

Question

Given the following conditions:

- An ATWS is in progress
- APRM's read 10%
- Manual rod insertion is in progress
- MSIV's are closed
- Pressure is being maintained at 850 psig using SRV's
- Suppression Pool temperature is 195°F and rising at 1°F/5 min.
- Suppression Pool level is 70" and lowering at 1"/20 min.
- Suppression Pool pressure is 22 psig and rising at 1 psi/15 min.

Based on the conditions above, which of the following describes the initial action and the reason for that action?

A Reduce RPV pressure to prevent exceeding the PSPL.

B Emergency Depressurize to prevent exceeding the HCTL.

C Emergency Depressurize to prevent exceeding the PSPL.

D Reduce RPV pressure to prevent exceeding the HCTL.

Answer D

References
Hope Creek Question -Q62056
HC.OP-EO.ZZ-101, Reactor Pressure Vessel Control
HC.OP-EO.ZZ-010
BWR Owners Group EPGs/SAG Appendix B - Section 5 - Cautions

Justification

References during Exam EOP-102 without entry conditions

A - INCORRECT - RPV pressure reduction will NOT affect need to ED based on PSPL.
B - INCORRECT - With RPV pressure at 850 psig and SP temperature at 195 °F and rising at 5°F/min, the HCTL will be exceeded in 35 min. IAW Step SP/T-9, a pressure reduction prior to an ED is warranted. ED is NOT required yet.
C - INCORRECT -SP level is at 22" and rising at 1"/15 min. and SP level is at 70 " and lowering at 1"/20 min. Since at these rates of change, it will be about 1 hour before the PSPL is exceeded, an ED is NOT yet appropriate.
D - CORRECT - With RPV pressure at 850 psig and SP temperature at 195 °F, rising at 5°F/min, the HCTL will be exceeded in 35 min. and with power still at 10%, IAW Step SP/T-9 a pressure reduction is appropriate.

Question Source Bank ☐ **Memory Level** ☑ **Comprehension Level**

Question History:

SXD review 7/21 - OK
AF 8/23 - removed comment, couple of NPSH questions, need EOP Caution 2
MB - 8/ 24 Made changes as requested
MB - 9/27 - after looking at NPSH questions decided this question asked essentially the same information as Q15, changed out this question.]
SXD - OK
AF - added References to EOP-102, need to look at graph to get answer.
MB - added references
Val - Hard question
MB - changed distractor A
MB - 11/17 - changed appropriate to initial

Question	14

Hope Creek RO Exam - Nov 2005

☑ RO
☐ SRO

Tier # 1 Group # 1

Importance 3.9

HC Obj:

295028 High Drywell Temperature / 5

EG2.1.30 Ability to locate and operate components, including local controls. (CFR: 41.7 / 45.7)

Question

Given the following conditions:

- A Large Break LOCA has occurred in the Drywell.
- "B" RHR has been aligned for Drywell Spray.
- Subsequently, the Control Room needs to be evacuated and the Remote Shutdown Panel (RSP) manned.
- After taking local control at the RSP, ALL Transfer switches are placed in EMER.

What is the EXPECTED status of Drywell Spray and what Control/Indication does the Operator have over "B" Drywell Spray valves at the RSP?

I. ___________

II. ___________

A I. Drywell spray continues.
II. Operator has both Control and Indication over the "B" Drywell Spray valves at the RSP.

B I. Drywell spray continues.
II. Operator has only Indication over the "B" Drywell Spray valves at the RSP.

C I. Drywell spray is terminated.
II. Operator has both Control and Indication over the "B" Drywell Spray valves at the RSP.

D I. Drywell spray is terminated.
II. Operator has only Indication over the "B" Drywell Spray valves at the RSP.

Answer D

References
Hope Creek Question - Q56161, EOP- Caution 1, LP 0302-000.00H-00134-13 Obj 8
HC.OP-IO.ZZ-0008, p. 8
NOH01REMS/D-01, P.19-21
HC.OP-EO.ZZ-0101, RPV Control

Justification

References during Exam None

A- INCORRECT - Drywell spray is terminated when control is transferred to the RSP
B - INCORRECT - Drywell spray is terminated when control is transferred to the RSP
C - INCORRECT - Operator has ONLY indication over the "B" Drywell Spray valves at the RSP
D - CORRECT - Drywell spray is terminated and the operator only has indication of the "B" Drywell spray valves at the RSP.

Question Source New ☐ Memory Level ☑ Comprehension Level

Question History:

SXD review 7/ 21 - OK
JD 8/2 - K/A - Locate & Operate - asked to write question to J. Munro about Locate & Operate question.
AF - 8/23 weak K/A mismatch - NOT
MB - SXD to resolve
SXD - use pump and valve numbers in distractors, perhaps re-sample
MB - 10/3 - Changed question stem to ask for location of equipment in addition to reason for operating.
SXD - OK
AF - added HV-F021A, possible K/A mismatch
MB - added HV-F021A, SXD to resolve K/A
MB - 11/01 - re-wrote question to address AF concerns
MB - Ok by Val

Question 15

☑ **RO**
☐ **SRO**

Tier # 1 **Group #** 1

Importance 3.5

HC Obj:

295030 Low Suppression Pool Wtr Lvl / 5

EK3.07 Knowledge of the reasons for the following responses as they apply to Low Suppression Pool Wtr Lvl — NPSH considerations for ECCS pumps:(CFR: 41.5/41.10/45.6/ 45.13)

Question

The plant has experienced a transient and the following is observed:

- Suppression Chamber pressure: 10 psig
- Suppression Pool temperature: 240 degrees F
- Suppression Pool level at 0"
- Reactor pressure: 100 psig
- RHR "A" pump flow: 10,000 gpm
- Core Spray "B" pump Flow: 1500 gpm
- All other low pressure ECCS pump are NOT in service.

Use the attached curves to determine if Net Positive Suction Head (NPSH) requirements are being met.

A There is sufficient NPSH for the "B" Core Spray Pump ONLY.

B There is sufficient NPSH for the "A" RHR pump ONLY.

C There is sufficient NPSH for both the "A" RHR pump and the "B" Core Spray Pump.

D There is NOT sufficient NPSH for any pump.

Answer A

References INPO Question 14383
EOP CAUTION 2

Justification

References during Exam EOP Caution 2

Using EOP Caution 2 and realizing that being above the curve is the area of Unacceptable operation:
The limiting temperature for CS pump at 5 psig and 1500 gpm = 232°F
The limiting temperature for CS pump at 10 psig and 1500 gpm = 244°F
Interpolating for 9 psig gives a Temperature limit of ~242°F for 9 psig. Since given temperature = 240°F this puts the B CS pump in the area of ACCEPTABLE operation.
The limiting temperature for RHR pump at 10 psig is 235°F, since Given temperature is 240°F this puts the pump in the region of UNACCEPTABLE operation.

This makes ONLY Answer A CORRECT.

Question Source Mod ☐ **Memory Level** ☑ **Comprehension Level**

Question History:

SXD reviewed 7/22 - OK
AF - 8/23 another NPSH question, 74.5" drives them to AB.155, 2 correct answers as written, changed "D" to any pump. Perhaps change question 13
MB - 8/ 24 Made changes as requested
AF - to check Caution
MB - edit stem to 100 psig vs. 1000 psig.
MB - 12/2 change SP pressure to 10 psig vs. 9 psig.

Question	16

☑ RO
☐ SRO

Tier # 1 **Group #** 1

Importance 4

HC Obj:

295031 Reactor Low Water Level / 2

EK2.10 Knowledge of the interrelations between Reactor Low Water Level and the foliowing Redundant reactivity control

Question

Given the following:

- The plant is operating at 100% power.
- A transient results in a scram setpoint being exceeded.
- The Reactor Protection System fails to automatically scram the reactor.

Without operator action, which of the following describes how the Control Rods will be automatically inserted to shutdown the reactor via the ARI system?

A An RPV level of minus 50 (-50) inches will ENERGIZE the ARI valves to depressurize the scram air header.

B An RPV level of minus 50 (-50) inches will DE-ENERGIZE the ARI valves to depressurize the scram air header.

C An RPV pressure of 1050 psig will ENERGIZE the ARI valves to depressurize the scram air header.

D An RPV pressure of 1050 psig will DE-ENERGIZE the ARI valves to depressurize the scram air header.

Answer A

References
INPO Question 22776
NOH01RRCS00-00, REDUNDANT REACTIVITY CONTROL SYSTEM (RRCS), p.8

Justification

References during Exam None

A - CORRECT - with RPV level < -38" the ARI valves are energized to depressurize the scram air header resulting in rod insertion.
B. - INCORRECT - valves are Energized to actuate, NOT de-energized.
C. - INCORRECT - ARI pressure setpoint is 1071 psig, NOT 1037 psig
D - INCORRECT - ARI pressure setpoint is 1071 psig, NOT 1037 psig.

Question Source Mod ☑ **Memory Level** ☐ **Comprehension Level**

Question History:

SXD review - 7/21 - Add (via the ARI system) to the end of the stem. Removed "control rod insertion will begin within 15 …) from all distractors
AF - 8/23 - 2 correct answers - feels like 1071 psig will energize ARI valves. Changed all distractor and correct answer to a value vs. > than a number or less than a number.
MB - 8/ 24 Made changes as requested
AF - OK
MB - remove immediately from distractors - 11/8

Question	17

☑ ***RO***

☐ ***SRO***

Tier # 1 ***Group #*** 1

Importance 3.8

HC Obj:

295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown / 1

EA1.02 Ability to operate and / or monitor the following as they apply to SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown RRCS

Question

The plant was operating at 98% power when a transient occurred. Following the transient 3 SRVs opened. 2 minutes later, reactor pressure is stable with 1 SRV open. NO operator actions have been taken.

Which of the following is correct for these conditions?

Both Recirculation Pumps __________

A have tripped.

B are running normally.

C are running at minimum speed.

D are running but will trip in 1.9 minutes when a time delay times out.

Answer A

References INPO Question 23485
NOH01RRCS00-00, REDUNDANT REACTIVITY CONTROL SYSTEM (RRCS), p.13

Justification

References during Exam None

A. CORRECT - Following the transient, all SRVs opened. Reactor pressure has to be greater than 1071 psig for all valves to open. Reactor pressure greater than 1071 psig causes both Recirc Pumps to trip. A is the only correct answer.
B. INCORRECT - plausible because may NOT have hit a trip condition.
C. INCORRECT - plausible because recirc pumps have runbacks, operator may incorrectly believe a runback condition has been met.
D - INCORRECT - plausible because a 3.9 minute timer does exist on RRCS, however it is for SLC initiation, NOT Recirc pump trip.

Question Source Mod ☐ ***Memory Level*** ☑ ***Comprehension Level***

Question History:

SXD Review 7/21 - removed # of SRV's from stem. Removed "off" from Distractor A
AF - 8/23 - changed all SRV's to 3 SRV's In stem
MB - 8/ 24 Made changes as requested
AF - OK
MB - OK
MB - 11/17 "D" deleted currently

Question	18

☑ RO
☐ SRO

Tier # 1 **Group #** 1

Importance 3.9

HC Obj: EOP103E006

295038 High Off-site Release Rate / 9

EK3.02 Knowledge of the reasons for the following responses as they apply to High Off-Site Release Rate — System Isolations (CFR:41.8 to 41.10/45.3)

Question

HC.OP-EO.ZZ-0103/4, Reactor Building & Rad Release Control, step RR-5, directs isolation of all primary systems discharging into areas outside Primary Containment or Reactor Building, except those systems required to assure adequate core cooling and/or shutdown the reactor.

In accordance with the EOP Bases document, HC.OP-EO.ZZ-103/4. Reactor Building & Rad Release Control, these systems are specifically exempted from isolation, because:

A systems operated for RPV control are given a higher priority than stopping a rad release.

B isolation of a EOP support system requires an upgrade of the Emergency Classification.

C they are required to support alternate reactor depressurization methods.

D additional radiological consequences from them are unlikely.

Answer A

References
INPO Question 25837
BWROG, EPGs/SAGs Appendix B, section 9 Radioactivity Release control
HC.OP-EO.ZZ-103/4. Reactor Building & Rad Release Control Bases Document - p. 13 & 14

Justification

References during Exam None

Per EOP Bases document 103/104:
The objectives of RPV Control, Primary Containment Control, and the EPG contingencies are given higher priority than the objectives of Radioactivity Release Control. Systems that must be operated to perform other steps of the EPGs are therefore NOT isolated in this step.
A - CORRECT matches bases document
B - INCORRECT - NOT in accordance with bases document
C - INCORRECT - NOT in accordance with bases document
D - INCORRECT - NOT in accordance with bases document

Question Source Bank ☑ **Memory Level** ☐ **Comprehension Level**

Question History:
SXD review 7/22 - Minor editorial changes (added procedure)
AF - 8/23 - NO comments
MB - OK 11/8

Question 19

☑ ***RO***
☐ ***SRO***

Tier # 1 ***Group #*** 1

Importance 2.5

HC Obj:

600000 Plant Fire On Site / 8

AK1.01 Knowledge of the operational implications of the following concepts as they apply to the Plant Fire On Site — Fire Classifications by type (CFR: 41.8 to 41.10 /45.3) ,

Question

A fire occurs in the Upper Cable Spreading Room (Control Equipment Mezzanine Room 5403).

- The installed fire protection system automatically actuates.
- The room must be entered to determine if the fire has been extinguished.

(1) What is the classification of the fire that is expected in this area?

AND

(2) What safety hazard, from the automatic system actuation, shall be considered prior to operators entering the Cable Spreading Room?"

A (1) Class C
(2) Suffocation from oxygen depletion due to the discharge of CO_2 in the area

B (1) Class B
(2) Suffocation from oxygen depletion due to the discharge of halon in the area

C (1) Class C
(2) Suffocation from oxygen depletion due to the discharge of halon in the area

D (1) Class B
(2) Suffocation from oxygen depletion due to the discharge of CO_2 in the area

Answer A

References INPO Question 24855
NOH01FIRPRO-02, FIRE PROTECTION, p.55, p. 63 and p.85

Justification

References during Exam None

A- CORRECT - Class C fire due to electrical equipment in area, Suffocation due to discharge of CO_2
B - INCORRECT - NOT a Class B fire and NO halon in that room
C - INCORRECT - NOT expecting to get a halon discharge in that room
D - INCORRECT - NOT a class B fire

Question Source Mod ☑ ***Memory Level*** ☐ ***Comprehension Level***

Question History:

SXD review - 7/21 - Changed water to Halon
AF - 8/23 - OK
MB - OK 11/8

Question	20

Hope Creek RO Exam - Nov 2005

☑ RO
☐ SRO

Tier # 1 **Group #** 1

Importance 3.3

HC Obj:

295005 Main Turbine Generator Trip / 3

AK2.04 Knowledge of the interrelations between MAIN TURBINE GENERATOR TRIP and the following: Main generator protection (CFR: 41.7/45.8)

Question

Given the following conditions:
- The plant is operating at 20% power
- A main generator load reject has just occurred
- A fault in the control circuit causes a power/load unbalance trip during the load reject

Which of the following is the immediate expected response of the Turbine Control Valves (TCVs) and the Reactor Protection System (RPS)?

A TCVs throttle close, RPS trips

B TCVs throttle close, RPS does NOT trip

C TCVs fast close, RPS trips

D TCVs fast close, RPS does NOT trip

Answer D

References
Hope Creek Question - Q61307,
HC.OP-AB.BOP-0002 Additional Information /Automatic actions and notes
NOH01MNTURB-02, MAIN TURBINE CONSTRUCTION AND COMPONENTS, p. 66

Justification

References during Exam None

CORRECT - TCVs fast close, RPS does NOT trip. The load reject causes the TCVs to fast close. The fast closure does NOT initiate a RPS trip because turbine load is <30%. Since power is within the capacity of the BPVs, NO pressure transient will trip RPS.
INCORRECT - TCVs throttle close, RPS does trip. The load reject causes the TCVs to fast close. The fast closure does NOT initiate a RPS trip because turbine load is <30%. Since power is within the capacity of the BPVs, NO pressure transient will trip RPS.
INCORRECT - TCVs fast close, RPS does trip. The fast closure does NOT initiate a RPS trip because turbine load is <30%. Since power is within the capacity of the BPVs, NO pressure transient will trip RPS.
INCORRECT - TCVs throttle close, RPS does NOT trip. The load reject causes the TCVs to fast close

Question Source Bank ☐ **Memory Level** ☑ **Comprehension Level**

Question History:
SXD review - 7/21 - OK
AF - 8/23 - OK
AF - 10/13 possible K/A mismatch
MB - OK 11/8

Question 21

Hope Creek RO Exam - Nov 2005

☑ **RO**
☐ **SRO**

Tier # 1 **Group #** 2

Importance 3.2

HC Obj:

295002 Loss of Main Condenser Vac / 3

AA2.02 Ability to determine and interpret the following as they apply to Loss of Main Condenser Vacuum's Reactor Power

Plant Specific:(CFR: 41.10/43.5/ 45.13)

Question

Given the following:
- All four Circulating Water Pumps are in operation
- Plant is operating at 100% power
- Circulating Water System Inlet temperature is 80°F
- Indicated Main Condenser pressure is 2.75 in HgA

Then, AP501 is removed from service.

Assume the remaining Circulating Pumps' Discharge Valves are reopened fully, NO rise in basin temperature and NO other operator actions are taken.

Using the provided figure, what is the expected condenser backpressure and what is the expected change in reactor power (if any) following the removal of Circulating Water Pump AP501 from service?

A 3.5 in HgA, reactor power increases (ie. Greater than 2%)

B 3.5 in HgA, reactor power stays the same (ie. Doesn't change more than 2%)

C 4.15 in HgA, reactor power increases (ie. Greater than 2%)

D 4.15 in HgA, reactor power stays the same (ie. Doesn't change more than 2%)

Answer B

References Hope Creek Question - Q55132
HC.OP-SO.DA-0001, Rev. 35, Attachment 5

Justification

References during Exam Attachment 5 from HC.OP-SO.DA-0001

A - INCORRECT - Reactor power should NOT change with a decrease in vacuum. If anything reactor power may go down a little bit due to increased condenser temperature and reduced condenser subcooling
B- CORRECT- 3.5 inHgA. If CW inlet temp does NOT change, then the condenser vacuum rises vertically on the graph until it reaches the line for three pump operation @ 80 degF. Since the inital back-pressure of 2.75 indicates 100 percent CF. Reactor power should remain the same
C - INCORRECT 4.15 - 3 pump ops at 70 percent CF.
D - INCORRECT 4.15 - 3 pumps ops at 70% CF

Question Source Bank ☐ **Memory Level** ☑ **Comprehension Level**

Question History:

SXD review 7/21 - OK
JD 8/2 - K/A asking for Reactor Power
8/3 - initially was going to change question to add reactor power change, decided to ask Steve on Monday
8/4 Re-wrote questions
AF - 8/23 change G to capitalize in "A"
MB - 8/ 24 Made changes as requested
AF - minor changes
MB - Incorporated changes
MB - added using provided figure 11/8

Question 22

☑ **RO**
☐ **SRO**

Tier # 1 **Group #** 2
Importance 3.4

HC Obj: NOH01RCIC00-R7

295008 High Reactor Water Level / 2

AK3.06 Knowledge of the reasons for the following responses as they apply to High Reactor Water Level — RCIC Turbine Trip

Question

During a transient, the RO started the RCIC system for reactor water level control using the appropriate operating procedure. Level rose above the High Reactor Water level at 54" after which it lowered below the Low Reactor Water level at -38".

Which of the following describes the reason for, and expected response of RCIC during the reactor water level transient?

A The RCIC Trip and Throttle Valve (HV-4282) will close on High Water Level and RCIC will automatically restart on Low Reactor Water Level.

B The RCIC Trip and Throttle Valve (HV-4282) will close on High Water Level and RCIC will have to be reset and manually started on Low Reactor Water Level.

C The RCIC Steam Supply Valve (F045) will close on High Water Level and RCIC will automatically restart on Low Reactor Water Level.

D The RCIC Steam Supply Valve (F045) will close on High Water Level and RCIC will have to be reset and manually started on Low Reactor Water Level.

Answer C

References NOH01RCIC00-02, REACTOR CORE ISOLATION COOLING SYSTEM, p22-23

Justification

References during Exam None

A - INCORRECT - Trip and Throttle valve does NOT close on Level 8
B - INCORRECT - Trip and Throttle valve does NOT close on Level 8
C - CORRECT - Steam supply valve will close and RCIC will auto restart at Level 2
D - INCORRECT - RCIC will auto restart at Level 2

Question Source Mod ☑ **Memory Level** ☐ **Comprehension Level**

Question History:

SXD review 7/22 - LOD = 1 - re-write question
8/3 - re-wrote question
AF - 8/23 - changed 58" to 54"
MB - 8/ 24 Made changes as requested
AF - OK
MB - 11/8 - OK

Question 23		Hope Creek RO Exam - Nov 2005
☑ RO ☐ SRO	Tier # 1 Group # 2 Importance 3	HC Obj:

295009 — Low Reactor Water Level / 2

AK1.02 — Knowledge of the operational implications of the following concepts as they apply to the Low Reactor Water Level — Recirculation pump net positive suction head

Question

The plant is currently at 27% power. Plans for the shift are to continue the startup and power ascension. A malfunction in the Feedwater Control System has resulted in the following:

- RPV level is 25 inches and trending down
- Total Feedwater flow is 2.5 mlb/hr and steady
- 3 Circ Water pumps are running
- Condenser Vacuum is 3.8" HgA and degrading

Assume NO operator actions have been taken. Which of the following statements is correct regarding the Reactor Recirculation system response based on these CURRENT plant conditions?

A Speed Limiter 1 (30% flow) is actuated to ensure Recirculation Pump net positive suction head protection based on feedwater flow.

B Speed Limiter 2 (45% flow) is actuated to ensure Recirculation Pump net positive suction head protection based on RPV level.

C Speed Limiter 2 (45% flow) is actuated to bring Condenser Vacuum back to normal.

D Speed Limiter 1 (30% flow) is actuated to bring Condenser Vacuum back to normal.

Answer A

References
New Question
NOH01RECIRC-02, Reactor Recirculation System, P. 53-55

Justification

References during Exam None

A - CORRECT - Total FW flow is ~17% which is < 20%, this causes a Speed Limiter #1 runback to ensure Recirc Pump NPSH
B - INCORRECT - Speed Limiter 1 is actuated, NOT Speed Limiter 2
C - INCORRECT - Speed Limiter 1 is actuated, NOT Speed Limiter 2
D - INCORRECT - Condenser vacuum is rising but still within normal limits. Must be > 4.5" to cause a Recirc pump runback.

Question Source New ☐ Memory Level ☑ Comprehension Level

Question History:

SXD Review 7/21 - minor editoral comments
AF - 8/23 - changed "A" to FW flow vs. RPV level, also AF to check numbers
MB - 8/ 24 Made changes as requested - AF still to verify Numbers
SXD - OK
AF - OK
MB - OK 11/8

Question 24

☑ **RO** ☐ **SRO**

Tier # 1 **Group #** 2

Importance 3.1

HC Obj: EOP202E003

295029 High Suppression Pool Wtr Lvl / 5

EK2.07 Knowledge of the interrelations High Suppression Pool Wtr Lvl and the following Drywell/ containment water level:(CFR: 41.7 /45.7/45.8)

Question

An Override step in HC.OP-EO.ZZ-0202, Emergency Depressurization, directs the operator to open the Inboard MSL Drain Valve (AB-HV-F016) when Containment water level is expected to exceed 48 feet.

Which one of the following describes the reason for this action?

Opening the Inboard Main Steamline Drain Valve _______________

A maintains the availability of the Main Steamline drain path for reactor vessel pressure control if required.

B ensures as much heat energy as possible is rejected to the Main Condenser to minimize the dynamic loading on Containment.

C maintains Containment water level below the SRV solenoids by establishing a drain path from the reactor vessel to the Main Condenser.

D ensures the SRV Tail Pipe Level Limit is NOT exceeded prior to emergency depressurization.

Answer A

References
INPO Question 21944
BWROG EPG/SAG's App. B - P 326
HC.OP-EO.ZZ-0202 flowchart
HC.OP-EO.ZZ-0202, Emergency Depressurization Bases, p.5

Justification

References during Exam None

A - CORRECT - per the BWROG guidelines - If primary containment water level rises above the elevation of the SRV solenoids, the SRVs may NO longer be operable. Other methods must then be used to control RPV pressure and prevent repressurization. Opening the inboard main steam line drain valve preserves the main steam line drains for future use.
B - INCORRECT but plausible, while Opening AB-HV-F016 does NOT reject any heat to the Main Condenser it could reject heat to the condenser if the F019 and F021 were open.
C - INCORRECT but plausible, while opening AB-HV-F016 does NOT necessarily maintain CNMT water level below the SRV solenoids, it may open a drain path to the main condenser.
D - INCORRECT but plausible, while opening AB-HV-F016 does NOT drain water from the steam lines, it could if both F019 and F021 were open.

Question Source Mod ☑ **Memory Level** ☐ **Comprehension Level**

Question History:
SXD reviewed 7/22 - OK
AF - OK
MB - OK 11/8
MB - 11/17 - Ok

Question 25		**Hope Creek RO Exam - Nov 2005**
☑ *RO* ☐ *SRO*	*Tier #* 1 *Group #* 2 *Importance* 3.8	*HC Obj:* RMSYS0E003 - d

295034 Secondary Containment Ventilation High Radiation / 9

EA1.01 Ability to operate and/ or monitor the following as they apply to Secondary Containment Ventilation High Radiation: Area radiation monitoring system:(CFR41.7/45.5/45.6)

Question

Refueling Floor radiation levels are rising as indicated by rising readings on the 3 refuel floor ARMs (New fuel storage vault channel A & B [RE-4813 A & B] and Spent fuel Pool ARM [RE-6607]).

What automatic actions will occur if ALL of these 3 area radiation monitors reach their "HIGH" setpoint?

I. Control Room Annunciator "NEW FUEL CRITICAL RAD HIGH" (E6-A4) will ALARM
II. Refuel Floor Evacuation Alarm is actuated
III. Reactor Building Ventilation System will Isolate
IV. Filtration Recirculation Ventilation System will Auto Start.

Note - Question is looking for ONLY systems/alarms that receive inputs from these AREA Radiation Monitors.

A II Only

B I and II Only

C II, III and IV

D I, II, III and IV

Answer B ***References*** NOH04000221C-01, RADIATION MONITORING SYSTEM p. 14

Justification ***References during Exam*** None

A - INCORRECT - In addition to the Evacuation alarm being sounded, CR will also receive the E6-A4, New Fuel Critical Rad High
B - CORRECT - per lesson plan will receive both the Refuel floor evacuation alarm and the new fuel critical alarm.
C - INCORRECT - RBVS and FRVS don't receive an input from Area Rad alarms
D - INCORRECT - RBVS and FRVS don't receive an input from Area Rad alarms

Question Source New ☑ ***Memory Level*** ☐ ***Comprehension Level***

Question History:

New Question 9/3
SXD - OK
AF - minor changes
MB - Incorporated changes
MB - 11/8 - OK

Question 26		**Hope Creek RO Exam - Nov 2005**
☑ RO ☐ SRO	**Tier #** 1 **Group #** 2	**HC Obj:** NOH01EOP103-00 Obj. 6
	Importance 2.9	

295036 Secondary Containment High Sump/Area Water Level / 5

EK1.01 Knowledge of the operational implications of the following concepts as they apply to the Secondary Containment High Sump/ Area Water Level — Radiation releases(CFR:41.8 to 41.10/45.3)

Question

Given the following:

- The RCIC turbine is on fire and the Fire Brigade has been actively spraying water on RCIC.
- The Fire Brigade reports steam coming out of the RCIC steam supply line.
- The Fire Brigade has just reported that the fire is under control and they should be securing shortly.
- RCIC pump room (4110) Floor level is 6"
- RHR Pump room "B" (4109) Floor level is 4"
- RHR Pump room "D" (4107) Floor level is 4"
- Core Spray Pump room "B" (4104) Floor level is 3"
- Core Spray Pump Pump room "D" (4105) Floor level is 3"
- "D" South Reactor Building Sump pump (DP-265) is tagged out for motor replacement
- Reactor Building HVAC Exhaust Rad level is 1.5 x 10-3 microcuries/ ml
- Refueling Floor HVAC Exhaust Rad level is 1.0 x 10-4 microcuries/ml

In addition to restoring floor levels to normal using all available sump pumps, which of the following correctly states the proper operator actions to be taken and/or the reasons for those actions:

I. Isolate all water discharging into the RHR pump rooms in order to terminate level challenges to RHR pump rooms.
II. Runback Recirc and initiate a manual scram.
III. Emergency Depressurize the Reactor in order to place primary in it's lowest possible energy state.
IV. Verify FRVS is inservice and RBVS is isolated in order to prevent/minimize off-site releases due to high radiation levels.

A I - Only

B II and IV

C II, III and IV

D I, II and III

Answer B

References HC.OP-EO.ZZ-0103/4, BASES pages 1, 3, 7, 8, 10
NOH01EOP1034, Lesson Plan p. 15

Justification

References during Exam HC.OP-EO.ZZ-0103/4 with entry conditions blacked out.

A - INCORRECT - Would NOT want to isolate Fire Protection water discharging to the RHR pump room.
B - CORRECT - Max Safe OP limit has been exceeded in 1 areas, the RCIC Room. Due to High radiation on in the reactor building you must assume RCS is discharging to Rx building from RCIC steam line. A manual Scram needs to be initiated. Due to High rad, need to start FRVS
C - INCORRECT - Max Safe OP limit has NOT been exceed in 2 or more areas. Therefore you don't want to Emergency Depressurize.
D - INCORRECT don't want to stop Fire Protection, don't need to Emergency Depressurize.

Question Source New ☐ Memory Level ☑ Comprehension Level

Question History:

New 9/20
SXD - Minor comments to read better
MB - 9/28 - Made changes as requested
AF - suggests removing B, C, D RHR pump sumps as they won't get water from Fire Protection unless really bad, NOT sure where Rad is coming from
MB - talk to SXD perhaps, Re-sample
MB - 10/27 - re-wrote question to have ALL affected equipment come from the South Reactor Building sump. Since all this equipment is located on the bottom of the Reactor Building it seems plausible that with Fire Water coming into RCIC pump room, sump pump could be overloaded and water could backup into other connected rooms through the sump. Rad is coming from RCIC steam line.
Val - times sump running is irrelevent - remove time, editorial change to Condition IV
MB - took times out, changed levels on RHR to 4", made editorial change to Condition IV
MB - 11/17 minor editorial changes
MB - 12/2 - removed RBVS is running from stem.

Question 27

Hope Creek RO Exam - Nov 2005

☑ **RO**
☐ **SRO**

Tier # 1 **Group #** 2

Importance 3.1

HC Obj:

500000 High CTMT Hydrogen Conc. / 5

EK2.02 Knowledge of the interrelations between High CTMT Hydrogen Conc. And the following — Containment oxygen monitoring systems(CFR: 41.7 / 45.7 /45.8)

Question

Given the following conditions:

Hope Creek was shutting down due to leaking fuel and the H2O2 monitors are inservice for de-inerting, when a transient occurred and the following conditions are present:

- Drywell H2 concentration is reading 1.5% by volume
- Drywell O2 concentration is reading 5.5% by volume
- Drywell Pressure is 1.5 psig and stable
- Reactor water level is +10" and rising slowly (lowest level ~ -0")
- Drywell temperature is 140°F

I - Assuming NO other operator actions have occurred, what is the status of the H2/O2 monitors?

II - Assuming the above readings are correct and Containment venting CANNOT be performed, what actions shall be taken with regards to the H2 Recombiners in accordance with HC.OP-EO.ZZ-0102, Primary Containment Control?

A I - H2/O2 monitors are in-service
II - H2 Recombiners shall be placed in service.

B I - H2/O2 monitors are in-service
II - H2 Recombiners shall NOT be placed in service.

C I - H2/O2 monitors are isolated
II - H2 Recombiners shall NOT be placed in service

D I - H2/O2 monitors are isolated
II - H2 Recombiners shall be placed in service.

Answer A

References
NOH01H202AN-01, Hydrogen Oxygen Analyzer System - p. 17
NOH01H2RECM-00, CONTAINMENT HYDROGEN RECOMBINER SYSTEM, p.8
HC.OP-EO.ZZ-0102(Q)-FC, PRIMARY CONTAINMENT CONTROL, step PC/H-1

Justification

References during Exam EOP-102

A. CORRECT - H2 Recombiners shall be placed in service due to High H2 concentration per EOP 102, concentration > 0.5% and < 2%
B. INCORRECT - Never received Containment Isolation, CNMT Isolation pressure is 1.68 psig
C. INCORRECT - H2 Recombiners shall be placed in service due to High H2 Concentration per EOP 102
D. INCORRECT - H2 Recombiners shall be placed in service due to High H2 Concentration per EOP 102

Question Source New ☐ **Memory Level** ☑ **Comprehension Level**

Question History:

SXD review - 7/21 - OK
AF - 8/23 - feels like "A" should be correct answer, for O2 monitors to be reading anything they would have had to been overriden and placed back in service. - Normally monitors are NOT in service and they would be reading. Changed stem to Drywell pressure of 1 psig and made "A" the correct answer. AF- to relook at question. Have SD look at question again. Made all monitors H2/O2 monitors
SD - Change justification
MB - 9/26 - Made changes as requested.
SXD - AF to relook at question
AF- change OPERABLE to In-service added EOP-102, operators will already have procedure anyway
MB - Incorporated changes
Val - asked change to lowest level to -250" vs. 0"
MB - couldn't change stem to -250" because this would cause H2O2 monitors to isolate, changed stem to add leaking fuel 11/8 - OK

MB - 11/17 - added DW temp of 140°F and H_2O_2 monitors inservice for de-inerting.

Question 28

- [x] RO
- [ ] SRO

Tier # 2 **Group #** 1

Importance 3.6

HC Obj: RHRSYSE012

203000 RHR/LPCI: Injection Mode

A1.04 Ability to predict and/or monitor changes in parameters associated with operating the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) controls including: (CFR: 41.5 / 45.5) — System pressure

Question

Hope Creek was at 100% when a Small Break LOCA occurred concurrent with a loss of ALL High pressure injection.

The Reactor is being depressurized using the SRV's due to level NOT being able to be maintained above TAF. ALL LPCI and Core Spray pumps have started as required. Reactor Pressure is 400 psig at this time.

Concerning the "A" RHR system ONLY, which of the following correctly describes the EXPECTED system parameters and configuration.

A LPCI "A" Injection valve is OPEN, "A" system flow indicates 10,000 gpm, "A" pump discharge pressure is approximately 400 psig.

B LPCI "A" Injection valve is OPEN, "A" system flow indicates 0 gpm, "A" pump discharge pressure is approximately 340 psig.

C LPCI "A" Injection valve is CLOSED, "A" system flow indicates 2,300 gpm, "A" pump discharge pressure is approximately 340 psig.

D LPCI "A" Injection valve is CLOSED, "A" system flow indicates 0 gpm, "A" pump discharge pressure is appoximately 340 psig.

Answer B

References Brunswick NRC Exam 2003, Q. 10
NOH01RHRSYSC-03, RESIDUAL HEAT REMOVAL SYSTEM, p.9-10, 53

Justification

References during Exam None

A - INCORRECT - System pressure is > RHR shutoff head, RHR flow should read 0
B - CORRECT - Injection valves open when system pressure < 450 psig, however Reactor pressure is still > shutoff head of pump, therefore indicated flow = 0 gpm.
C - INCORRECT - LPCI Injection valves OPEN when Rx pressure < 450 psig.
D - INCORRECT - LPCI Injection valves OPEN when Rx pressure < 450 psig.

Question Source Mod

- [ ] Memory Level
- [x] Comprehension Level

Question History:

9/7 - New - had to re-sample K/A, initial K/A was NOT an RO level. RO's NOT required to know bases of Tech Specs for RHR.
SXD - Check Hope Creek References for correct pressures
MB - Per HC lesson plan - LPCI pump shutoff head = 366 psig, normal pressure 171 psig w/ 10,000 gpm flow. Also per lesson plan LPCI Injection valves OPEN when reactor pressure lowers to < 450 psig.
SXD - OK
AF - changed pressure to 340 psig, changed valves to valve
MB - Incorporated change
MB 11/8 - OK

Question 29

Hope Creek RO Exam - Nov 2005

☑ **RO** ☐ **SRO**

Tier # 2 **Group #** 1

Importance 3.5

HC Obj:

205000 Shutdown Cooling

A3.03 Ability to monitor automatic operations of the Shutdown Cooling System(RHR Shutdown Cooling Mode) including lights and alarms (CFR:41.7/45.5)

Question

Given the following Plant conditions:

Hope Creek is in OPCON 3 Cooling down for a Refueling Outage,
"A" Shutdown Cooling is being placed in service and is currently in the following status:

- "A" RHR fill and vent has been completed. However, the F007A - RHR Pump min-flow valve's breaker was inadvertantly left closed.
- "A" RHR Loop has been warmed up.
- Both Reactor Recirc Pumps have been secured.

The RO is lining up "A" RHR system for Shutdown cooling and valves are currently lined up as follows:

- F009 - Shutdown Cooling INBD ISLN MOV - Open
- F008 - Shutdown Cooling OUTBD ISLN MOV - Open
- AP202 RHR PUMP - Running
- F015A - RHR Loop A Ret to Recirc - Throttled Open
- F007A - "A" RHR pump min-flow - Closed
- F024A - "A" RHR Full Flow test valve - Closed
- F027A - "A" Torus Spray Inj valve - Closed

To reduce an RCS cooldown the RO throttles closed on F048A when the following alarm is received.

"RHR A S/D CLG & MIN FL VLV OPEN" alarm is received in the control room.

Assuming NO Operator actions are taken, which of the following conditions will result:

A F008 and F009 will Auto Close when the min-flow valve F007A begins to open.

B F008 and F009 will Auto Close on Low RPV level 3 (+12.5")

C NO Auto Actions will occur, this is an expected alarm for the above conditions.

D F008 and F009 will Auto Close on Low RPV level 1 (-129")

Answer B

References NOH01RHRSYSC-03, RESIDUAL HEAT REMOVAL SYSTEM, p. 30

Justification

References during Exam None

A - INCORRECT - F008 and 9 will NOT Auto Close based on min-flow valve position.
B - CORRECT - Having the Min-flow valve open and taking suction from Reactor vessel will cause Reactor Vessel to lower, when vessel level reaches Low RPV Level 3, F008 and 009 will Auto Close.
C - INCORRECT - Reactor vessel will lower due to Min-flow open and taking suction Reactor vessel.
D - INCORRECT - F008 and F009 will auto close on Low RPV level 3 and level should NOT get to Low RPV level 1.

Question Source New ☐ **Memory Level** ☑ **Comprehension Level**

Question History:

SXD review 7/27 - OK
AF 8/23 - made all bullets, changed "A" to when F007A starts to open vs. gets full open. AF feels this could be a correct answer because 3 minutes after valve gets full open, the valves will close on low level.
MB - 8/24 - Made changes as requested
SXD - OK
AF - OK
MB - 11/8 - minor editorial changes to stem

Question 30

Hope Creek RO Exam - Nov 2005

☑ **RO**
☐ **SRO**

Tier # 2 **Group #** 1

Importance 2.8

HC Obj: STACS0E018

400000 Component Cooling Water

A2.02 Ability to (a) predict the impacts of the following on the CCWS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation: (CFR: 41.5 / 45.6) High/low surge tank level

Question

SACS and TACS are in a normal at power lineup. TACS is being supplied by SACS loop "A". A leak develops in the line to the Recirculation MG set "A" lube oil cooler which is greater than the makeup capacity to the expansion tank causing a low-low-low expansion tank level.

Select the response of SACS/TACS, with NO operator action?

A TACS will transfer to SACS loop "B". Later, TACS will automatically isolate.

B TACS will remain aligned to SACS loop "A".

C TACS will transfer to SACS loop "B". TACS is only isolated manually.

D TACS to SACS connections will immediately isolate on Low-Low-Low level in "A" SACS expansion tank.

Answer A

References HC.OP-AB.COOL-0002
Hope Creek Bank - Q56926

Justification

References during Exam None

A - CORRECT - Low-Low-Low level will cause TACS to transfer. The leak will not be isolated so on a low-low-low level in the B SACS loop expansion tank, TACS will isolate.
B - INCORRECT -Will transfer to SACS loop "B".
C - INCORRECT - TACS will automatically isolate on low low low level in the "B" expansion tank.
D - INCORRECT - Will transfer to SACS loop "B".

Question Source Bank ☑ **Memory Level** ☐ **Comprehension Level**

Question History:

New 9/7
SXD - OK
RJC - 10/6 - SRO rather than RO (are they required to Know bases)
SXD - wait on Archie, find part in NUREG where only match 1 part of 2 part K/A
MB - Found part - ES-401 pages 5 & 6
AF - similar question on audit, SXD to resolve
SXD - Re-sample
MB - 10/27 - re-sampled - bank Question
MB - 11/8 minor change to distractor A

Question 31

Hope Creek RO Exam - Nov 2005

☑ **RO**
☐ **SRO**

Tier # 2 **Group #** 1

Importance 3.3

HC Obj:

206000 HPCI

K5.05 Knowledge of the operational implications of the following concepts as they apply to the HPCI — Turbine speed control

Question

Given the following conditions:
- The HPCI system running in automatic at rated flow.
- The flow element providing feedback to the flow controller begins to fail downscale, slowly.

How will actual HPCI turbine speed and system flow initially respond?

A Turbine speed will increase and flow will increase

B Turbine speed will decrease and flow will decrease

C Turbine speed will decrease and flow will remain at rated

D Turbine speed will increase and flow will remain at rated

Answer A

References Hope Creek Question Q56448
NOH01HPCI00-02, HIGH PRESSURE COOLANT INJECTION SYSTEM, p.30

Justification

References during Exam None

Correct answer:turbine speed will increase and flow will increase

The following distractors are incorrect as follows:
·turbine speed will increase and flow will remain at rated-Incorrect- As flow feedback lowers, controller will raise turbine speed and, with it actual flow rate will raise
·turbine speed will decrease and flow will decrease-Incorrect- As flow feedback lowers, controller will raise turbine speed and, with it actual flow rate will raise
·turbine speed will decrease and flow will remain at rated-Incorrect- As feedback lowers, controller will raise turbine speed and, with it actual flow rate will raise

Question Source Bank ☐ **Memory Level** ☑ **Comprehension Level**

Question History:

SXD review - 7/21 - OK
AF - OK
AF - K/A mismatch
MB - 11/8 - OK
MB - 11/17 - added initially

Question 32

☑ **RO**
☐ **SRO**

Tier # 2 **Group #** 1

Importance 3

HC Obj:

209001 LPCS

K2.01 Knowledge of electrical power supplies to the following Pump power (CFR41.7)

Question

Hope Creek has experienced a transient and a partial loss of Offsite power.

Current conditions are as follows:

- 500KV bus 10X faults and trips

- Reactor has SCRAMMED and all rods are INSERTED
- Reactor water level is -135" and rising slowly
- Drywell Pressure is 1.35# and lowering slowly (Max. Pressure ~1.5#)
- "C" CS pump NORMAL/EMERGENCY TAKEOVER switch is(was) in the EMERGENCY position
- A and B Diesel Generators FAILED TO START

Based on the above conditions and using the provided drawing, what is the status of the Core Spray Pumps?

A All Core Spray Pumps are running

B A, B, and D Core Spray Pumps are running

C Only C Core Spray Pump is running

D Only D Core Spray Pump is running

Answer B

References

NOH01CSSYS0-01, CORE SPRAY SYSTEM p.16
NOH01EAC00-02, CLASS 1E AC POWER DISTRIBUTION
066-01: Class 1E AC Power Distribution (Training drawing)
027-01: Core Spray System (Training Drawing)

Justification

References during Exam E-0001

A: INCORRECT - "C" CS pump will NOT have started because it's Takeover switch is in the EMERGENCY Position
B: CORRECT - The Loss of the Red Lion Line and the Circuit breaker faults will have caused a loss of Bus Section 10X and Station Service XFMR 1BX501, however 1AX501 will still be energized from Offsite power, therefore power to 10A402 and 10A404 will auto transfer to 1AX501 causing all of the 4.16KV buses to be energized. As stated above "C" CS pump will NOT have started, leaving A, B and D CS pumps running.
C: INCORRECT - "C" CS pump will NOT have started because it's Takeover switch is in the EMERGENCY Position
D: INCORRECT - A and B Diesel Generators failing to start will NOT cause their respective buses to be de-energized because they will have received power from 1AX501

Question Source New ☐ **Memory Level** ☑ **Comprehension Level**

Question History:

SXD reviewed 7/22 - give students 500KV switchyard print
AF - 8/23 - add bullets, made failed to open all caps. Made "B" correct answer, changed reference to be handed out to E-0001 to be given out.
MB - 8/24 - Made changes as requested
AF - OK
Val - replace top 3 breakers with 500KV bus 10X faults and trips
MB - Made changes as requested

Question	33

Hope Creek RO Exam - Nov 2005

☑ RO
☐ SRO

Tier # 2 **Group #** 1

Importance 3.8

HC Obj:

211000 SLC

K4.04 Knowledge of SLC design feature(s) and or interlock(s) which provide for the following — Indication of fault in explosive valve firing circuits (CFR41.7)

Question

Hope Creek was operating at full power when an instrument air line break caused the outboard MSIVs to go closed. The following then occurred:

- The reactor failed to scram and attempts to drive rods were unsuccessful.
- The CRS ordered SLC injection.
- Both SLC pump AP208 and BP208 START pushbuttons have been depressed.
- SLC pump control bezel start pushbuttons are backlit RED.
- The squib valve continuity lights are LIT.
- Pump discharge pressure is 1395 psig.
- Reactor Pressure is currently 1025 psig.

Based on these indications which of the following correctly describes the status of the SLC system?

A SQUIB valves are CLOSED, with SLC pumps running therefore, SLC is NOT injecting.

B SQUIB valves are OPEN, with SLC pumps running, therefore SLC is injecting.

C SQUIB valves are OPEN, however, the SLC pumps are NOT running, therefore SLC is NOT injecting.

D SQUIB valves are CLOSED AND SLC pumps are NOT running, therefore, SLC is NOT injecting.

Answer A

References INPO Question 20790
NOH01SLCSYS-00, STANDBY LIQUID CONTROL SYSTEMS, p.27-29

Justification

References during Exam None

A - CORRECT - the pump control bezel start pushbuttons backlit RED, along with pump discharge pressure of 1395 psig indicate the pumps are running. Squib valve continuity lights being lit, indicate valves are closed, therefore NO injection is occurring.
B - INCORRECT - Squib valves are closed
C - INCORRECT - Squib valves are closed
D - INCORRECT - SLC pumps are running

Question Source Mod ☐ Memory Level ☑ Comprehension Level

Question History:

SXD review 7/21 - Minor editorial changes
AF - 2nd bullet - CRS NOT SS, added give figure of Control Bezels - AF to find figure number. Made LIT all caps.
MB - 8/24 - Made changes as requested
SXD - Don't need figure
MB - Removed figure from references
SXD -OK
AF - OK
MB - 11/8 - changed closed to CLOSED

Question	34

Hope Creek RO Exam - Nov 2005

☑ RO
☐ SRO

Tier #	2	Group #	1

Importance 3

HC Obj:

212000 RPS

K3.11 Knowledge of the effect that a loss or malfunction of the RPS will have on the following — Recirculation system (CFR41.7/45.6)

Question

Given the following:

- The Reactor is initially at 20% power
- The Main Turbine is synchronized to the grid and loaded
- The RX RECIRC PUMPS RPS TRIP BYP alarm (C1-E3) is NOT illuminated
- A loss of "B" RPS Bus has occurred

What is the operational effect of a fast closure of all Turbine Control Valves during this condition?

A EOC-RPT trip of Recirculation Pump A and NO trip of Recirculation Pump B

B EOC-RPT trip of both Recirculation Pumps

C EOC-RPT trip of Recirculation Pump B and NO trip of Recirculation Pump A

D Both Recirculation Pumps running with half-scram inserted

Answer B

References
Hope Creek Question - Q61263,
HC.OP-AB.ZZ.IC-0003 discussion section step 2
NOH01RECIRC-02, Reactor Recirculation System, p.37 and p. 69

Justification

References during Exam None

Justification:
·EOC-RPT trip of both Recirculation Pumps - Correct, loss of RPS bus power, at any reactor power level, in conjunction with the cited Turbine Control Valve fast closure will result in EOC-RPT trip of both Recirculation Pumps. This occurs due to a loss of the automatic bypass for EOC-RPT when less than about 30% power (first stage pressure less than 135.7 psig). The keylock bypass of the EOC-RPT trip is removed with the Main Turbine loaded. The RX RECIRC PUMPS RPS TRIP BYP alarm is cleared when the RECIRC PUMP TRIP A/B SYSTEM DISABLE switch is placed in the NORM position. This defeats the bypass of the RPT trips.
·EOC-RPT trip of Recirculation Pump A and NO trip of Recirculation Pump B - Incorrect, both pumps will trip.
·EOC-RPT trip of Recirculation Pump B and NO trip of Recirculation Pump A - Incorrect, both pumps will trip.
·Both Recirculation Pumps running with half-scram inserted - Incorrect, both pumps will trip.

Question Source Bank ☐ **Memory Level** ☑ **Comprehension Level**

Question History:
SXD Review 7/21 - OK
AF - OK
MB - 11/8 - OK

Question 35

☑ ***RO***
☐ ***SRO***

Tier # 2 ***Group #*** 1

Importance 2.9

HC Obj: IRMSYSE003

215003 IRM

K4.04 Knowledge of the IRM design feature(s) and or interlock(s) which provide for the following Varying system sensitivity levels using range switches (CFR41.7)

Question

Which of the following correctly explains how IRM system sensitivity level is varied using the range switches:

Placing the Range switch from Range 6 to Range 7 ____________

A changes which pulse height discriminators are placed in service.

B changes which input resistors and which attenuators are placed in service.

C changes which log integrators are placed in service.

D changes which voltage pre-amps and which attenuators are placed in service.

Answer D

References NOH01IRMSYS-01, INTERMEDIATE RANGE MONITORING SYSTEM - p. 8-9

Justification

References during Exam None

A - INCORRECT - Pulse height discriminators are used in the SRM detectors NOT the IRMs
B - INCORRECT - Input resistors are used in the APRMs NOT the IRMs
C - INCORRECT - Log integrators are used in the SRM detectors
D - CORRECT - changing the range switch from 6 to 7 will change both which voltage pre-amp is placed in service and which attenuator is placed in service

Question Source New ☑ ***Memory Level*** ☐ ***Comprehension Level***

Question History:

New 9/20
SXD - OK
AF - OK
MB - 11/8 - changed distractor C to Log integrator.

Question 36

Hope Creek RO Exam - Nov 2005

☑ **RO**

☐ **SRO**

Tier # 2 **Group #** 1

HC Obj: IRMSYSE008, SRMSYSE01

Importance 2.5

215003 IRM

K2.01 Knowledge of electrical power supplies to the following IRM Channels/ detectors (CFR41.7)

Question

A Loss of 24VDC occurs to 1AD307 DC Distribution Panel.

Which of the following describes the effect on NI's:

	SRM	IRM	APRM
A	NO change	fails low	NO change
B	fails low	NO change	NO change
C	fails low	fails low	NO change
D	fails low	fails low	fails low

Answer C

References NOH01DCELEC-00, DC ELECTRICAL DISTRIBUTION, p.38
NOH01IRMSYS-01, Intermediate Range Monitoring System, p26
Simplified Training prints for SRM, IRM and APRMs

Justification

References during Exam None

A - INCORRECT - SRM's are powered from 24VDC and would fail downscale
B - INCORRECT - IRM's are powered from 24VDC and would fail downscale
C - CORRECT - SRM's and IRM's are powered from 24 VDC and would fail downscale, APRM's are powered from 120 VAC panels and would remain unchanged
D - INCORRECT - APRM's are powered from 120 VAC and would NOT fail downscale

Question Source New ☐ **Memory Level** ☑ **Comprehension Level**

Question History:

SXD Review - 7/21 - LOD 1.0 - rewrite question to make it more difficult
8/3 - Re-wrote question
AF - is it memory or comprehension. Changed to comprehension
MB - 8/24 - Made changes as requested
AF - OK
MB - OK - similar to 77

Question 37

☑ **RO**
☐ **SRO**

Tier # 2 **Group #** 1

Importance 3.4

Hope Creek RO Exam - Nov 2005

HC Obj: SRMSYSE006

215004 Source Range Monitor

K1.02 Knowledge of the physical connections and/or cause-effect relationships between Source Range Monitor and the following: Reactor Manual Control

Question

The following plant conditions exist:

- Reactor Mode Switch is in STARTUP/STANDBY
- All IRMs are on Range 3
- Source Range Monitor (SRM) A is reading 0.5 cps
- SRMs B and C are reading 8.3 x 10E4
- SRM D mode switch is in STANDBY
- A rod block signal has been generated.

Which one of the following has caused the rod block?

A SRM Detector Wrong Position

B SRM Downscale

C SRM Upscale

D SRM Inoperable

Answer D

References INPO Question 21837
NOH01SRMSYS-01, SOURCE RANGE MONITORING (SRM) SYSTEM, p32

Justification

References during Exam None

A - INCORRECT - Detector Wrong Position does NOT generate a Rod Block
B - INCORRECT - SRM Downscale bypassed with Associated IRM range 3
C - INCORRECT - SRM Upscale doesn't come in until 1E5 cps
D - CORRECT - With Reactor Mode Switch NOT in RUN and SRM detector channel switch out of operate a Rod Block on SRM INOP will be generated.

Question Source New ☐ **Memory Level** ☑ **Comprehension Level**

Question History:

New 9/7
SXD - OK
AF - Minor changes
MB - Incorporated changes
MB - 11/8 - OK
MB - 11/17 - put all IRM's on range 3 vs. some on Range 2

Question 38

Hope Creek RO Exam - Nov 2005

☑ **RO** ☐ **SRO**

Tier # 2 **Group #** 1

HC Obj: NOH01LPRM00-01
Obj 2

Importance 3.2

215005 APRM / LPRM

G2.1.28 Knowledge of the purposes and function of major system components and controls (CFR: 41.7)

Question

With the plant at 100% power, APRM A is indicating 99% and has the following LPRM input signals:

- 5 LPRMs reading between 95 and 100
- 7 LPRMs reading between 80 and 95
- 5 LPRMs reading between 50 and 80
- 4 LPRMs reading between 35 and 50

If the HIGHEST reading LPRM is BYPASSED, the "A" APRM output is ___________

and the difference between the "A" APRM indicated power and the calculated (heat balance) core thermal power is _______________

A higher
higher

B lower
lower

C lower
higher

D higher
lower

Answer C

References
INPO Question 24521
NOH01APRM00-01, AVERAGE POWER RANGE MONITORING (APRM) SYSTEM, p. 10
NOH01LPRM00-01, LOCAL POWER RANGE MONITORING (LPRM) SYSTEM

Justification

References during Exam None

Question is asking function of the averaging amplifier in the APRM circuit.
A - INCORRECT - By removing the highest, the average of all remaining LPRMs will be lower.
B - INCORRECT - If APRM output lowers, since initial power given is 100% the absolute difference must rise.
C - CORRECT - APRM output will lower and absolute difference will be higher.
D - INCORRECT - While the averaging amplifier will adjust for removing LPRM input, it continues to average remaining LPRMs which will mathematically be a lower value.

Question Source Mod ☐ **Memory Level** ☑ **Comprehension Level**

Question History:

New 9/7
SXD - OK
AF - changed number of LPRM strings
MB - made changes as requested
MB - confusing wording, changed to fill in the blank 11/8

Question	39

☑ RO
☐ SRO

Tier # 2 **Group #** 1

Importance 3.5

Hope Creek RO Exam - Nov 2005

HC Obj: RCIC00E012

217000 RCIC

K1.01 Knowledge of the physical connections and/or cause-effect relationships between RCIC and the following — Condensate storage and transfer system

Question

Given the following

- Hope Creek is operating at 100% power.
- The RCIC system is in standby with a suction from the CST.
- The Quarterly HPCI flow rate test is in progress and taking longer than expected.

Then, Suppression Pool High Level alarm has just been received.

What is the expected response of RCIC to the Suppression Pool High Level alarm?

A NO effect since RCIC suction valves do NOT transfer on High Suppression Pool Level

B RCIC Suppression Pool Suction Valve HV-F031 receives an OPEN signal, when it gets FULL OPEN, the RCIC CST Suction Valve HV-F010 will go CLOSED

C RCIC Suppresson Pool Suction Valve HV-F031 receives an OPEN signal AND RCIC CST Suction Valve HV-F010 receives a CLOSED signal.

D RCIC Suppression Pool Suction Valve HV-F031 receives an OPEN signal, RCIC CST suction valve HV-F010 does NOT receive any signal.

Answer A

References NOH01RCIC00-02, REACTOR CORE ISOLATION COOLING SYSTEM, p46
Brunswick Exam 2003, Q27 modified

Justification

References during Exam None

A - CORRECT - Per lesson plan neither HV-F010 or HV-F031 receive a signal on High Suppression Pool level
B - INCORRECT - HV-F031 does NOT receive an OPEN signal on High Suppression Pool level
C - INCORRECT - see "B" above
D - INCORRECT - see "B" above

Question Source Mod ☑ **Memory Level** ☐ **Comprehension Level**

Question History:

9/7 - New - Had to re-sample initial question concerned RCIC/RHR interconnect which has been removed.
SXD - K/A mismatch
MB - wrote new question
SXD -OK
AF - Minor changes, re-word search looking for NO and NOT, replace * bullet with - bullet
MB - Made changes as requested
MB - 11/8 - OK

Question	40

Hope Creek RO Exam - Nov 2005

☑ RO
☐ SRO

Tier #	2	Group #	1

Importance 3.2

HC Obj: ADSSYSE004

218000 ADS

G2.1.28 Knowledge of the purpose and function of major system components and controls.

Question

With all systems operable and the station at 100% power, a seismic event causes a blackout condition (loss of offsite power and all EDGs fail to close on their respective busses) and a small break LOCA.

Conditions are as follows after the seismic event:

- Drywell pressure 1.09 psig and stable
- Reactor level is lowering slowly and just crossing minus 129" (-129") now (T=0)

Based on this information, which of the following is true?

A 105 seconds from T=0 the ADS valves will automatically open.

B 300 seconds from T=0 the ADS valves will automatically open.

C 405 seconds from T=0 the ADS valves will automatically open.

D The ADS valves will NOT automatically open unless conditions change.

Answer D

References Hope Creek Question - Q56457 - modified
HC.OP-SO.SN-0001 section 3.3.1

Justification

References during Exam None

Per HC.OP-SO.SN-0001 section 3.3.1
A - INCORRECT - NO RHR or Core Spray pumps will be running, ADS will NOT initiate
B - INCORRECT - NO RHR or Core Spray pumps will be running, ADS will NOT initiate
C - INCORRECT - NO RHR or Core Spray pumps will be running, ADS will NOT initiate
D- CORRECT - until power is restored to either the RHR or Core Spray pumps, ADS will NOT initiate.

Question Source Mod ☐ Memory Level ☑ Comprehension Level

Question History:

Modified Hope Creek Question Q56457 on 9/20
SXD - Minor comments
MB - 9/27 - Made changes as requested
AF - K/A mismatch
MB - minor editorial changes

Question 41

Hope Creek RO Exam - Nov 2005

☑ **RO** ☐ **SRO**

Tier # 2 **Group #** 1

Importance 3.9

HC Obj:

223002 PCIS/Nuclear Steam Supply Shutoff

A4.02 Ability to manually operate and/or monitor in the control room Manually initiate the system (CFR:41.7/45.5 to 45.8)

Question

Select the action(s) that will ONLY close all the NS4 outboard isolation valves other than the MSIVs.

A	"B" and "C" NS4 logic channels are deenergized.
B	"B" NS4 logic manual initiation collar is armed and pushbutton is depressed.
C	"A" and "D" NS4 logic channels are deenergized.
D	"D" NS4 logic manual initiation collar is armed and pushbutton is depressed.

Answer D

References
Hope Creek Question - Q53931
NOH01NSSSS0-00, NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM (NSSSS) - p.10, p.13
Training Print 045-01: Nuclear Steam Supply Shutoff System

Justification

References during Exam None

IAW B21-1090-0062 and HC.OP-SO.SM-0001 -
A - INCORRECT - this will cause a full group one [MSIV] isolation [e.g. MSIV's will close]
B - INCORRECT - this will cause NO isolation
C - INCORRECT - this will cause a full NS4 isolation and the MSIV's will close
D - CORRECT - "D" NSSSS logic manual initiation collar is armed and push-button is depressed.-Correct

Question Source Bank ☑ **Memory Level** ☐ **Comprehension Level**

Question History:

SXD review 7/21 - Minor editorial change - LOD 1.5 - evaluate making question more difficult
AF 8/23 - 2 correct answers C and D, added ONLY to stem.
MB - 8/24 - Made changes as requested
AF - OK
MB - OK 11/8

Question 42

Hope Creek RO Exam - Nov 2005

☑ **RO**
☐ **SRO**

Tier # 2 **Group #** 1

Importance 3.9

HC Obj:

239002 SRVs

A4.06 Ability to manually operate and/or monitor in the control room Reactor water level (CFR: 41.7/45.5 to 45.8)

Question

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

- Reactor water level is 35 inches
- An SRV inadvertently opens

With NO operator action, which one of the following describes Reactor Water level response?

Reactor Water level will:

A lower and then return to 35 inches

B lower and remain below 35 inches

C rise and then return to 35 inches

D rise and remain above 35 inches

Answer C

References Hope Creek Question ID - 22077
NOH01FWCONTC-02, FEEDWATER CONTROL SYSTEM, p.11

Justification

References during Exam None

A - INCORRECT - lower and then return to 35 inches (see answer C)
B - INCORRECT - lower and remain below 35 inches (see answer C)
C - CORRECT - rise and then return to 35 inches. RPV Swells up on the RPV pressure reduction when the SRV initially opens. RPV level returns to 35 inches due to DFCS setpoint of 35 inches.
D - INCORRECT - rise and remain above 35 inches

Question Source Bank ☐ **Memory Level** ☑ **Comprehension Level**

Question History:

SXD review 7/21 - Minor Editorial changes
AF - 8/23 OK
MB - 11/8 - OK

Question 43

☑ **RO**

Tier # 2 **Group #** 1

HC Obj: FWCONTE011
FWCONTE011

FWCONTE011

FWCONTE011

☐ **SRO** **Importance** 2.8

259002 Reactor Water Level Control

K3.06 Knowledge of the effect that a loss or malfunction of the Reactor Water Level Control will have on the following Main Turbine (CFR:41.7/45.6)

Question

The plant is operating at 70% reactor power with 2 Reactor Feed Pumps (RFPs) running in automatic with the Master Level PDS level set at 35 inches.

- A Narrow Range level is reading 36"
- B Narrow Range level is reading 35" and is the Median Controlling channel
- C Narrow Range level is reading 34"

When B Narrow Range level fails to 33"

Assuming NO operator action, which of the following describes the initial plant response?

A RCIC initiates because reactor water level lowers to Level 2.

B Actual Reactor water level will remain at 35 inches.

C Main Turbine trips because level reaches Level 8.

D Actual water level rises 1 inch.

Answer D **References** INPO Question 20357
NOH01FWCONTC-02, FEEDWATER CONTROL SYSTEM, p11

Justification **References during Exam** None

A - INCORRECT - with Indicated level < programmed level, actual level will rise
B - INCORRECT - with indicated level < programmed level, actual level will rise
C - INCORRECT - Level does NOT rise to Level 8
D - CORRECT - When B fails it is NO longer the Median value, B Narrow range channel will NO longer be Median Select, that will transfer to "C" because it's setpoint is 1" low actual level will rise one inch until "C" Channel is reading 35"

Question Source Mod ☐ **Memory Level** ☑ **Comprehension Level**

Question History:

New 9/7
SXD - add more justification
MB - added more justification
SXD -OK
AF - had me change correct answer to more correct answer.
MB - made changes as requested
MB - 11/8 - minor editorial changes

Question 44		Hope Creek RO Exam - Nov 2005
☑ RO ☐ SRO	Tier # 2 Group # 1	HC Obj: NOH01RBVENT-01 Obj. 35
	Importance 3.6	

261000 SGTS

K3.02 Knowledge of the effect that a loss or malfunction of the SGTS will have on the following — Off-site release rate (CFR: 41.7/45.6)

Question

Given the following conditions:

- Identified leak rate has increased
- Drywell pressure is 1.1 psig and rising slowly
- Reactor Building Exhaust Radiation is reading 1 E-4 microcuries/cc and rising

The CRS has directed venting of the drywell be performed per HC.OP-SO.GS-0001, CONTAINMENT ATMOSPHERE CONTROL SYSTEM OPERATION.

The RO opens the following valves to start venting the Drywell:

- HD-9372A, DRYWELL PURGE DRYWELL VENT EXH DAMPER
- HV-4951, PRI CNTMT VENT TO CPCS BYPASS
- HV-4952, PRI CNTMT TO CPCS INBD ISLN DMPR

Drywell pressure is lowering and off-site release rate rising when the Reactor Building Ventilation Exhaust Hi-Hi radiation alarm is received.

The RO reports that HV-4951 has failed OPEN and CANNOT be CLOSED. All other valve/dampers responded as expected.

Based on these conditions what is the expected condition of:

I - HD-9372A, DRYWELL PURGE DRYWELL VENT EXH DAMPER is _______
II - HV-4952, PRI CNTMT TO CPCS INBD ISLN DMPR is _______
III - Off-site release rate is ________

A
I - CLOSED
II -CLOSED
III - lowering

B
I - OPEN
II -CLOSED
III - lowering

C
I - CLOSED
II -OPEN
III - lowering

D
I - OPEN
II -OPEN
III - rising

Answer A

References HC.OP-SO.GS-0001(Q), CONTAINMENT ATMOSPHERE CONTROL SYSTEM OPERATION, p. 44
NOH01RBVENT-01, REACTOR BUILDING VENTILATION, p. 49

Justification

References during Exam M-57 sht 1, M-76

A - CORRECT - Both HV-4952 and HV-9272A will close on a Hi Hi Radiation Signal isolating the release.
B - INCORRECT - HD-9372A does NOT remain open on a Hi Hi Radiation Signal it recloses
C - INCORRECT - HV-4952 does not remains Open on a Hi Hi Radiation Signal
D - INCORRECT - see B above.

Question Source New ☐ Memory Level ☑ Comprehension Level

Question History:

New 9/21

-- Archie to verify that HV-4952 remains OPEN if it had been over-ridden open.
SXD - OK (Archie to look at)
AF - K/A mismatch, NOT touching K/A, try to chop out extra words
MB- added a failure of HV-4951 in and changed answer to D
AF - OK
MB - 11/8 - Resample
MB - 11/10 - re-wrote question based on feedback from RJC

Question 45

Hope Creek RO Exam - Nov 2005

☑ **RO** ☐ **SRO**

Tier # 2 **Group #** 1

Importance 3.1

HC Obj:

262001 AC Electrical Distribution

K4.03 Knowledge of AC Electrical distribution design feature(s) and or interlock(s) which provide for the following — Interlocks between automatic bus transfer and breakers (CFR:41.7)

Question

With the plant in a normal electrical lineup for 100% power, the TRIP pushbutton is pressed for breaker 52-40201, Normal Feed Breaker for 10A402 on Control Room panel 10C651E.

Which choice below describes the response of the 10A402 Bus and "B" EDG?

A The Alternate Feed Breaker, 52-40208 will close energizing Bus 10A402."B" EDG will NOT be running.

B Bus 10A402 will be de-energized. The "B" EDG will NOT be running.

C Bus 10A402 will be de-energized. The "B" EDG will be running with its output breaker open.

D The "B" EDG will be running and its output breaker will close energizing Bus 10A402.

Answer B

References Hope Creek Question - Q53557,
NOH01EAC00-02, CLASS 1E AC POWER DISTRIBUTION, p.27

Justification

References during Exam None

CORRECT - Bus 10A402 will be de-energized. The "B" EDG will NOT be running. The automatic transfer to the alternate feed and the start of the Diesel will NOT occur if the normal breaker is manually tripped.
INCORRECT - The Alternate Feed Breaker, 52-40208 will close energizing Bus 10A402."B" EDG Lockout will prevent the EDG start and output breaker closure. The automatic transfer to the alternate feed will NOT occur if the normal breaker is manually tripped.
INCORRECT - Bus 10A402 will be de-energized.The "B" EDG will be running with its output breaker open. The automatic start of the Diesel will NOT occur if the normal breaker is manually tripped.
INCORRECT - The "B" EDG will start and its output breaker will close energizing Bus 10A402.The automatic start of the Diesel will NOT occur if the normal breaker is manually opened

Question Source Bank

☑ **Memory Level** ☐ **Comprehension Level**

Question History:

SXD review 7/21 - minor editorial changes
AF - 8/23 made NOT all caps in "A"
MB - 8/24 - Made changes as requested
AF - minor changes, K/A mismatch
MB - made changes as requested, SXD to resolve K/A
MB - 11/8 - OK

Question 46

☑ **RO**
☐ **SRO**

Tier # 2 **Group #** 1

Importance 2.8

HC Obj:

262002 UPS (AC/DC)

K6.02 Knowledge of the effect that a loss or malfunction of the following will have on the UPS (AC/DC) DC electrical power (CFR:41.7/45.7)

Question

Hope Creek is at 100% power with the following lineup on 120V Class 1E Cyberex 20KVA Inverter 1AD481:

- CB-20 - 125V DC Power Breaker Closed
- CB-201 - 480V AC Normal Power Breaker Closed
- CB-301 - 480V AC Backup Power Breaker Open
- Auctioneered Bypass Switch is in the BYPASS D1 Position
- Manual Bypass Switch is in the NORMAL Position

An Operator inadvertently opens the CB-21 (Battery Output from Auctioneered Circuit).

What effect will that have on Class 1E Instrument Distribution Panel 1AJ481?

Class 1E Panel 1AJ481 will be …

A de-energized due to Auctioneered Bypass Switch being in the BYPASS D1 Position.

B energized from 480V AC Backup Power.

C energized from 480V AC Normal Power.

D de-energized due to CB-301 - 480V AC Backup Power Breaker being Open.

Answer C

References NOH01EAC00-02, CLASS 1E AC POWER DISTRIBUTION, p. 60-62

Justification

References during Exam Figures 5, 6 and 8 of NOH01EAC00-02, AV2114D.vsd and AV2114F.vsd, AV-2114C

A - INCORRECT - Auctioneer Bypass - Allows bypassing of one of the two Auctioneer Diodes (either diode can perform the design function) since either diode can perform the design function, bypassing diode 1 will have NO EFFECT.
B. - INCORRECT -Breaker CB-301 is given as OPEN and there are NO auto closures for this breaker.
C. - CORRECT - Power is normally supplied to 120V AC Distribution Panels from the Normal AC Power source -> Rectified to DC and then inverted back to AC. Since backup DC Power is lost, normal AC Power will still be available and the Distribution Panel will be powered as it normally is.
D. - INCORRECT - Panel 1AJ481 is NOT de-energized.

Question Source New ☐ **Memory Level** ☑ **Comprehension Level**

Question History:

SXD review - 7/22 - OK
AF - bullets, added figure 5 - have SD to look at it.
SXD - Get him a copy of figure
SXD - No figure due to direct lookup
AF - minor changes
MB - Made changes as requested
MB - 11/8 - minor change to 4th stem

Question 47

Hope Creek RO Exam - Nov 2005

☑ **RO** ☐ **SRO**

Tier # 2 **Group #** 1

Importance 2.5

HC Obj:

263000 DC Electrical Distribution

A1.01 Ability to predict and/or monitor changes in parameters associated with operating the DC Electrical distribution controls including — Battery charging/discharging rate (CFR:41.5/45.5)

Question

Control Room annunciator D3-F2 "125VDC SYSTEM TROUBLE" is alarming. Upon investigation the Operator determines that Digital Point D4631 "125VDC BATTERY CHARGER 1AD413" is in alarm and Battery Charger 1AD414 is out of service. On panel 10C650 the Operator reports the following:

125VDC Switchgear 10D410:
- Bus Voltage is reading 125 VDC
- Bus Current is reading 220 Amps

The following is indicated on the 125VDC Battery Charger, 1AD413, control panel:

- DC Voltmeter is reading 125 VDC
- DC Ammeter is reading 200 Amps
- Timer switch is at 0
- AC PWR ON light is lit
- DC Under Voltage light if off
- DC Over Voltage light is off
- Hi Voltage Shutdown light is off
- Insufficient Charging Current light is OFF

WITH NO OPERATOR ACTION, which one of the following describes the expected 10D410 bus voltage trend and the reason for that trend?

The bus voltage will . . .

A lower because the bus load exceeds the charger's capacity.

B rise because an equalizing charge is being provided.

C rise because a malfunction of the float charge is indicated.

D lower because AC power is NOT being supplied to the charger.

Answer A

References
INPO Question 24538
NOH01DCELEC-00, DC ELECTRICAL DISTRIBUTION, p25-26, p.19-20

Justification

References during Exam None

A - CORRECT - with Switchgear Load > Charger Output voltage will lower over time
B - INCORRECT - Equalizing Charge is NOT being provided with Timer switch at 0.
C - INCORRECT - Float charge is malfunctioning because charge voltage should be > bus voltage, however this will cause voltage to lower, NOT rise over time.
D - INCORRECT - 2 AC on lights indicate charger has AC power

Question Source Mod ☐ **Memory Level** ☑ **Comprehension Level**

Question History:

SXD review - 7/22 - OK
AF - changed Bus current to 220 amps and DC ammeter to 200 amps, deleted Equalizing light is off, and changed Insufficient charging current light to OFF.
MB - 8/24 - Made changes as requested
AF - minor changes
MB - Made changes as requested
MB - 11/8 - changed INOP to out of service
MB - 11/17 - Archie to check on float
MB - 11/18 - No float light

Question 48

- [x] **RO**
- [ ] **SRO**

Tier # 2 **Group #** 1

Importance 2.9

HC Obj: EDG000E029

264000 EDGs

A2.04 Ability to (a) predict the impacts of the following on the EMERGENCY GENERATORS (DIESEL/JET) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:
(CFR: 41.5 / 45.6)

Consequences of operating under/over excited

Question

Diesel Generator "A" 1AG400 has just been synchronized with the Class 1E bus 10A401 resulting in the following generator indications:

- 60.0 Hz
- 200 KW
- 200 KVAR
- 4.280 KV

Which one of the following actions are REQUIRED in accordance with HC.OP-SO.KJ-0001, EMERGENCY DIESEL GENERATORS OPERATION, to restored generator parameters within acceptable limits and the reason for this action?

A Lower reactive load using the GOVERNOR DECREASE PB to prevent generator overvoltage.

B Lower reactive load using the VOLTAGE CONTROL LOWER PB to prevent generator winding overheating.

C Raise real load using the VOLTAGE CONTROL RAISE PB to prevent generator winding overheating.

D Raise real load using the GOVERNOR INCREASE PB to prevent reverse power.

Answer D

References HC.OP-SO.KJ-0001, EMERGENCY DIESEL GENERATORS OPERATION, p. 33 and 34
NOH01EDG000-02, EMERGENCY DIESEL GENERATORS (EDG)
Limerick 2005 Exam Question 11

Justification

References during Exam None

A - INCORRECT - with generator sync'd to grid, GOV PB changes Real Load NOT reactive load
B - INCORRECT - Local control procedure has you ADJUST KiloVar loading to approx. 100 to 500 KvARs using VOLTAGE CONTROL RAISE/LOWER Control Handle, since KVAR loading is already 200 KVARs, this does NOT need to be done.
C - INCORRECT - with the generator sync'd to the grid, VOLTAGE CONTROL RAISE/LOWER Control Handle will change Reactive load, NOT real load.
D - CORRECT per procedure precaution 3.1.3.

Question Source Mod

- [ ] **Memory Level**
- [x] **Comprehension Level**

Question History:

New 9/20
SXD - Check location
MB - Verified location and terminology taken directly from Procedure
SXD -OK
AF - Changed to Control Room vs. Remote shutdown
MB - Made changes as requested
MB - minor change 4.2KV

Question 49

Hope Creek RO Exam - Nov 2005

☑ **RO** ☐ **SRO**

Tier # 2 **Group #** 1

HC Obj: NOH01INSAIR-01 - Obj R7

Importance 2.9

300000 Instrument Air

A3.02 Ability to monitor automatic operations of the Instrument Air including Air temperature (CFR 41.7/45.5)

Question

Hope Creek is at 100%.

Instrument Air status is as follows:

- 00K107, Service Air Compressor - Disassembled for Compressor work
- 10K107, Service Air Compressor - Tripped due to Low Lube Oil Pressure - currently being investigated
- 10K100, Emergency Instrument Air Compressor - Running
- Instrument Air Pressure - 90 psig stable

A SACS/TACS AUTO ISOLATION alarm is received on low pressure.

The Operators take the Mode Switch to shutdown and stablize the plant at a Reactor level of +35" (lowest level = +10").

Assuming NO operator actions are taken and Instrument Air loads after the trip equal Instrument Air loads before the trip, what effect will this have on the Instrument Air system.

A It will have NO effect on the Instrument Air System, instrument air pressure shall be ~ equal to pre-trip value.

B Discharge air temperature will increase until the Air Compressor trips on Discharge Air Temperature high, instrument air pressure will be lower than pre-trip value.

C Cooling water supply flow will decrease until the Air Compressor trips on Low Cooling Water Supply pressure, instrument air pressure will be lower than pre-trip value.

D Reactor water level dropping to 10" causes the Air Compressor to trip on Low RPV Level, instrument air pressure will be lower than pre-trip value.

Answer A

References NOH01INSAIR-01, INSTRUMENT AIR SYSTEM. P.13-14

Justification

References during Exam None

A - CORRECT - Since EIAC is running and it is cooled by RACS and trips on low RPV level of -38", a loss of TACS should have NO effect on EAIC and instrument air pressure should remain constant.
B - INCORRECT - EIAC is cooled by TACS, plausible distractor, if candidate thinks cooling water is isolated to compressor, discharge air temperature would increase and may cause compressor trip.
C - INCORRECT - EIAC is cooled by TACS, plausible distractor, if candidate thinks cooling water is isolated to compressor, cooling water supply flow would decrease and may cause compressor trip.
D - INCORRECT - RPV level must drop to -38" to cause EIAC to trip.

Question Source New ☐ **Memory Level** ☑ **Comprehension Level**

Question History:

SXD Review 7/21 - LOD - 1.0 - re-write to make more difficult
8/4 - Re-wrote question.
AF - another Instrument air question similar to number 8. Weak K/A match. Look at changing question to better match K/A
MB - I think K/A is ok based on the fact that it is testing student's knowledge of what is cooling the EIAC, if question stem was changed from a loss of TACS to a loss of RACS, Instrument air temp would increase and air compressor could trip on high instrument air temp.
SXD - to resolve
SXD -OK
AF - bullets
MB - Made changes as requested
MB - OK 11/8

Question	50

- [x] RO
- [ ] SRO

Tier # 2 **Group #** 1

Importance 2.6

HC Obj: MNPWR0E014

262001 A.C. Electrical Distribution

K5.02 Knowledge of the operational implications of the following concepts as they apply to A.C. ELECTRICAL DISTRIBUTION: (CFR: 41.5 / 45.3) Breaker control

Question

What function does opening the toggle switch associated with the 500KV breaker Emergency Trip handle accomplish?

A Enables removal of kirk key at the breaker.

B Removes control power for the breaker.

C Transfers control to the blockhouse.

D Activates the breaker counter.

Answer B

References Hope Creek Question - Q68164
NOH01MNPWR0-04, MAIN POWER SYSTEM, p21

Justification

References during Exam None

A - INCORRECT - Kirk Key can be removed with toggle in either position.
B - CORRECT - Interrupts breaker control circuit to prevent electrical operation.
C - INCORRECT - No local control at the breaker except for test.
D - INCORRECT - The breaker counter is mechanical.

Question Source Bank

- [x] Memory Level
- [ ] Comprehension Level

Question History:

SXD review 7/21 - OK
MB - 7/28 - Need to add references
AF - 8/23 - 3rd Instrument air question. Double jeopardy with number 8 and 49. Perhaps throw out and get a different K/A. Maybe change out number 8.
MB - Revisit 8, 49, 50
SXD - OK Questions
MB - 9/27 - Re-sampled due to AF comments, New Question
SXD -OK
AF - OK
MB - changed operating to opening in stem 11/8

Question	51

- [x] RO
- [ ] SRO

Tier # 2 **Group #** 1

Importance 2.7

HC Obj: RACS00E005

400000 Component Cooling Water

K6.01 Knowledge of the effect that a loss or malfunction of the following will have on the Component Cooling Water Valves (CFR:41.5/45.5)

Question

With the plant operating at 100% power, 1-ED-TE-2617, RACS HX Outlet temperature fails downscale.

How will the following be affected?

I - 1-ED-TV-2617, "RACS HX 1AE/BE 217 BYPASS" valve

II - HV-2537A, "HX AE 217 INLET" valve

III - Actual RACS temperature

A
I - goes closed
II - fails open
III - goes down

B
I - goes open
II - fails closed
III - goes up

C
I - goes closed
II - unaffected
III - goes down

D
I - goes open
II - unaffected
III - goes up

Answer D

References NOHO1RACS00C-02, Reactor Auxiliary Cooling System - p.15

Justification

References during Exam None

A - INCORRECT - TV-2617 goes full open on a failure of it's input signal, causing RACS temperature to increase
B - INCORRECT - HV-2537A does NOT get a temperature signal and therefore will remain as is.
C - INCORRECT - TV-2617 fails open on a failure of it's input signal causing RACS temperature to increase
D - CORRECT - TV-2617 will Open, HV-2537A remains unaffected and RACS temperature will increase because less flow is going through the HX.

Question Source New

- [ ] Memory Level
- [x] Comprehension Level

Question History:

8/24 - new
SXD - OK
AF- changed fails open to goes open
MB - made changes as requested
MB - OK 11/8

Question 52

☑ **RO**
☐ **SRO**

Tier # 2 **Group #** 1

HC Obj: SRMSYSE006

Importance 3.4

215004 Source Range Monitor

A1.03 Ability to predict and/or monitor changes in parameters associated with operating the SOURCE RANGE MONITOR (SRM) SYSTEM controls including RPS status

Question

Which set of conditions within the Source Range Monitoring System, would generate a Reactor Protection system SCRAM signal:

A Shorting links - REMOVED
short period trip in 2 channels

B Shorting links - REMOVED
upscale trip in 1 channel

C Shorting links - INSTALLED
upscale trip in 2 channels

D Shorting links - INSTALLED
short period trip in 1 channel

Answer B

References NOH01SRMSYS-01, Source Range Monitoring (SRM) System - p. 24
INPO Question - 20334

Justification

References during Exam None

A - INCORRECT - short period trip will NOT generate a RPS scram signal - only get ALARM
B - CORRECT - Upscale trip on 1 channel with the shorting links removed will generate a SCRAM
C - INCORRECT - with shortling links installed a trip will NOT be generated on Upscale trip
D - INCORRECT - never get a scram on short period trip, only get ALARM

Question Source Bank ☑ **Memory Level** ☐ **Comprehension Level**

Question History:

8/24 - New
SXD - OK
AF - OK
MB - OK 11/8

Question 53

☑ **RO**
☐ **SRO**

Tier # 2 **Group #** 1

Importance 3.3

HC Obj:

223002 PCIS/Nuclear Steam Supply Shutoff

K6.04 Knowledge of the effect that a loss or malfunction of the following will have on the PRIMARY CONTAINMENT ISOLATION SYSTEM/ NUCLEAR STEAM SUPPLY SHUT-OFF : Nuclear boiler instrumentation (CFR: 41.7 / 45.7)

Question

While operating RHR in shutdown cooling, reactor water level transmitter LT-N080A fails downscale.

SELECT the response of the RHR shutdown cooling supply valves, HV-F008 and HV-F009.

A Both RHR shutdown cooling supply valves will automatically close.

B Only one of the RHR shutdown cooling supply valves automatically close and the second RHR shutdown cooling supply valve will close if low level is sensed by LT-N080B.

C Only one of the RHR shutdown cooling supply valves automatically close and the second RHR shutdown cooling supply valve will close if LT-N080C fails downscale.

D Neither RHR shutdown cooling supply valve will change position automatically.

Answer D

References Hope Creek Question - Q53932
NOH01RHRSYSC-03, RESIDUAL HEAT REMOVAL SYSTEM, P.30

Justification

References during Exam None

- - Both RHR shutdown cooling supply valves will automatically close. -Incorrect - the trip must occur in both channels "a" and "b"/"c" and "d" to cause any isolation
- - Neither RHR shutdown cooling supply valve will change position automatically. Correct - the trip must occur in both channel "A" and "B" to cause an isolation
- - Only one of the RHR shutdown cooling supply valves automatically close and the second RHR shutdown cooling supply valve will close if Level 3 is sensed in the "B" NSSSS logic. -Incorrect - the trip must occur in both channels to cause any isolation. Only one would close and only when the second signal is received.
- - Only one of the RHR shutdown cooling supply valves automatically close and the second RHR shutdown cooling supply valve will close if Level 3 is sensed in the "C" NSSSS logic. -Incorrect - the trip must occur in both channels to cause any isolation

Question Source Bank ☑ **Memory Level** ☐ **Comprehension Level**

Question History:

SXD review 7/21 - OK
AF - OK
MB - 11/8 - OK

Question **54**			**Hope Creek RO Exam - Nov 2005**
☑ ***RO*** ☐ ***SRO***	***Tier #*** 2 ***Group #*** 2	***Importance*** 2.9	***HC Obj:***

201006 RWM

K6.03 Knowledge of the effect that a loss or malfunction of the following will have on the RWM — Rod Position indication

Question

There is a Control Rod with an inoperable notch position reed switch. When looking at the Rod Worth Minimizer display screen for that rod, how would it's position be indicated?

A RWM would display a suggested substitute position.

B RWM would display a default value of "- -"

C RWM would display the last known good position.

D RWM would display a default value of "00"

Answer A

References INPO Question 1885
NOH01RODMIN-01, ROD WORTH MINIMIZER p.15

Justification

References during Exam None

A - CORRECT - per Lesson Plan - If a control rod is moved to a position with a failed reed switch, the RWM program will:a)Allow a single notch insert or withdraw permissive to allow the control rod to be moved to verify its actual position. b)Suggest to the operator a substitute position, which is its calculated inferred position.
B - INCORRECT - See "A"
C - INCORRECT - See "A"
D - INCORRECT - See "A"

Question Source Mod ☑ ***Memory Level*** ☐ ***Comprehension Level***

Question History:

SXD review - 7/22 - Had questions talk to Archie about what would be displayed. Perhaps change inop notch position to a given position (ie. 12). If you pull rod from 10 to 12 and position 12's reed switch is INOP is 12 displayed.
AF - OK
MB - make double dash bigger.

Question 55		Hope Creek RO Exam - Nov 2005
☑ RO ☐ SRO	Tier # 2 Group # 2 Importance 3.3	HC Obj: RECCONE015

202002 Recirculation Flow Control

A2.07 Ability to (a) predict the impacts of the following on the Recirculation flow control and (b) based on those predications, use procedures to correct, control, or mitigate the conseqences of those abnormal operation — Loss of feedwater signal inputs

Question

Given the following conditions:

- Unit startup is in progress to 100%
- Reactor power is 90%
- Reactor water level is 35"
- FW control is in 3 Element control
- 3 Primary condensate pumps are running
- 3 Secondary condensate pumps are running
- 3 RFP's are running
- A Loop Feed flow indicates - 6.1 E6 lbs/hr
- B Loop Feed flow indicates - 6.1 E6 lbs/hr
- Both Recirc pumps are running in Master Manual control with recirc pump speed and total core flow at ~90%

An event occurs causing "A" Loop Feed flow to fail downscale.

What effect if any will this feedwater signal failure have on the Recirculation Flow Control circuit?

A NO effect

B Both Recirc pumps scoop tubes will lockup at their current position

C Both Recirc pumps will runback to their Speed Limit #2 (45%) speed and stablize there.

D Both Recirc pumps will runback to their Speed Limit #1 (30%) speed and stablize there.

Answer A

References HC.OP-IO.ZZ-0003, Startup from Cold Shutdown to Rated Power - p.42-45
NOH01RECCON-02, Reactor Recirculation Flow Control System - p.30-32

Justification

References during Exam Power to Flow Map

A - CORRECT - NO effect. Loop feed flow does not cause any type of recirc pump runback.
B - INCORRECT - Scoop tube lockup would have occurred if problem had been a recirc pump flow indication vs. Loop flow.
C - INCORRECT - no speed limiter signals were generated.
D - INCORRECT - no speed limiter signals were generated.

Question Source New ☐ Memory Level ☑ Comprehension Level

Question History:

8/24 - New
SXD - OK
RJC - 2 part K/A
MB - Per NUREG 1021 ES-401 page 5-6 "When selecting or writing questions for K/As that test coupled knowledge or abilities (e.g., the A.2 K/A statements in Tiers 1 and 2 and a number of generic K/A statements, such as 2.4.1, in Tier 3), try to test both aspects of the K/A statement. If that is NOT possible without expending an inordinate amount of resources, limit the scope of the question to that aspect of the K/A statement requiring the highest cognitive level (e.g., the (b) portion of the A.2 K/A statements) or substitute another randomly selected K/A."
AF - Flow must be > 75% before pump trip in order to get runback, will be a hard question
MB - changed power to 90%, deleted Condensate pump trips
MB - added "Causing" on stem

Question 56

Hope Creek RO Exam - Nov 2005

☑ **RO**
☐ **SRO**

Tier # 2 **Group #** 2

Importance 3.8

HC Obj:

219000 RHR/LPCI: Torus/Pool Cooling Mode

K4.03 Knowledge of RHR/LPCI Torus/Pool Cooling Mode design feature(s) and or interlocks which provide for the following — Unintentional reduction in vessel injection flow during accident conditions

Question

Given the following plant conditions:

- Drywell pressure	3.2 psig
- Drywell temperature	170°F
- Suppression Pool pressure	1.8 psig
- Suppression Pool temperature	96°F
- Reactor water level	+ 25 inches
- RPV pressure	400 psig

The plant has scrammed on high Drywell pressure and the actions of both Primary Containment Control and RPV Control are being carried out.

The RHR system was in a normal lineup at the beginning of the transient and all automatic actions occurred as designed.

The CRS orders Suppression Pooling Cooling started on the "A" RHR Loop. Which of the following switch manipulations will have to be performed in order to start Suppression Pool Cooling on the "A" RHR Loop IAW HC.OP-SO.BC-0001, RHR System Operation?

A
- AUTO OP OVRD must be pressed on BC-HV-F017A, RHR LOOP A LPCI INJ MOV before valve can be closed.
- Once valve is closed then BC-HV-F024A, RHR LOOP A TEST RET MOV can be opened by depressing it's INCR pushbutton.

B
- BC-HV-F017A, RHR LOOP A LPCI INJ MOV must be closed by depressing it's closed pushbutton.
- Once F017A is closed then BC-HV-F024A, RHR LOOP A TEST RET MOV can be opened by depressing it's INCR pushbutton.

C
- AUTO OP OVRD must be pressed for BC-HV-F017A, RHR LOOP A LPCI INJ MOV prior to depressing it's CLOSED pushbutton.
- Once F017A is closed then AUTO CL OVRD must be pressed for BC-HV-F024A, RHR LOOP A TEST RET MOV prior to depressing it's INCR pushbutton.

D
- AUTO CL OVRD must be pressed on BC-HV-F017A, RHR LOOP A LPCI INJ MOV before valve can be closed.
- Once valve is closed then AUTO OP OVRD must be pressed on BC-HV-F024A, RHR LOOP A TEST RET MOV prior to opening F024A.

Answer C

References INPO Question 2069
HC.OP-SO.BC-0001(Q) - Rev. 40, RESIDUAL HEAT REMOVAL SYSTEM OPERATION, p. 23, Note 5.5.5

Justification

References during Exam None

A - INCORRECT - AUTO CL OVRD must be pressed on F024A before valve can be opened with LPCI initiation signal present.
B. INCORRECT - must depress AUTO OP OVRD for F017A prior to closing F017A with LPCI signal present
C. CORRECT - per Procedure Note 5.5.5 - If a LPCI Initiation signal is present, the AUTO OP OVRD must be pressed on BC-HV-F017A(B) RHR LOOP A(B,C,D) LPCI INJ MOV, before the valve can be closed. The AUTO CL OVRD must be pressed on BC-HV-F024A(B) RHR LOOP A(B) TEST RET MOV, and BC-HV-F017A(B) must be closed before BC-HV-F024A(B) can be opened.
D. INCORRECT - Must Depress AUTO OP OVRD on F017A NOT AUTO CL OVRD

Question Source Mod ☐ **Memory Level** ☑ **Comprehension Level**

Question History:

SXD Review 7/22 - verify pushbutton labels are correct
AF- funky bullets - added RPV pressure = 400 psig in Stem.
MB - 8/24 - Made changes as requested
MB - 11/8 changed OPEN to INCR

Question 57

Hope Creek RO Exam - Nov 2005

☑ RO

☐ SRO

Tier # 2 **Group #** 2

HC Obj: NOH01MSTEAMC-02 - OBJ R14

Importance 4.2

239001 Main and Reheat Steam

A3.01 Ability to monitor automatic operations of the Main and Reheat system including Isolation of main steam system (CFR:41.7/45.5)

Question

The plant is shutting down for a refueling outage.

Current plant conditions are as follows:
- Mode Switch - STARTUP
- Reactor Power - 4%
- Reactor Pressure - 1000 psig
- Reactor Level - 35"
- Condenser vacuum - 3.5" in HgA
- All MSIV's open

An event occurs:
3 Minutes later plant conditions are as follows:
- Mode Switch - SHUTDOWN
- Reactor Power - All Rods inserted
- Reactor pressure - 700 psig decreasing
- Reactor Level - (-50" lowering)
- Condenser Vacuum - 23" in HgA Degrading

Based on the above conditions and assuming NO operator actions, what is the status of the MSIV's and explain the reason for that status.

A MSIV's all OPEN - NO automatic closure signal exists

B MSIV's all CLOSED - due to 1 Automatic Closure signal - Low Reactor Pressure

C MSIV's all CLOSED - due to 1 Automatic Closure signal - Low Condenser Vacuum

D MSIV's all CLOSED - due to 2 Automatic Closure signals - Low Reactor Pressure and Low Condenser Vacuum

Answer C

References NOH01MSTEAMC-02, MAIN STEAM SYSTEM p.24

Justification

References during Exam TS table 3.3.2

A - INCORRECT - Condenser Vacuum of > 21.5" will cause MSIV's to close. Plausible distractor - this isolation can be bypass with a keylock switch.
B - INCORRECT - Low Reactor Pressure MSIV closure signal is bypassed when Mode Switch is NOT in RUN
C - CORRECT - Low Condenser vacuum setpoint of 21.5" has been reached and limit has NOT been bypassed.
D - INCORRECT - Low Reactor Pressure MSIV closure signal is bypassed when Mode Switch is NOT in RUN

Question Source New ☐ **Memory Level** ☑ **Comprehension Level**

Question History:

SXD Review 7/21 - LOD 1.0 re-write question
8/4 - Wrote new question
AF - lot of bullets, changed vacuum to " HgA and made NO all Caps.
MB - 8/24 - Made changes as requested
AF - asked to delete some bullets
MB - Made changes as requested
MB - 11/8 changed reference to TS table 3.3.2

Question 58			Hope Creek RO Exam - Nov 2005
☑ RO ☐ SRO	Tier # 2	Group # 2	HC Obj: NOH01FWHEAT-00, obj R8 and R9
	Importance 2.5		

245000 — Main Turbine Gen. / Aux.

K1.02 — Knowledge of the physical connections and/or cause effect relationships between Main Turbine Generator / Aux and the following — Condensate system (CFR:41.2 to 41.9 / 45.7 to 45.8)

Question

Hope Creek is operating at 75% power with all controls in automatic when a leak develops in Feedwater Heater 4A causing level to rise.

The Operator observes A7-E2, Feedwater Heater Trip annunciator illuminates and FWH 4A level rises to 30" before stablizing at 30".

Assuming NO operator actions which of the following correctly describes the expected response of Main Turbine Generator MW and Reactor Power to this event:

Main Turbine Generator MW ____I____

Reactor Power _____II________

A I - lowers
II - lowers

B I - lowers
II - rises

C I - rises
II - lowers

D I - rises
II - rises

Answer D

References
HC.OP-AB.BOP-0001, Feedwater Heating, P. 1
NOH01MNCOND-01, CONDENSATE SYSTEM, p.34
NOH01FWHEAT-00, FEEDWATER HEATER EXTRACTION, VENT AND DRAIN SYSTEM, p. 34-37

Justification

References during Exam None

A - INCORRECT - Expected response to this transient is that FW level will rise until the Hi Hi level is reached. At this point all turbine inputs to the FWH are isolated. Since extraction flow being removed from the turbine goes down, MW go up. In addition, FW heating of the 4A FWH goes away, therefore FW temperature goes down. Since FW temperature goes down, but steam pressure remains constant, reactor power rises.
B - INCORRECT - see A
C - INCORRECT - see A
D - CORRECT - see A

Question Source New — ☐ Memory Level — ☑ Comprehension Level

Question History:
New 8/24
SXD - OK
AF - OK
MB 11/8 - changed goes up to rises

Question	59

Hope Creek RO Exam - Nov 2005

☑ RO
☐ SRO

Tier # 2 **Group #** 2

Importance 2.9

HC Obj: RWOVERE008

268000 Radwaste

A2.01 Ability to (a) predict the impacts of the following on the Radwaste and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those abnormal operation — System rupture (CFR:41.5/ 43.5/ 45.3/ 45.13)

Question

Hope Creek is returning to service following a refueling outage.

The plant is currently in OPCON 4, when the Radwaste Operator reports, the Equipment and Floor Drain system needs to be removed from service for 8 hours due to a rupture in the system.

How will the loss of the Equipment and Floor Drain System affect the plant startup?

With the Equipment and Floor Drain System unavailable:

A the plant will be unable to place RHR in suppression pool cooling mode, leading to suppression pool temperature problems.

B the plant will be unable to get rid of excess water in the main condenser caused by reactor heatup, leading to condenser hotwell level problems.

C the plant will be unable to backwash and precoat RWCU filter demins, leading to reactor vessel chemistry problems.

D reactor startup will NOT be affected because the High Conductivity sumps can be realigned to accept water from the Equipment and Floor Drain System.

Answer C **References** NOH01RWOVER-01, RADWASTE SYSTEM OVERVIEW p. 15

Justification **References during Exam** None

A - INCORRECT - Radwaste system is NOT needed to place RHR in suppression pool cooling mode.
B - INCORRECT - water in the main condenser can still be put in the CST and NOT be released as Radwaste.
C - CORRECT - per lesson plan, plant wil be unable to backwash and precoast a RWCU F/D, this could lead to cheminstry problems in the reactor vessel.
D - INCORRECT - High Conductivity sump is NOT connected to the Equipment and Floor Drain System.

Question Source New ☑ **Memory Level** ☐ **Comprehension Level**

Question History:

New 8/24
SXD - Bad distractors
MB - 9/27 - changed distractors
SXD - may need to change 1st distractor, check lesson plan
MB - 10/3 - changed 1st distractor.
SXD - OK
AF - making decisions to change OPCON's is NOT the RO's job, need a better distractor A
MB - 10/25 - changed distractor A
MB - resample
MB - 11/10 - after speaking with RJC re-wrote question.
MB - 11/17 changed reactor startup to plant startup

Question	60		Hope Creek RO Exam - Nov 2005
☑ RO ☐ SRO	Tier # 2 Group # 2	Importance 3.2	HC Obj:

272000 Radiation Monitoring

K5.01 Knowledge of the operational implications of the following concepts as they apply to the Radiation Monitoring — Hydrogen injection operation's effect on process radiation indications

Question

The plant was operating at full power with indicated H2 injection flow at 10 SCFM, when FE-104 (Flow Input to Hydrogen Flow Controller - FIC-601) fails LOW (ie. A LOW flow is INPUT into FIC-601).

Which of the following describes the expected result?

The Hydrogen Flow Control Valves will ____(I)_____

Main Steam Line Radiation Levels will _____(II)_____

A I - open
II - rise

B I - open
II - lower

C I - close
II - rise

D I - close
II - lower

Answer A

References INPO Question 8753
NOH01HWC100-01, HYDROGEN WATER CHEMISTRY INJECTION SYSTEM, p. 12
M-101-0 sht 1 & 2

Justification

References during Exam M-101 sheet 1

A - CORRECT - FIC-601 attempts to maintain a certain H2 flow to the Secondary Condensate Pumps, when this flow input fails LOW - FIC-601 will attempt to raise H2 flow by opening the H2 Flow Control valves, opening these valves will result in Rising Main Steam Line Radiation Levels.
B - INCORRECT - H2 FCV's will open
C - INCORRECT - While the FCV's will open rapidly, there is NO Low Recirc Dissolved Oxygen Level alarm.
D - INCORRECT - FCV's will open.

Question Source Mod ☐ Memory Level ☑ Comprehension Level

Question History:

SXD review - 7/27 - Minor editorial changes
AF - question asking for what happens on a local Chemistry panel. Changed answer key. To rise, lower
MB - 8/24 - Made changes as requested
MB - 11/8 added reference to be given to students
MB - 11/17 - changed 601 to 104

Question	61

- [x] RO
- [ ] SRO

Tier # 2 **Group #** 2

Importance 2.7

HC Obj: NOH01DWVENT-02

223001 Primary Containment System and Auxiliaries

K2.09 Knowledge of electrical power supplies to the following: (CFR: 41.7) Drywell cooling fans: Plant-Specific

Question

Hope Creek is at OPCON 4 with the Drywell Cooling Fans aligned as follows:

- A1 fan in AUTO - NOT Running
- A2 fan in MAN - Running in HIGH
- B1 fan in MAN - NOT Running
- B2 fan in MAN - NOT Running
- The Alternate Incoming Feeder Breaker to 10A401 is tagged out for Maintenance
- All of the remaining fans are tagged out for Maintenance

I&C was working on the 4.16KV bus 10A401 and caused the normal feeder breaker to trip open and a LOP signal to be sent to the "A" EDG. The "A" EDG came up to speed and restored power to the bus.

Assuming NO operator actions which of the following describes the DW Cooling Fans status following the transient:

A Running Fans - A1, B1, A2
NOT Running Fans - B2

B Running Fans - A1, A2
NOT Running Fans - B1, B2

C Running Fans - A2
NOT Running Fans - A1, B1, B2

D Running Fans - A1, A2, B2
NOT Running Fans - B1

Answer A

References
INPO Question 20343
NOH01DWVENT-02, Drywell Ventilation System, p15
HC.OP-SO.GT-0001, DRYWELL VENTILATION SYSTEM OPERATION, p.5

Justification

References during Exam None

A - CORRECT - Bus 10A401 powers MCC 10252, on a LOP all of the #1 fans lose power - A1, B1, etc. Any #1 fan that was running will have a Low flow condition on it. Since the #2 fans did NOT lose power, if they sense a Low flow and are in AUTO they will start. This causes fan D2 to start. In addition, all #2 fans that were running will continue to run, this leaves fans A2 and C2 running. When bus 10A401 is re-powered ALL #1 fans get a start signal, this causes all #1 fans to start.
B - INCORRECT - this would be correct, if you thought only fans that were running previously received a start signal.
C - INCORRECT - this would be correct, if you thought that the bus was stripped on a LOP (it is stripped on a LOCA)
D - INCORRECT- this would be correct if bus 10A401 powered all the #2 fans.

Question Source Mod

- [ ] Memory Level
- [x] Comprehension Level

Question History:

Modified Question 9/20
SXD - OK
AF - changed to OPCON 4, changed to I&C problem, still NOT sure about Question
MB - 11/8 - reduced to just A and B fans.

Question	62

Hope Creek RO Exam - Nov 2005

☑ **RO**
☐ **SRO**

Tier # 2 **Group #** 2

Importance 3.1

HC Obj: CIRCWAE006

256000 Reactor Condensate System

A1.07 Ability to predict and/or monitor changes in parameters associated with operating the REACTOR CONDENSATE SYSTEM controls including: (CFR: 41.5 / 45.5) System lineup

Question

Given the following:
- 100% power operation with all Circulating Water pumps in service
- Loss of the 10A502 4.16KV Switchgear (Loss of power to the Circulating Water pump discharge valve hydraulic power units)

Which of the following describes the response of the Circulating Water pump discharge valves (HV-2152A,B,C, and D):

A All failing full closed

B HV-2152A and C failing full closed, and the HV-2152B and D failing as is

C HV-2152A, and C failing as is, and the HV-2152B and D failing full closed

D All failing full open

Answer C

References Ref: HC.OP-SO.DA-0001
Hope Creek Bank - Q54966

Justification

References during Exam None

A - INCORRECT - All failing full closed. A and C fail as is which is open.
B - INCORRECT - HV-2152A and C failing full closed, and the HV-2152B and D failing as is. Opposite of actual.
C - CORRECT - HV-2152A, and C failing as is, and the HV-2152B and D failing full closed. The HCU valves are aligned per HC.OP-SO.DA-0001.
D - INCORRECT - All failing full open. B and D fail closed on the 502 bus loss.

Question Source Bank ☑ **Memory Level** ☐ **Comprehension Level**

Question History:

New 9/8
SXD - OK
AF - look at re-writing question to CREF question and use Hi rad signal with damper closed. Put standby train in Manual, other one won't auto start.
MB- 10/27 - re-sampled - new bank question
MB - 11/8 removed figure from handout

Question	63

Hope Creek RO Exam - Nov 2005

☑ RO
☐ SRO

Tier # 2 **Group #** 2

Importance 3.2

HC Obj: RXVESSE004

290002 Reactor Vessel Internals

K1.20 Knowledge of the physical connections and/or cause effect relationships between Reactor Vessel Internals and the following — Nuclear Instrumentation (CFR:41.2 to 41.9/ 45.7 to 45.8)

Question

Which of the following correctly states how the LPRM strings are mounted in the Reactor Vessel:

A The LPRM's are mounted in dry tubes.
The incore tube assembly is installed or removed from below the core.

B The LPRM's are mounted in wet tubes.
The incore tube assemby is installed or removed from below the core.

C The LPRM's are mounted in dry tubes.
The incore tube assemby is installed or removed from above the core.

D The LPRM's are mounted in wet tubes.
The incore tube assembly is installed or removed from above the core.

Answer D

References NOH01RXVESS-02, Reactor Vessel and Internals - p.17
NOH01LPRM00-01, P. 7 LPRM lesson plan

Justification

References during Exam None

A - INCORRECT - the SRM's and IRMs are mounted in dry tubes that enter from below the core.
B - INCORRECT - Control rod guide tubes are preforated with 4 holes to cool the nuclear instrumentation, the nuclear instrumentation is housed in dry tubes.
C - INCORRECT - the LPRM's strings are contained in a wet tube housing and the assembly is installed and removed from above the core but the dry tubes enter below the core.
D - CORRECT - the LPRM's are mounted in wet tubes.

Question Source New ☑ **Memory Level** ☐ **Comprehension Level**

Question History:

New 8/29
SXD - OK
RJC - LOD 1
SXD - Leave in
AF - change answer to "A"
MB - 11/8 changed enter to installed or removed, changed answer back to "C", clarified stem wording.
MB - 11/17 - Archie to look up
MB - 11/18- Mounted in wet tubes

Question 64

- [x] **RO**
- [ ] **SRO**

Tier # 2 **Group #** 2

Importance 3.8

HC Obj: DWVENTE003

214000 Rod Position Information System

A4.02 Ability to manually operate and/or monitor in the control room: Control rod position (CFR: 41.7 / 45.5 to 45.8)

Question

Given the following conditions:

- control rod withdrawal signal is present
- control rod 46-35 has a Data Fault indicated on the Rod Select Module

This indicates control rod 46-35 has…

A one even reed switch is closed.

B one odd reed switch is closed.

C two or more even reed switches are closed.

D two or more odd reed switches are closed.

Answer C

References Hope Creek Question - Q55925
HC.OP-SO.SF-0001, Rev 9, Attachment 1

Justification

References during Exam None

A - INCORRECT - an odd reed switch closed. This causes a "--" on the four rod display and a Rod Drift if no rod motion signal is present but not a Data Fault.
B - INCORRECT - an even reed switch closed is the normal configuration - no alarm given.
C - CORRECT - 2 or more even reed switches are closed gives a Data fault.
D- INCORRECT - 2 or more ODD reed switches closed will give a "--" on the four rod display, but not a data fault.

Question Source Bank

- [x] **Memory Level**
- [ ] **Comprehension Level**

Question History:

Re-sampled 9/14
SXD - K/A mismatch
MB - 9/27 - inserted correct K/A based on re-sample
SXD -OK
AF - OK
MB - 10/27 - Re-sampled due to over sampling of 223001
MB - 11/8 - OK

Question	65

Hope Creek RO Exam - Nov 2005

☑ RO
☐ SRO

Tier # 2 *Group #* 2

Importance 3.2

HC Obj: FPCC00E010

233000 Fuel Pool Cooling and Clean-up

K3.01 Knowledge of the effect that a loss or malfunction of the FUEL POOL COOLING AND CLEAN-UP will have on following: (CFR: 41.7 /45.6) Fuel pool temperature

Question

Given the following conditions:

- The plant is in Operational Condition 1, two weeks after a refueling outage
- The Fuel Pool Cooling system is operating with one pump and heat exchanger in service
- The Fuel Pool to Reactor Cavity Gates are installed
- NO makeup water sources are available
- Assume NO evaporative losses

Which of the following is the effect on Spent Fuel Pool water temperature and level if a leak develops on the common FPCC Pump Suction?

A Fuel pool temperature will remain stable and water level will lower slightly then stabilize.

B Fuel pool temperature will rise and water level will continuously lower.

C Fuel pool temperature will rise and water level will lower slightly and stabilize.

D Fuel pool temperature will remain stable and water level will continuously lower.

Answer C

References Ref: M-53-1
Hope Creek Bank - Q56203

Justification

References during Exam None

A - INCORRECT - FPCC is lost - Fuel Pool temp will rise.
B - INCORRECT - Level will lower to the bottom of the weir overflow pipe and then stop.
C - CORRECT: rise and water level will lower slightly and stabilize. The skimmer surge tank will drain and the FPCC pumps will trip. Fuel pool level will drain to the bottom of the weir overflow pipe then stop. Water temp will increase because FPCC is lost. Temperature rise causes water to expand, level maintained at the weir.
D - INCORRECT - FPCC is lost - Fuel Pool temp will rise

Question Source Bank ☐ Memory Level ☑ Comprehension Level

Question History:

New 9/21
SXD - OK
AF - minor comments
MB - made changes as requested
MB - 10/31 - re-sampled due to oversampling of 223001
MB - 11/8 - OK

Question 66

Hope Creek RO Exam - Nov 2005

☑ **RO**
☐ **SRO**

Tier # 3 **Group #**

Importance 3.1

HC Obj:

2.1.21 Generic

Ability to obtain and verify controlled procedure copy (CFR: 45.10 / 45.13)

Question

Which one of the following identifies procedures considered "valid working copies" without DCRMS verification prior to use?

A Only Department Implementing Procedures (DIP's) in the Control Room.

B Any procedures stamped "Controlled Copy" in RED.

C DIP's in the Control Room and at operations field locations.

D Nuclear Administrative Procedures and Department Administrative Procedures stamped "Controlled Copy" in RED.

Answer C

References INPO Question 23092, Question taken from Salem NRC Exam 11/02
NC.DM-AP.ZZ-0005, Step 5.1.2
SHOP 109

Justification

References during Exam None

A - INCORRECT - per DMAP-05, step 5.1.2 Operations Department DIPs in the Control Room and Operation field Locations and Emergency Plan DIPs in Emergency Response facilities are the most current version of the procedure and can be used without DCRMS verification prior to use. Can also use DIP's in field locations.
B - INCORRECT - see "A" above, in addition step 5.1.1 states with the exceptions noted in 5.1.2 below, all procedures shall be verified as valid working copies prior to use.
C - CORRECT - see "A" above
D - INCORRECT - see "A" above

Question Source Bank ☑ **Memory Level** ☐ **Comprehension Level**

Question History:

New 8/29
SXD - OK
AF - NOT sure if we have objective that they need to know this.
MB - 11/14 - OK

Question 67

☑ RO
☐ SRO

Tier # 3 Group #

Importance 2.5

HC Obj:

2.1.14 Generic

Knowledge of system status criteria which require the notification of plant personnel (CFR: 43.5 / 45.12)

Question

Which one of the following identifies the event(s), if any, that REQUIRE a Plant Page announcement in accordance with Plant Operating procedures:

EVENT 1 - When the Reactor is critical
EVENT 2 - When the Reactor is scrammed

A EVENT 1 - YES
EVENT 2 - YES

B EVENT 1 - YES
EVENT 2 - NO

C EVENT 1 - NO
EVENT 2 - YES

D EVENT 1 - NO
EVENT 2 - NO

Answer A

References NC.NA-AP.ZZ-0005, STATION OPERATING PRACTICES, p 19, attachment 3 page 17

Justification

References during Exam None

A - CORRECT - per NAP-5 have to announce both reactor critical and scrams
B - INCORRECT - see "A" above
C - INCORRECT - see "A" above
D - INCORRECT - see "A" above

Question Source New ☑ Memory Level ☐ Comprehension Level

Question History:

New 9/8
SXD - OK
AF - OK
MB - 11/8 changed events to make it clear which is required.

Question 68

☑ **RO**
☐ **SRO**

Tier # 3 **Group #**

Importance 3.4

HC Obj:

2.1.33

Generic

Ability to recognize indications for system operating parameters
which are entry level condition for Technical Specifications
(CFR: 43.2 / 43.3 /45.3)

Question

During Plant startup the following conditions are observed:

TIME	RPV Pressure
0700	172 psig
0715	191 psig
0730	205 psig
0745	233 psig
0800	373 psig

Which one of the following is the latest time at which heatup must be secured in order to prevent exceeding the Technical Specification limit for heatup at the CURRENT heat up rate?

A 0800

B 0815

C 0830

D 0845

Answer B

References
Hope Creek Question - Q56983
Steam Tables
Tech Spec

Justification

References during Exam Steam Tables

Justification
172 psig = 186.7psia=376F
191psig=205.7 psia = 384°F
205 psig – 219.7 psia = 390F
233 psig = 247.7 psia = 400F
373 psig = 387.7 psia = 442F-This gives a 42F change in 15 mins. Current heatup rate is 42F every 15 min (168 degrees/hr). 0815 - Correct- At this rate we must terminate the H/U by 0815 to keep from exceeding the allowable heatup, we would be at 484°F (this would be 100 degrees/hr).

Question Source Bank ☐ **Memory Level** ☑ **Comprehension Level**

Question History:

SXD review 7/21 - OK
AF - 2 answers. Changed 7:30 pressure to 205 psig to make only 1 correct answer
MB - 8/24 - Made changes as requested
MB - 11/8 - relooked at numbers OK
MB - 11/17 Archie to look at
MB - 11/18 OK

Question 69

Hope Creek RO Exam - Nov 2005

☑ **RO**
☐ **SRO**

Tier # 3 **Group #**

Importance 3.7

HC Obj:

2.2.1

Generic

Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity.

Question

The plant is shutdown with ‘B’ RHR in shutdown cooling, OPCON 4. Inservice stroke time testing needs to be performed on the discharge valve of the ‘A’ recirculation pump prior to commencing startup.

What precautions/limitiations exist to allow/prevent this evolution to take place?

A As long as RPV vessel level is offscale high on all Narrow Range instruments, Shutdown cooling may be secured and the recirculation discharge valve stroked without loss of decay heat removal and vessel stratification.

B System Operating procedures for both Recirculation system and RHR system prohibit the opening of Recirculation pump discharge valves while RHR is in Shutdown Cooling, to prevent core bypass flow and vessel stratification.

C This evolution can only be performed after the ‘B’ Recirc pump is placed in service and establishment of forced circulation through the vessel is assured.

D Prior to stroking the discharge valve on ‘A’ Recirculation pump, the suction valve must be verified closed.

Answer D

References
Hope Creek Question - Q56375
HC.OP-IO.ZZ-0002 section 3.2.5
HC.OP-SO.BC-0002, 3.2.11

Justification

References during Exam None

·"Prior to stroking the discharge valve on ‘A’ Recirculation pump, the suction valve must be verified closed. Correct
·"This evolution can only be performed after the ‘B’ Recirc pump is placed in service and establishment of forced circulation through the vessel is assured."- Incorrect- The ‘can only’ distractor is wrong because the word “only “ is used, along with the combination of RHR and Recirc pump combinations would still require the suction valve closed while stoking the valve
·"System Operating procedures for both Recirculation system and RHR system prohibit the opening of Recirculation pump discharge valves while RHR is in Shutdown Cooling, to prevent potential core bypass flow and vessel stratification." - Incorrect- The ‘SOP’ distractor is wrong because the IO allows this condition and applicable exception to the SO guidance
·"As long as RPV vessel level is pegged high on all Narrow Range instruments, Shutdown cooling may be secured and the recirculation discharge valve stroked without potential problem of loss of decay heat removal and vessel stratification." – Incorrect- The ‘RPV vessel level’ is wrong because minimum level for natural circulation is +80” which is well above the Narrow Range detector capability to read, and does NOT assure the appropriate level.

Question Source Bank ☑ **Memory Level** ☐ **Comprehension Level**

Question History:

SXD review 7/21 - OK
AF - OK
AF - K/A mismatch
MB - 11/8 - OK
MB - 11/17 - Archie to look at
MB - 11/18 deleted last part of "D"

Question 70

Hope Creek RO Exam - Nov 2005

☑ **RO**

☐ **SRO**

Tier # 3 **Group #**

Importance 2.8

HC Obj:

2.2.34 Generic

Knowledge of the process for determining the internal and external effects on core reactivity (CFR: 43.6)

Question

A Reactor startup from Cold Shutdown is in progress.

The ECP was calculated based upon the following:

- Reactor Coolant temperature at 140 °F
- Total Core Flow at 30 X 10E6 lbm/hr
- At time of criticality, Reactor has been shutdown for 40 hours
- Feedwater temperature 120 °F

Which of the below will result in criticality later in the rod pull sequence than the Predicted ECP?

A Total Core Flow is increased to 35 x 10E6 lbm/hr

B Feedwater temperature drops to 100°F

C Criticality occurs 30 hours after shutdown

D Reactor Coolant temperature drops to 125°F

Answer C

References INPO Question - 25685
GFES

Justification

References during Exam None

A - INCORRECT - Change in Core Flow has NO effect on criticality until voiding occurs. Criticality as predicted.
B - A drop in FW temperature of 5°F is a net positive reactivity effect if FW is injecting. If not, there is NO effect. Criticality will occur earlier or as predicted.
C - CORRECT - 30 hours vs. 40 hours results in criticality occurring at a higher Xe concentration requiring rods to be withdrawn more, therefore later than predicted.
D - INCORRECT - A reactor coolant drop in temperature is a net positive reactivity effect. Criticality earlier than predicted.

Question Source Bank ☐ **Memory Level** ☑ **Comprehension Level**

Question History:

SXD - OK
AF - minor changes
MB - made changes as requested.
MB - 11/8 - OK
MB - 11/17 - "A" changed decrease to increase

Question 71

Hope Creek RO Exam - Nov 2005

☑ **RO**
☐ **SRO**

Tier # 3 **Group #**

Importance 2.6

HC Obj: ADMPROE057

2.3.1 Generic

Knowledge of 10 CFR 20 and related facility radiation control requirements (CFR: 41.12 / 43.4. 45.9 / 45.10).

Question

Radiation Protection technicians have surveyed the Refuel Floor Reactor Head Laydown Area during an outage and obtained the following results:

- Highest Area Dose Rate one foot from any source in the room: 72 mr/hr
- Airborne Concentration: 0.15 DAC
- Smear Results: 750 dpm/100 cm2 gamma

Based on these results the area shall be posted as a:

I. Radiation Area
II. High Radiation Area
III. Very High Radiation Area
IV. Contaminated Area
V. Airborne Radioactivity Area

A I, and V

B I, IV, and V

C I and IV

D II and IV

Answer A

References Hope Creek Question - Q76884 (Modified slightly)
NC.NA-AP.ZZ-0024, rev 13, p.23

Justification

References during Exam None

A - CORRECT - Airborne rad area > 10% or .10 DAC
B - INCORRECT - NOT a Contaminated Area - must be > 1000 dpm/100cm2
C - INCORRECT - NOT a Contaminated area and it is an Airborne Area
D - INCORRECT - NOT a High Radiation Area - must be > 100mr/hr

Question Source Bank ☑ **Memory Level** ☐ **Comprehension Level**

Question History:

SXD review 7/21 - LOD 1.5 - evaluate writing a more difficult question
Changed question out with another HC bank question that seems more difficult
AF - OK
RJC - Remove should, change to shall
MB - Made changes as requested - 10/6
MB - 11/8 - OK

Question	72

Hope Creek RO Exam - Nov 2005

☑ RO
☐ SRO

Tier # 3 Group #

Importance 2.9

HC Obj:

2.3.10 Generic

Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure (CFR: 43.4 / 45.10)

Question

Given:

- The Shift Manager declared a Site Area Emergency thirty (30) minutes ago.
- The TSC is NOT activated.
- The OSC is activated.
- EOP actions outside the control room are necessary to vent the scram air header.
- The maximum expected exposure is 1500 mRem
- The task is NOT going to require entry into a Harsh Environment Area
- Acts of sabotage are NOT suspected
- Area Radiation Monitors (ARMs) on the Reactor Building 102' elevation are alarming.

Which one of the following describes the requirements to perform the directed actions of venting the scram air header?

A The operator may NOT enter Reactor Building until the TSC is activated.

B The operator shall be assigned to an OSC team of at least 2 people.

C The operator is NOT required to be a qualified emergency response member as long as at least ONE member of his team is.

D The operator may perform actions independently as a single person OSC team.

Answer B

References INPO Question 267
NC.EP-EP.ZZ-0202, P15

Justification

References during Exam None

A - INCORRECT - there are NO requirements that prohibit Reactor Building entry until the TSC is activated.
B - CORRECT - You can use use a single person OSC team as long as 3 criteria are met - expected exposure is less than 1000 mR, Task does NOT require entry into a Harsh Environment Area, and Acts of Sabotage are NOT suspected. Since exposure is > 1000 mR must be on a team of at least 2 people
C - INCORRECT - ALL personnel who are selected for OSC teams are qualified emergency response members.
D - INCORRECT - see "B" above. Task does NOT meets criteria for single member OSC team.

Question Source New ☑ Memory Level ☐ Comprehension Level

Question History:

New 8/29
SXD - K/A mismatch??
MB - 9/27 - I don't think so, this question is asking for what procedures need to be followed to reduce excessive personnel exposure during an emergency

-- Need assistance from Archie to determine proper Hope Creek Procedure that gives this guidance --
AF - No Procedure reference. Archie to see if we have procedure guidance
MB - 11/1 - changed question to match EP-EP 202 criteria.
MB - 11/8 raised rad to 1500 MR change answer to B

Question 73

- [x] **RO**
- [ ] **SRO**

Tier # 3 **Group #**

Importance 4.0

HC Obj: ABRPV6E003

2.4.49 Generic

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

Question

Given the following condtions:

- Power ascension is in progress following a refuel outage
- Reactor power is 97%
- PSV-F013P opens inadvertently and does NOT reclose

Select the immediate operator action.

A Depress the "Reset Logic Armed" pushbutton for "B" Low-Low set logic.

B Lock the mode switch in SHUTDOWN.

C Reduce reactor power to 95%.

D Dispatch the operator to remove the SRV fuses.

Answer C

References HC.OP-AB.RPV-0006
Hope Creek Bank - Q77604

Justification

References during Exam None

A - INCORRECT - Subsequent action
B - INCORRECT - Retainment override if unable to close the valve.
C - CORRECT - Immediate action.
D - INCORRECT - Subsequent action.

Question Source Bank

- [x] **Memory Level**
- [ ] **Comprehension Level**

Question History:

New 8/29
SXD - OK
AF - Job link, don't think anybody will know this, perhaps make more operationally oriented, what conditions do they need to start fire pump manually - activate fire systems from the control room, ar.qk-0002 - give flowchart.
MB - 10/27 - re-sampled
MB - 11/8 - OK

Question	74

Hope Creek RO Exam - Nov 2005

☑ RO
☐ SRO

Tier # 3 **Group #**

Importance 3.3

HC Obj:

2.4.39

Generic

Knowledge of the RO's responsibilities in emergency plan implementation (CFR: 45.11)

Question

You are a licensed Reactor Operator assigned to the WIN Team, in the WIN Team office. You do NOT have assigned responsibilities in the Emergency Response Organization (ERO).

A transient occurs that results in the declaration of an ALERT Emergency and Accountability.

To which of the following locations do you report?

A The Control Room

B The Processing Center

C The OSC

D The Hope Creek Cafeteria

Answer C

References
INPO Question 25692
Lesson Name - OVERVIEW - NEPOVERVIEWC, p.15

Justification

References during Exam None

A - INCORRECT per Overview lesson plan All Ops personnel report to OSC for accountability.
B - INCORRECT see A above.
C - CORRECT see A above
D - INCORRECT see A above

Question Source Mod ☑ **Memory Level** ☐ **Comprehension Level**

Question History:

New 9/8
SXD - Add justification - low level of difficulty
MB - 9/27 - -- Need assistance from Archie to determine which procedure provides guidance to Licensed Operators NOT on shift as to where they report during an emergency. My answer was based on GET info. --
SXD - Very GET, may need to re-sample, Archie to provide procedure guidance if possible
RJC - Make sure it is a requirement
MB - Archie to check
AF - Got procedure reference
MB - 11/8 - OK
MB- 11/17 changed "B" and "D" distractors

Question 75

Hope Creek RO Exam - Nov 2005

☑ **RO**
☐ **SRO**

Tier # 3 **Group #**

HC Obj: NOH01FPCC00 Obj. 10

Importance 3.3

2.4.31 Generic

Knowledge of annunciators alarms and indications / and use of the response instructions. (CFR: 41.10 / 45.3)

Question

Overhead Annunciator Window Box A4-D4 "CNDS/REFUEL STOR & XFR - SYS TROUBLE" has 2 pieces of red tape diagonally placed across the annunciator window in the shape of an "X".

Which one of the following describes the significance of this indication in accordance with SH.OP-AP.ZZ-0030, Operator Burden Program?

A The entire annunciator window is inoperable.

B One input to the annunciator window is inoperable.

C Indicates that a T-MOD has been written against the annunciator window.

D Indicates that a design change request notification has been submitted against the annunciator window.

Answer A

References INPO Question 818
SH.OP-AP.ZZ-0030, Operator Burden Program, p.4

Justification

References during Exam HC.OP-AR.ZZ-0003, window A4-D4

A - CORRECT - per SHOP-30 p.4 - if the entire annunciator window is inoperable then 2 pieces of red tape diagonally placed across the annunciator window in the shape of an "X"
B - INCORRECT - per SHOP-30 p.4 - if one or more inputs of a multiple input annunciator are inop then red tape should be placed diagonally across the annunciator window.
C - INCORRECT - While a notification should have been written against the annunciator window, the red "X" indicates that the annunciator is INOP.
D - INCORRECT - per SHOP-30 p.4 - if a design change request notification has been written against an instrument a piece of red tape should be placed across the instrument to alert the operator that the instrument is NOT reliable. NOT 2 pieces of red tape in an "X"

Question Source Mod ☑ **Memory Level** ☐ **Comprehension Level**

Question History:

SXD review 7/27 - NOT SRO level - re-write
Re-wrote question - 7/29 - somewhat based on INPO Question 22362
JD - 8/2 - Why are C,D plausible
MB - 8/3 - Changed AB.CONT-0005, Irradiated Fuel damage to EO.ZZ-0103/4 since Radiation levels in the Reactor Building are rising and operator may be concerned about reactor building release.
AF - OPCON 5 vs. Mode 5, changed Fuel Pool to LPCI and changed RHR to Fire water.
MB 8/30 - Realized Question should be RO -went back to original question.
SXD - OK
AF -OK
MB - 11/8 changed annunciator window, changed notification to T-MOD

Question 76

Hope Creek SRO Exam - Nov 2005

☐ **RO**
☑ **SRO**

Tier # 1 **Group #** 1

Importance 3.8

HC Obj:

295003 Partial or Complete Loss of AC / 6

AG2.1.32 Ability to explain and apply system limits and precautions
(CFR 41.10/ 43.2/ 45.12)

Question

Hope Creek was operating at 30% power when a Station Blackout (loss of all onsite and offsite power) occurred causing a Reactor Scram.

Current plant conditions are as follows:

- Drywell temperature - 300°F decreasing slowly
- RPV pressure - 273 psig decreasing slowly
- Reactor Power - all rods fully inserted
- Reactor level - (-100" decreasing)
- RCIC - tagged out and disassembled
- HPCI - tripped on overspeed and will NOT restart
- "A" EDG - tagged out for maintenance
- "B" EDG - running unloaded - output breaker failed open on anti-pump circuitry
- "C" EDG - tripped on Bus differential overcurrent
- "D" EDG - failure to start - low air pressure ~20 psig

Based on these conditions, the Control Room Supervisor shall:

A direct the NEO to reset the Bus differential overcurrent on the "C" EDG and restart the "C" EDG.

B direct the RO to depress the TRIP pushbutton on the "B" EDG output breaker and verify output breaker closes.

C enter procedure HC.OP-EO.ZZ-0202, Emergency Depressurization based on high Drywell temperature.

D enter procedure HC.OP-EO.ZZ-0202, Emergency Depressurization before Reactor Water Level decreases to -129".

Answer B

References HC.OP-AB.ZZ-0135, Station Blackout// Loss of Offsite Power// Diesel Generator Malfunction p. 2

Justification

References during Exam EOP Flowchart - 202 and 101, 102 with NO entry conditions

A - INCORRECT - bus differential current should NOT be reset without electrical maintenance determining and correcting the cause.
B - CORRECT - per HC.OP.AB.ZZ-0135, Station Blackout p. 18 step 5.16 - The Anti-pump circuitry on the D/G output breaker could cause the output breaker to fail open. To load the D/G under this condition the operator must depress the TRIP push-button (even though the breaker is already tripped) to reset the logic. When the TRIP push-button is released, then the breaker will close and the D/G will load.
C - INCORRECT - Emergency Depressurization procedure should NOT be entered until DW temperature exceeds 340°F and current drywell temperature is decreasing.
D - INCORRECT - Emergency Depressurization procedure should NOT be entered until is less than -129" but before level decreases to -185"

Question Source New ☐ **Memory Level** ☑ **Comprehension Level**

Question History:

SXD review 7/27 - NOT SRO level - re-write
8/2 - re-wrote question
AF - bullets, give the Operator the EOP flowchart. Look at possible double jeopardy.
MB - 8/24 - Made changes as requested
MB - 11/8 - OK

Question 77			**Hope Creek SRO Exam - Nov 2005**
☐ RO ☑ SRO	*Tier #* 1	*Group #* 1	*HC Obj:* 0AB151E003
	Importance 3.3		

295004 Partial or Total Loss of DC Pwr / 6

AA2.04 Ability to determine and interpret the following as they apply to Partial or Total loss of DC power:(CFR: 41.10 J 43.5 / 45.13) System Lineups

Question

Given the following condtions:

- The Reactor is in Operational Condition 4.
- The NEO's are performing a system lineup on 24 VDC.
- Plant startup operations are in progress.
- The negative battery charger for the "A" ±24 VDC System is found to be out of service.
- The positive battery charger for the "B" ±24 VDC System is placed on an equalizing charge.
- All other equipment was found to be aligned for normal operation.

Which of the following Technical Specifications (if any) needs to be entered if this condition were to remain for 24 hours?

I. 3.3.1 - Reactor Protection Instrumentation
II. 3.8.2.2 - D. C. Sources - Shutdown

A I. YES
II. YES

B I. YES
II. NO

C I. NO
II. YES

D I. NO
II. NO

Answer B

References
Hope Creek Bank - Q61702 - Modified
HC.OP-AB.ZZ-0151, Sections 2.1, 4.5 & 5.1
0301-000.00H-000069-13, Sections VII.A.1, VII.B.1-3, & Figures 32 & 33
HCGS Incident Report 86-067

Justification

References during Exam 3.8.2.2 and 3.3.1

The negative charger only charges the negative battery while the positive charger only charges the posrtive battery. Even with the positive charger operating in the Equalizer mode, the negative battery will be discharged resulting in the loss of the DC bus.
A - INCORRECT - TS 3.8.2.2 Does NOT need to be entered because it only addresses 125V DC NOT 24V DC.
B - CORRECT - Loss of -24VDC will cause the IRMs to be INOP in Mode 4 need at least 2 channels per trip system, since on "A" channel there are NO trip systems OPERABLE TS 3.3.1 needs to be entered.
C - INCORRECT - INCORRECT see "A" above.
D - INCORRECT - INCORRECT see "B" above

Question Source Mod ☐ *Memory Level* ☑ *Comprehension Level*

Question History:
New 9/21
SXD - OK
AF -OK
MB - 11/1 - changed out question with Bank question Q61702
MB - 11/8 similar to 36, not SRO - Re-sample
MB - 11/10 - after talking to RJC decided to re-write to make more SRO. Re-wrote question.

Question 78

Hope Creek SRO Exam - Nov 2005

☐ RO ☑ SRO

Tier # 1 **Group #** 1

HC Obj: NOH01AB0000-01 - obj. 6

Importance 3.8

295006 SCRAM / 1

AG2.1.32 Ability to explain and apply system limits and precautions (CFR 41.10/ 43.2/ 45.12)

Question

Hope Creek is at 20% power following a startup from a refueling outage when the plant scrams.

The Control Room Supervisor has entered HC.OP-AB.ZZ-0000, Reactor Scram and has the following plant conditions:

- RPV Level - (+33" stable)
- RPV Pressure - 1000 psig stable
- Mode Switch - Locked in Shutdown position
- All Control Rods fully inserted

You have reached step S-8:

"IF Conditions permit
THEN RESET the Scram
AND INSERT a Half Scram (if required)

Which of the following conditions would REQUIRE you to INSERT a Half Scram?

I. APRM channels "A" and "C" INOP
II. IRM channels "E" and "F" INOP
III. 1 Reactor Vessel Steam Dome Pressure High Transmitter INOP

A I Only

B II Only

C I, II Only

D I, II, and III

Answer A

References NOH01AB0000-01, Reactor Scram AB-0000 - p. 17-18
Tech Spec 3.3.1

Justification

References during Exam 3.3.1

A - CORRECT - Per TS 3.3.1 b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems, place at least 1 trip system in the tripped condition within one hour and take the ACTION required by Table 3.3.1-1
For the APRM's in OPCON 3 - Minimum OPERABLE Channels per Trip System is 2, therefore if 2 APRM were INOP you would only have 1 in that Trip System OPERABLE and would need to insert a Half-scram
B - INCORRECT - Per TS 3.3.1-1 in OPCON 3 you are only required to have 2 IRM's OPERABLE per trip system, since you have 3 available having 1 INOP still leaves 2 that are OPERABLE and so you would NOT have to insert a Half-Scram
C - INCORRECT - see B above
D - INCORRECT - see B above, also Reactor Steam Dome Pressure High transmitter is only required in OPCON 1 or 2, since you are in OPCON 3 this would be N/A

Question Source New ☐ **Memory Level** ☑ **Comprehension Level**

Question History:

New 8/30
SXD - borderline SRO, write something to address objective Obj 6 in AB0000-01
MB - 10/3 wrote new question to address OBJ 6 (SRO ONLY) objective of Lesson Plan in AB0000-01
SXD - OK
AF - K/A mismatch, want 3.3.1
MB - talk to SXD, I think they should know what puts them in a Tech Spec without having to have Tech Specs.
MB - minor editorial change to stem 11/8
MB - 11/17 - changed to startup from a refueling outage.

Question 79

Hope Creek SRO Exam - Nov 2005

☐ **RO**

☑ **SRO**

Tier # 1 **Group #** 1

HC Obj: NOH01MSTEAMC-02 OBJ 3

Importance 3.7

295019 Partial or Total Loss of Inst. Air / 8

AA2.02 Ability to determine and interpret the following as they apply to Partial or Total loss of Instrument Air:(CFR: 41.10/43,5/ 45.13) Status of safety-related instrument air system loads

Question

Hope Creek is operating at 100% power when an Instrument Air line in the Turbine Building ruptures. The air compressors are unable to keep up with the loss of air and Instrument Air pressure is lowering.

What will the long term Reactor Pressure Vessel level control and pressure control strategy be for the loss of Instrument Air in accordance with HC.OP-AB.ZZ-0000, Reactor SCRAM?

A Bypass valves for pressure control, Maximize CRD for level control.

B SRVs for pressure control, Maximize CRD for level control.

C SRVs for pressure control, HPCI/RCIC for level control.

D Bypass Valves for pressure control, HPCI/RCIC for level control.

Answer C

References INPO Question 25895
NOH01MSTEAMC-02, MAIN STEAM SYSTEM, p. 46

Justification

References during Exam None

A - INCORRECT - Condenser is NOT available and NO condensate line up is possible due to level control valves fail closed on a loss of air.
B - INCORRECT - CRD flow control valves fail closed on a loss of air
C - CORRECT - Outboard MSIVs will go closed on a loss of air, therefore NO steam for feedpumps or use of the main condenser for decay heat. Condensate will be unavailable due to NO feedpath on a loss of air.
D - INCORRECT - Condenser is NOT available for pressure control

Question Source Mod

☐ **Memory Level** ☑ **Comprehension Level**

Question History:

New 9/8
SXD - Look for procedure tie in.
MB - 9/27 - Added per AB-SCRAM
SXD -OK
AF - Another Inst. Air question, minor questions
MB - Made changes as requested.
MB - 11/8 OK

Question	80

Hope Creek SRO Exam - Nov 2005

☐ RO
☑ SRO

Tier #	1	Group #	1

Importance 3.3

HC Obj:

295028 — High Drywell Temperature / 5

EG2.4.50 — Ability to verify system alarm setpoints and operate controls identified in the alarm response manual. (CFR 45.3)

Question

Given the following conditions:

- A small steam leak has occured in the drywell causing a reactor scram	
- Two control rods are at position	06
- RPV level	+30 inches
- RPV pressure	920 psig
- Suppression pool level	75 inches
- Suppression pool temperature	80 °F
- Drywell pressure	3 psig
- Average drywell temperature	330 °F and rising at 1°F per minute
- Suppression chamber pressure	3 psig

Which of the following describes the next operator action(s) in accordance with the Emergency Operating Procedures?

A Shutdown the Reactor Recirculation Pumps and Drywell Cooling Fans and initiate one loop of drywell spray.

B Verify all injection into the RPV except SLC, CRD and RCIC is terminated and prevented and then emergency depressurize the reactor.

C Rapidly depressurize the reactor using the main turbine bypass valves.

D Initiate suppression chamber sprays and commence a normal reactor cooldown. (Less than 90 F per hour)

Answer B

References Hope Creek Question Q56045
HC.OP-EO.ZZ-0102 Bases, step DW/T-5

Justification

References during Exam EOP 101a, 102

A - INCORRECT - ·Shutdown the Reactor Recirculation Pumps and Drywell Cooling Fans and initiate one loop of drywell spray.-incorrect- CANNOT DW Spray since outside of DWT-P curve.
B - CORRECT - ·Verify all injection into the RPV except SLC, CRD and RCIC is terminated and prevented and then emergency depressurize the reactor.-correct- EOP-0202 step ED-3
C - INCORRECT - ·Rapidly depressurize the reactor using the main turbine bypass valves.-incorrect- EOP-101A prevents use of BPVs in this situation
D - INCORRECT - ·Initiate suppression chamber sprays and commence a normal reactor cooldown. (Less than 90 F per hour)-incorrect- must stabilize pressure until S/D under all conditions without Boron

Question Source Bank ☐ **Memory Level** ☑ **Comprehension Level**

Question History:

SXD review - 7/29 - OK
AF - give EOP flowcharts during exam
MB - 8/24 - Made changes as requested
MB - 11/8 - OK - Look at Distractor "A"

Question 81

Hope Creek SRO Exam - Nov 2005

☐ **RO**

☑ **SRO**

Tier # 1 **Group #** 1

HC Obj: NOH01PRICON-02
Obj. R9c

Importance 4.2

295030 Low Suppression Pool Wtr Lvl / 5

EA2.01 Ability to determine and interpret the following as they apply to Low Suppression Pool Water level (CFR:41.10/ 43.5/ 45.13) Suppression Pool level

Question

With the plant operating at 100% power the RO reports to you that Suppression Pool Level has drifted out of the allowable Technical Specification value.

Investigation reveals that a small leak has developed on the Instrument line for Suppression Pool Level transmitter LT-4805-1 just downstream of valve V9982.

Using the attached figure, how will the reading on LT-4805-1 compare to ACTUAL Suppression Pool level and what is the Technical Specification bases for maintaining ACTUAL level at the proper level.

A
- LT-4805-1 will read HIGHER than ACTUAL level.
- Bases for maintaining level is to ensure adequate NPSH exists for ALL pumps (HPCI, RCIC, LPCI and CSS) to inject following a Design Basis LOCA.

B
- LT-4805-1 will read LOWER than ACTUAL level.
- Bases for maintaining level is to ensure adequate NPSH exists for ALL pumps (HPCI, RCIC, LPCI and CSS) to inject following a Design Basis LOCA.

C
- LT-4805-1 will read HIGHER than ACTUAL LEVEL.
- Bases for maintaining level is to ensure primary containment pressure will NOT exceed design pressure during a primary system blowdown.

D
- LT-4805-1 will read LOWER than ACTUAL LEVEL.
- Bases for maintaining level is to ensure primary containment pressure will NOT exceed design pressure during a primary system blowdown.

Answer D

References NOH01PRICON-02, Primary Containment Structure - p.21
Tech Spec - bases 3.5.3 and 3.6.2

Justification

References during Exam Figure showing LT-4805-1

A - INCORRECT - leak on high pressure side of tap will cause indicated level to read LOWER than actual.
B - INCORRECT - Bases for Suppression Pool level is either: Ensure a sufficient supply of water is available to the HPCI, CSS and LPCI systems - NOT the RCIC pump. OR to ensure primary containment pressure will NOT exceed design pressure during a primary system blowdown.
C - INCORRECT - leak on high pressure side of tap will cause indicated level to read LOWER than actual
D - CORRECT

Question Source New ☐ **Memory Level** ☑ **Comprehension Level**

Question History:

New 8/30
SXD - minor comments
MB - 9/24 - Made changes as requested
AF - OK
MB - 11/8 - OK - Both chose "C"

Question 82

Hope Creek SRO Exam - Nov 2005

☐ **RO**
☑ **SRO**

Tier # 1 **Group #** 1

Importance 3.6

HC Obj:

295018 Partial or Complete loss of CCW

G2.4.30 Knowledge of which events related to system operations/status should be reported to outside agencies

Question

Hope Creek is operating at 100% when a partial loss of Reactor Auxiliary Cooling System (RACS) flow to the Reactor Water Cleanup (RWCU) System Non-regenerative Heat Exchanger resulted in an automatic isolation of RWCU Inlet Outboard Isolation Valve HV-F004, due to RWCU Non-Regenerative Heat Exchanger discharge high temperature isolation signal. NO other isolation valves were actuated. Plant remains stable at 100% power.

Which of the following identifies a proper assessment of 10CFR50.72 , Notifications?

The event is:

A reportable per 10CFR50.72 within 1 hour.

B reportable per 10CFR50.72 within 4 hours.

C reportable per 10CFR50.72 within 8 hours.

D NOT reportable per 10CFR50.72.

Answer D

References 10CFR50.72
Hope Creek Event Classification Guide - Section 11

Justification

References during Exam ECG Section 11
Tech Bases for 11.3.3

A - INCORRECT - Did NOT cause a deviation from Tech Specs.
B - INCORRECT - An RPS actuation was NOT initiated as a result of this signal.
C - INCORRECT - Only 1 system affected, NOT required to be reported.
D - CORRECT - This event is NOT reportable per 10CFR50.72 as item (b)(3)(iv)(B)(2) requires containment isolation signals affecting more than 1 system. This signal only affects 1 system.

Question Source New ☐ **Memory Level** ☑ **Comprehension Level**

Question History:

New 9/21
SXD - OK
AF - minor comments, wants tech bases for ECG 11.3.3
MB - Made minor editorial changes, SXD to resolve giving the tech bases
MB - 11/8 - OK

Question	83				
☐ **RO** ☑ **SRO**		**Tier #** 1	**Group #** 2	**HC Obj:**	NOH01MSTEAMC-02 - OBJ R12
		Importance	3.7		

295009 — Low Reactor Water Level / 2

AA2.02 — Ability to determine and interpret the following as they apply to Low Reactor Water Level (CFR: 41.10/ 43.5 / 45.13) — Steam flow/ feed flow mismatch

Question

Hope Creek is operating at 75% power to remove the 6A Feedwater Heater from service due to a problem on the Bleeder trip valve with the following conditions:

- Feedwater control is in 3 element control
- A Steam Flow indicates - 2.5 E6 lbs/hr
- B Steam Flow indicates - 2.5 E6 lbs/hr
- C Steam Flow indicates - 2.5 E6 lbs/hr
- D Steam Flow indicates - 2.5 E6 lbs/hr
- FW flow (N001A) indicates - 5.0 E6 lbs/hr
- FW flow (N001B) indicates - 5.0 E6 lbs/hr
- Reactor Water level - Normal at 35" stable
- Reactor Pressure - 1000 psig stable
- Generator MW - 750 MW
- Suppression Pool Temperature - 87°F

An event occurs.
1 Minute after event initiation the following conditions are observed:

- A Steam flow indicates - 1.7 E6 lbs/hr
- B Steam flow indicates - 2.5 E6 lbs/hr
- C Steam flow indicates - 2.5 E6 lbs/hr
- D Steam flow indicates - 2.5 E6 lbs/hr
- FW flow (N001A) indicates - 5.0 E6 lbs/hr
- FW flow (N001B) indicates - 5.0 E6 lbs/hr
- Reactor Water level is 38" and lowering slowly
- Reactor Pressure - 990 psig stable
- Generator MW - 670 MW
- Suppression Pool Temperature - 89°F

Based on the above conditions, what event has happened and what procedure shall you direct the operators to respond to the event?

A "A" Steam line's input to Total Steam flow has partially failed causing Steam flow/Feed flow mismatch, go to procedure HC.OP-AR.ZZ-0007 window F-1, "DFCS ALARM/TRBL"

B "A" Main Turbine Stop Valve has failed closed, go to procedure HC.OP-AB.BOP-0002, MAIN TURBINE

C An SRV has opened on the "A" steam line, go to procedure HC.OP-AB.RPV-0006, SAFETY RELIEF VALVE

D The 6A Feedwater heater bleeder trip valve has failed, go to procedure HC.OP-AB.ZZ-0001, TRANSIENT PLANT CONDITIONS

Answer C

References HC.OP-AB.RPV-0006, Safety Relief Valve p.1
NOH01MSTEAMC-02, MAIN STEAM SYSTEM,

Justification

References during Exam None

A - INCORRECT - While "A" steam line's input to Total Steam flow could cause the difference in indicated Steam Flow, it would NOT cause Generator MW to decrease.
B - INCORRECT - While "A" Main stop valve failing closed would cause a decrease in MW, it would NOT cause Reactor pressure to decrease, it would increase.
C - CORRECT - A safety on "A" steam line would cause, "A"'s steam line flow to decrease, MW to decrease and Reactor Pressure to decrease.
D - INCORRECT - 6A's bleeder trip valve going closed would cause MW to go up NOT down.

Question Source New ☐ ***Memory Level*** ☑ ***Comprehension Level***

Question History:

SXD review 7/27 - LOD 1 - re-write
8/1 - re-wrote question - MB
AF - added bullets, changed values to HC numbers, on "C" changed safety to SRV.
MB - 8/24 - Made changes as requested
AF - Lots of Info, may want to take info out, minor comments
MB - 11/8 - added Suppression Pool Temperature
MB - 11/17 minor change to stem

Question	84

Hope Creek SRO Exam - Nov 2005

☐ RO
☑ SRO

Tier # 1 **Group #** 2

HC Obj: NOH01EO102P-00 - Obj R6

Importance 4

295010 High Drywell Pressure / 5

AG2.4.6 Knowledge of symptom based EOP mitigation strategies
(CFR: 41.10 / 43.5 / 45.13)

Question

Following a station blackout event the STA reports the following parameters to the CRS:

- RPV Level minus 35 inches
- Drywell temperature 330°F and rising 1°F/min
- Drywell pressure of 5 psig

Which of the following ACTIONS SHALL be taken and what is the REASON for the action?

A ACTION - Spray the Drywell
REASON - Convection cooling of the Drywell is needed to prevent over pressure condition in the drywell.

B ACTION - Spray the Drywell
REASON - Evaporative cooling of the Drywell is needed to prevent over pressure condition in the drywell.

C ACTION - Emergency Depressurize
REASON - Evaporative cooling would result in rapid Drywell pressure reduction to less than atmospheric and possible implosion of the Drywell.

D ACTION - Emergency Depressurize
REASON - Convection cooling would result in rapid Drywell pressure reduction to less than atmospheric and possible implosion of the Drywell.

Answer C

References
HC.OP-EO.ZZ-0102, flowchart and bases p. 9
NOH01EO102P-00, HC.OP-EO.ZZ-0102 PRIMARY CONTAINMENT CONTROL DRYWELL (TEMPERATURE / PRESSURE AND HYDROGEN)
INPO Question - 21160

Justification

References during Exam Flowchart E-0102 (minus entry conditions)

A - INCORRECT - Per the curve DWT-P the plant is in the UNSAFE region, therefore you do NOT want to initiate Drywell spray. If SRO miss calculates or mis-readings the Curve DWT-P they may think they are in the Safe region.
B - INCORRECT - see "A" above.
C - CORRECT - Since the operator cannot Spray the drywell, the only other option is to Emergency Depressurize, since DW temperature is approaching 340°F and you cannot use Drywell spray DWT-5.
D - INCORRECT - while the operator does wish to Blowdown, the Reason per the bases is that Evaporative cooling could result in Drywell pressure reducing to < 2 psig and causing a Drywell implosion.

Question Source Bank ☐ **Memory Level** ☑ **Comprehension Level**

Question History:
New 8/30
SXD - minor comments
MB - 8/24 - Made changes as requested
AF - minor comments
MB - Incorporated changes as requested.
MB - 11/8 minor changes to stem

Question 85

- [ ] **RO**
- [x] **SRO**

Tier # 1 **Group #** 2

Importance 3.8

HC Obj:

295012 High Drywell Temperature

AA2.01 Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE : (CFR: 41.10 / 43.5 / 45.13) Drywell temperature

Question

Which one of the following identifies the bases for the Drywell Average Air Temperature Limiting Condition for Operation (LCO)?

In the event of a DBA, initial drywell average air temperature is assumed to be less than or equal to:

A 135°F so that the resultant peak accident temperature is maintained below 300°F during main steam line break conditions and is consistent with the safety analysis.

B 135°F, so that the containment peak air temperature does NOT exceed the design temperature of 340°F during LOCA conditions and is consistent with the safety analysis.

C 150°F so that the resultant peak accident temperature is maintained below 300°F during main steam line break conditions and is consistent with the safety analysis.

D 150°F so that the containment peak air temperature does NOT exceed the design temperature of 340°F during LOCA conditions and is consistent with the safety analysis.

Answer B

References Tech Specs 3.6.1.7 and bases

Justification

References during Exam None

A - INCORRECT - maintain peak temperature < 340°F NOT 300°F and accident is LOCA vs. Main Steam line break.
B - CORRECT per bases of 3.6.1.7
C - INCORRECT - temperature is 135°F vs. 150°F and see "A" above
D - INCORRECT - temperature is 135°F vs. 150°F

Question Source New

- [x] **Memory Level**
- [ ] **Comprehension Level**

Question History:

New 9/21
SXD - OK
AF - OK
MB - 11/8 - fixed typo

Question 86

☐ **RO**
☑ **SRO**

Tier # 2 **Group #** 1

Importance 3.3

HC Obj:

206000 HPCI

G2.1.14 Knowledge of system status criteria which require the notification of plant personnel. (CFR: 43.5 / 45.12)

Question

Hope Creek is in a Startup, you are the Operations Field Supervisor performing a Secondary Containment inspection when you discover a TMOD tag on some temporary instrumentation connected to the steam piping on the HPCI turbine. The TMOD tag is inside a contaminated area and is damaged and unreadable.

Who are you REQUIRED to notify to correct the condition in accordance with NC.DE-AP.ZZ-0030(Q), Control of Temporary Modifications?

A The Control Room Supervisor

B The Shift Manager

C The Responsible Engineer

D Duty Radiation Protection Technician

Answer C

References NC.DE-AP.ZZ-0030(Q) - CONTROL OF TEMPORARY MODIFICATIONS, p. 19, Section 5.7.1

Justification

References during Exam None

A - INCORRECT - Procedure requires you to notify the Responsible Engineer
B - INCORRECT - See "A"
C - CORRECT - See "A"
D - INCORRECT - See "A"

Question Source New ☑ **Memory Level** ☐ **Comprehension Level**

Question History:

New 8/31
SXD - K/A mismatch - NOT notifying outside agencies, notify Plant personnel
MB - 9/27 - Wrote new Question
SXD - OK
AF - OK
MB - 11/8 - minor editorial changes

Question	87				Hope Creek SRO Exam - Nov 2005
☐ RO	Tier #	2	Group #	1	HC Obj: CSSYS0E013
☑ SRO	Importance		3.2		

209001 LPCS

A2.02 Ability to (a) predict the impacts of the following on the LPCS and (b) based on those predictions, use procedures to correct control or mitigate the consequences of those abnormal operation (CFR: 41.5/ 43.5/ 45.3/ 45.13) Valve closures

Question

A transient has occurred on Hope Creek.

- Drywell pressure peaked at 4 psig
- Drywell pressure is now 1 psig and steady
- RPV level is 18" and dropping
- RPV pressure is 230 psig and lowering
- "A" Core Spray pump is running
- "C" Core Spray pump has tripped
- HV-F005A CORE SPRAY INBOARD ISOLATION MOV was CLOSED to terminate injection from the "A" Core Spray System
- HV-F004A CSS Loop Upstream Injection valve is OPEN

In accordance with the guidance provided in HC.OP-SO.BE-0001, Core Spray System Operation, to raise RPV water level using the "A" Core Spray pump, the CRS shall direct the RO to:

A Throttle HV-F005A from the control room to control RPV level

B Throttle HV-F005A locally to control RPV level

C Fully open and then fully close HV-F005A from the control room to control RPV level

D Fully open and then fully close HV-F005A locally to control RPV level

Answer B

References NOHO1CSSYS0-01, Core Spray System p. 18
HC.OP-SO.BE-0001, Core Spray System Operation, p. 3 and 5

Justification

References during Exam None

A - INCORRECT - can't throttle F005A from control room per Procedure SO.BE-0001
B - CORRECT - HV-F005A should be throttled to limit pump flow and Throttling of HV-F005A can only be performed locally.
C - INCORRECT - don't want to fully open F005A with only one pump running.
D - INCORRECT - don't want to fully open F005A with only one pump running

Question Source New ☐ Memory Level ☑ Comprehension Level

Question History:

New 8/30
SXD - check to see if answer is required by procedure.
MB - Wrote new question
SXD - OK
RJC - are actions being asked direct the RO to, put in accordance with "add procedure"
MB - Added procedure guidance
AF - K/A mismatch
MB - reworded distractors for clarity
MB - 12/2 changed distractors C & D to make them clearly incorrect.

Question	88

- [ ] RO
- [x] SRO

Tier # 2 **Group #** 1

HC Obj: NOH01APRM00-01, Obj R9

Importance 3.7

215005 APRM / LPRM

A2.02 Ability to (a) predict the impacts of the following on the APRM/ LPRM and (b) based on those predictions, use procedures to correct control or mitigate the consequences of those abnormal operation (CFR: 41.5/ 43.5/ 45.3/ 45.13) Upscale or downscale trips.

Question

A reactor startup is in progress at 9% power, OPCON 2, when the Startup Level Control Valve LV-1785 fails FULL OPEN with the following results:

- FULL SCRAM
- RO reports 4 rods at positions between 04 and 08
- APRMs are DOWNSCALE
- RPV level fell to a low of 15" and is now slowing rising
- The PO is able to control the Startup Level Control Valve LV-1785 in MANUAL
- RPV Pressure is 900 psig and slowly trending down.

Which of the following is the CAUSE of the SCRAM?
What DIRECTION is required?

A CAUSE - RPV High Water Level
DIRECTION - RESET the SCRAM and insert control rods per HC.OP-AB.ZZ-0000

B CAUSE - APRM Upscale
DIRECTION - RESET the SCRAM and insert control rods per HC.OP-AB.ZZ-0000

C CAUSE - RPV High Water Level
DIRECTION - ENTER HC.OP-EO.ZZ-0101A, determine a success path and insert control rods

D CAUSE - APRM Upscale
DIRECTION - ENTER HC.OP-EO.ZZ-0101A, determine a success path and insert control rods

Answer D

References
INPO Question 25648
HC.OP-AB.ZZ-0000, step S-8
HC.OP-EO.ZZ-0101 - RPV control - Entry conditions
NOH01APRM00-01, Average Power Range Monitoring System - p. 40

Justification

References during Exam None

A - INCORRECT - RPV High Water Level does NOT give a SCRAM, also based on RPV level of 15" and rising, RPV High Water level setpoint was NEVER reached.
B - INCORRECT - APRM upscale trip was received due to insertion of cold water causing APRM's to rise to 12-15% and generate a SCRAM, since NO E-101 parameters have been reached the proper course is to enter AB-000.
C - INCORRECT - Should NOT enter EO-101A since NO EO-101 parameters have been met.
D - CORRECT - Should enter EO-101A since it is not certain that the reactor is shutdown from all conditions.

Question Source Bank

- [ ] Memory Level
- [x] Comprehension Level

Question History:

New 8/31
SXD - circled OPCON 2, make sure 9% is still OPCON 2
MB - 9/27 - Per IO-0003, Operator is to withdraw control rods to 7-10% PRIOR To placing Mode Switch to RUN, therefore Plant could be at 9% and NOT be in RUN.
SXD - OK
AF - correct answer D
MB - SXD to look at.
MB - 11/8 - OK

Question 89

☐ **RO**
☑ **SRO**

Tier # 2 **Group #** 1
Importance 4

HC Obj:

259002 Reactor Water Level Control

G2.1.33 Ability to recognize indications for system operating parameters which are entry level conditions for technical specifications. (CFR: 43.2 / 43.3 / 45.3)

Question

With the plant at 100% Power, I&C reports to you that LT-N080A, RPV Low Level to NS4 ISLN and RPS Trip Logic has failed it's Quarterly surveillance.
LT-N080C is also out of service and in the tripped condition.

Given Tech Spec section 3.3, Instrumentation.

Assuming NO other instruments are out of service and that LT-N080A CANNOT be repaired.

When is the plant REQUIRED to be in HOT SHUTDOWN?

A Within 7 hours (you are in 3.0.3).

B Within 12 hours.

C Within 7 days + 12 hours.

D HOT SHUTDOWN is NOT REQUIRED, provided "A" channel is placed in tripped condition within 12 hours.

Answer D

References Tech Spec 3.3
M-42 sht 2

Justification

References during Exam Tech Spec section 3.3

A - INCORRECT - since both channels are on 1 trip system you just need to place the trip system in trip within 12 hours.
B - INCORRECT - not in 3.0.3
C - INCORRECT - see "A" above
D - CORRECT - see "A" above

Question Source New ☐ **Memory Level** ☑ **Comprehension Level**

Question History:

New 8/31
SXD - add justification and references, check on Instrument #
MB - 9/27 - --Archie - How does Hope Creek determine which Instruments satisfy which tech Specs --, logs surv. Requirements for surveillances
AF - NO instrument will require you to shutdown, ask steve what his thought process was.
SXD - AF to provide 2 instruments that will cause a problem in tech Specs.
MB - 11/8 - OK

Question	90

Hope Creek SRO Exam - Nov 2005

☐ RO
☑ SRO

Tier # 2 **Group #** 1

Importance 3.7

HC Obj:

264000 EDGs

A2.08 Ability to (a) predict the impacts of the following on the EDGs and (b) based on those predictions, use procedures to correct control or mitigate the consequences of those abnormal operation (CFR: 41.5/ 43.5/ 45.3/ 45.13) Initiation of emergency generator room fire protection system.

Question

At 1000 on November 28th, with Hope Creek operating at 100%, you are performing a Post-Maintenance Run on the "A" Diesel Generator.

1 NEO, 1 Mechanical Supervisor and a Vendor Representative are at the Diesel observing the run.

At 1010, you observe the "A" Diesel Generator trip and receive a FIRE alarm from the "A" Diesel Generator room. The NEO calls you and informs you:

- he heard an explosion coming from the "A" Diesel Generator and it appears that the #8 cylinder has a hole in it.
- the Vendor Representative has been hit by a piece of metal and is bleeding on the floor
- the room is filling up with smoke

You dispatch the fire brigade and a medical team to the scene.
The NEO and the Mechanical Supervisor pull the injured Vendor Representative from the room and wait for medical.

At 1020, the Fire Brigade reports back that there is NO fire on the scene, however, the Vendor Representative has died from the injury.

The Mechanical Supervisor estimates that it will take 1 week to repair the "A" Diesel Generator

The plant has remained at 100% power throughout this event.

How soon is the first report to the NRC REQUIRED to be made?

A There are NO requirements to notify the NRC.

B within 1 hour.

C within 4 hours.

D within 8 hours.

Answer B

References ECG section 9.3, 9.2

Justification

References during Exam Event Classification Guide - section 9, section 11
bases for section 9 and 11

A - INCORRECT - There are requirements to notify the NRC within 1 hour for an explosion on site.
B - CORRECT - based on ECG 9.3.1, for an explosion in the protected area, you must declare an Unusual event and notify the NRC within 1 hour.
C - INCORRECT - Unit Shutdown to comply with Tech Specs and Reporting a Fatality are 4 hour reports, however you were required to notify them within 1 hour for the explosion.
D - INCORRECT - This would be required if the plant were in a seriously degraded condition, however the explosion on site is more restrictive.

Question Source New ☐ **Memory Level** ☑ **Comprehension Level**

Question History:

New 9/22
SXD - OK
AF - get out tech bases and ask Steve is a Loud bang from an Piece of equipment destroying itself is considered an explosion or should I just write explosion in stem
MB - changed to explosion.
MB - 11/8 - added bases to handouts, minor editorial change to stem

Question 91

Hope Creek SRO Exam - Nov 2005

☐ **RO**
☑ **SRO**

Tier # 2 **Group #** 2

Importance 3.3

HC Obj: CRDHYDE033

201001 CRD Hydraulic

G2.1.28 Knowledge of the purpose and function of major system components and controls. (CFR: 41.7)

Question

Hope Creek is at 80% power when a single event/malfunction occurred affecting the CRD system.

NO operator actions have been performed.
The operator observed the following indications BEFORE the event and AFTER the CRD System stabilized.

	BEFORE	AFTER
CRD flow controller - flow indications	63 gpm	25 gpm
CRD flow controller - setpoint	63 gpm	63 gpm
Cooling Water flow	63 gpm	25 gpm
Cooling Water Pressure	25 psid	5 psid
Drive Water Pressure	270 psid	50 psid
Charging Pressure	1475 psig	1650 psig

Which of the following is the cause of this event ___I___
Assuming NO Operator actions, what actions are REQUIRED by Tech Specs? ____II___

A I. The flow control valve failed closed.
II. Be in HOT SHUTDOWN within 12 hours

B I. The Stabilizing valves have failed closed.
II. Be in HOT SHUTDOWN within 12 hours

C I. The flow control valve failed closed.
II. NO Actions are required by Tech Specs because the Control Rods are still OPERABLE.

D I. The Stabilizing valves have failed closed.
II. NO Actions are required by Tech Specs because the Control Rods are still OPERABLE.

Answer C

References Tech Spec 3.1.3.1
NOH01CRDHYD-01, CONTROL ROD DRIVE HYDRAULICS, p.11

Justification

References during Exam Tech Spec 3.1.3.1

A - INCORRECT - Flow control valve failing closed will give the above indications. Flow Control valve failing closed will cause the Control Rods to be trippable but inop for causes other than being mechanically bound, however because more than 8 control rods are INOP need to be in HS within 12 hours per action c.
B - INCORRECT - Stablizing valves failing closed will NOT cause the above indications.
C - CORRECT - While the control valve is still failed close, the Control Rods are still considered OPERABLE because the operator can NOT move them.
D - INCORRECT - see "C" above

Question Source New ☐ **Memory Level** ☑ **Comprehension Level**

Question History:

SXD - Not SRO - re-write
MB - 9/27 - re-wrote question to make more SRO level
SXD - OK
AF - K/A mismatch
MB - 11/8 changed values in condition 2

Question	92

Hope Creek SRO Exam - Nov 2005

☐ RO
☑ SRO

Tier # 2 **Group #** 2

Importance 4

HC Obj:

202001 Recirculation

G2.1.23 Ability to perform specific system and integrated plant procedures during all modes of plant operation. (CFR: 45.2 /45.6)

Question

While operating at 100% power, the "A" Recirc pump trips on Bus differential.

Reactor power decreases to 55%. Core and Recirculation Loop flows are as follows:

- A Recirc Loop Flow is 0 gpm
- A Recirc Jet Pump flow is 7.0E 6 lbm/hr
- B Recirc Loop Flow is 20,000 gpm
- B Recirc Jet Pump flow is 31.0E 6 lbm/hr
- OPRM's are OPERABLE

What operator ACTION is required?
What Tech Spec(s) need to be entered to address this condition?

A ACTION - Insert Control Rods to exit the Power-Flow Operating Map REGION I.
Tech Spec - 3.4.1.1

B ACTION - Insert Control Rods to exit the Power-Flow Operating Map REGION I.
Tech Spec - 3.4.1.1 and 3.4.1.3

C ACTION - Insert Control Rods to exit the Power-Flow Operating Map REGION II.
Tech Spec - 3.4.1.1

D ACTION - Insert Control Rods to exit the Power-Flow Operating Map REGION II.
Tech Spec - 3.4.1.1 and 3.4.1.3

Answer A

References
INPO Question - 22668
Tech Spec Section 2
HC.OP-SO.BB-0002(Q), REACTOR RECIRCULATION SYSTEM OPERATION, Attachment 2

Justification

References during Exam
HC.OP-SO.BB-0002(Q), Attachment 2
TS 3.4.1.1 and 3.4.1.3

A - CORRECT - Based on the Core Flow and power given, the plant is in REGION I, since OPRM's are OPERABLE, Operator needs to insert rods to reduce power to clear APRM upscale alarms and exit Region I per RPV-0003.
B - INCORRECT - Don't need to enter 3.4.1.3
C - INCORRECT - Core flow and Power given place unit in REGION I, NOT REGION II
D - INCORRECT - Don't need to enter 3.4.1.3

Question Source Mod

☐ Memory Level ☑ Comprehension Level

Question History:

New 8/31
SXD - add references -
MB - Added references, modified question somewhat.
SXD - OK
AF - A, B both correct
MB - Changed distractors A and C
MB - 11/8 - changed Tech Spec Mechanisms to Tech Specs

Question 93

Hope Creek SRO Exam - Nov 2005

☐ **RO**

☑ **SRO**

Tier # 2 **Group #** 2

HC Obj: NOH01MNGEN0-02, Obj R11a

Importance 3.8

245000 Main Turbine Gen. / Aux.

A2.05 Ability to (a) predict the impacts of the following on the Main Turbine Gen. / Aux and (b) based on those predictions, use procedures to correct control or mitigate the consequences of those abnormal operation (CFR: 41.5/ 43.5/ 45.3/ 45.13) Generator trip

Question

Hope Creek is shutting down to repair a Recirc pump vibration problem. The plant is currently at 40% power when a complete loss of stator cooling water occurs. Assume all other generator conditions were normal prior to the loss of stator cooling water.

A. What is the expected plant response assuming NO operator actions?
B. What actions shall you direct?

A
A. Recirc pump runback to 30% (12 second T.D.), Turbine-Generator runbacks load to < 23.5%, Turbine and Generator automatically trip
B. Go to AB.ZZ-0000, Reactor SCRAM and perform actions as required.

B
A. Recirc pump runback to 45% (12 second T.D.), Turbine-Generator runbacks load to < 23.5%, Turbine and Generator automatically trip
B. Go to AB.ZZ-0000, Reactor SCRAM and perform actions as required.

C
A. Recirc pumps runback to 30% (12 second T.D.), Turbine-Generator runbacks load to < 23.5%, Turbine and Generator do NOT automatically trip
B. Go to AB.BOP-0002, Main Turbine and perform actions as required.

D
A. Recirc pump runback to 45% (12 second T.D.), Turbine-Generator runbacks load to < 23.5%, Turbine and Generator do NOT automatically trip
B. Go to AB.BOP-0002, Main Turbine and perform actions as required.

Answer C

References HC.OP-AB.BOP-0002, Main Turbine
NOH01STATWC-01, Stator Water Cooling system, p 20

Justification

References during Exam None

A - INCORRECT - with power < 25% NO automatic Reactor Scram will Occur or is required. Since power is < 7,055 Stator amps No automatic turbine trip occurs.
B - INCORRECT - Recirc pumps runback to 30% NOT 45%
C - CORRECT - Turbine should runback to < 23.5%, no automatic trips should occur, recirc pumps runback to 30% and CRS should enter AB.BOP-0002 to address this condition.
D - INCORRECT - Recirc pumps runback to 30% NOT 45%.

Question Source New ☐ **Memory Level** ☑ **Comprehension Level**

Question History:

New 9/22
SXD - OK
RJC - should/ shall
MB - Made changes as requested
MB - 11/8 - resample
MB - 11/10 after speaking with RJC decided to re-write question vs. re-sample - re-wrote question.

Question 94			**Hope Creek SRO Exam - Nov 2005**
☐ ***RO*** ☑ ***SRO***	***Tier #*** 3 ***Group #***	***Importance*** 4.4	***HC Obj:*** CRDHYDE013

2.1.7 Generic

Ability to evaluate plant performance and make operational judgments based on operating characteristics / reactor behavior / and instrument interpretation. (CFR: 43.5 / 45.12 / 45.13)

Question

Given the following

- A Reactor Scram occurred.
- There are still 20 rods at Position 48.

The following sequence of events takes place:

- Scram is reset
- ARI is reset.

Then, there is a break in the scram air header.

Which of the following methods shall you direct the RO to pursue in order to insert control rods?

A Direct individual scramming of Control Rods with SRI Switches locally.

B Direct operator to manually drive Control Rods.

C Attempt an additional manual scram.

D Direct de-energizing scram solenoids by removing the RPS fuses.

Answer B

References
Susquehanna Exam August 2003
NOH01CRDHYD-01, CONTROL ROD DRIVE HYDRAULICS
EO-0101A

Justification

References during Exam EOP Flowcharts with entry conditions blacked out

A - INCORRECT - Without air, scram inlet and outlet valves should already be open.
B - CORRECT -
C - INCORRECT - Without air, scram cannot be reset since the discharge vent and drain valves remain closed.
D - INCORRECT - Scram inlet/outlet valves already open on loss of air.

Question Source Bank ☐ ***Memory Level*** ☑ ***Comprehension Level***

Question History:

New 9/8
SXD - Need to add justification, seems to be from another plant, make Hope Creek
MB - 9/26 - changed to a different question
SXD -OK
RJC - Should/shall, add procedure
MB - added shall, can't find procedure reference because procedures don't seem to give specifics, procedure only says, perhaps Archie can provide specific procedure guidance
MB - Minor editorial change

Question 95

- [ ] RO
- [x] SRO

Tier # 3 **Group #**

Importance 2.9

HC Obj:

2.1.34 Generic

Ability to maintain primary and secondary plant chemistry within allowable limits (CFR: 41.10 / 43.5 / 45.12)

Question

The plant was operating at 20% power. Plant Chemistry reported to the Main Control Room the following chemistry parameters:

- Reactor pH	8.8
- Reactor Water conductivity	11 micromhos/cm
- Reactor Water chlorides	0.150 ppm

Six hours later the plant enters OPCON 2, Chemistry reports the following:

- Reactor pH	6.5
- Reactor Water conductivity	0.9 micromhos/cm
- Reactor Water chlorides	0.150 ppm

Assuming chemistry conditions remain constant from this point forward, which one of the following actions (if any) is appropriate for these plant conditions?

A NO actions are required.

B Be in OPCON 3 within 6 hours and OPCON 4 within 30 hours.

C Restore Chlorides to within spec within 18 hours or perform an engineering evaluation.

D Restore chlorides to within limits within 48 hours or be in OPCON 3 within the next 12 hours and OPCON 4 within the following 24 hours.

Answer D

References INPO Question 24577
UFSAR 5.2.3.2.2.2 and UFSAR Table 5.2-8

Justification

References during Exam UFSAR 5.2.3.2.2.2 and Table 5.2-8

A - INCORRECT - Chlorides are out of specification, need to be restored within 48 hours..
B - INCORRECT - plausible because based on given conditions for OPCON 2, plant chemistry would be in spec if plant returned to OPCON 1.
C - INCORRECT - plausible if only look at Action b.
D - CORRECT - Per Action c.2 with Chlorides out of spec in OPCON 2, 3 and 4 they must be restored to within limits within 48 hours or be in OPCON 3 within the next 12 hours and OPCON 4 within the following 24 hours.

Question Source Mod

- [ ] Memory Level
- [x] Comprehension Level

Question History:

SXD review 7/27 - Talk to licensee ensure correct answer is correct and once conductivity is < limit, they exit the condition and can return to power.
AF - OK
MB - changed distractors to make it clearer

Question 96

Hope Creek SRO Exam - Nov 2005

☐ **RO**
☑ **SRO**

Tier # 3 **Group #**

Importance 3.3

HC Obj:

2.2.20 Generic

Knowledge of the process for managing troubleshooting activities (CFR: 43.5 / 45.13)

Question

Which of the following condition(s) would REQUIRE Field Engineering to review a Troubleshooting Plan developed in accordance with SH.OP-AP.ZZ-0008, OPERATIONS TROUBLESHOOTING AND EVOLUTIONS PLAN DEVELOPMENT:

I. Equipment is NOT removed from service or tagged and presents a risk of tripping the plant either directly or as a result of causing a major plant transient. (Very High Risk)

II. Equipment is NOT removed from service or tagged. Could result in an unexpected load reduction, a plant transient, or a reportable event. Should NOT result in a reactor, turbine, or generator trip. (High Risk)

III. Equipment is NOT removed from service or tagged. Could have an effect on plant equipment but shall NOT present a risk of causing an unexpected load reduction, plant transient or reportable event. (Medium Risk)

IV. Equipment is removed from service or tagged such that troubleshooting or testing activities shall NOT adversely affect the operation or safety of the plant. (Low Risk)

A I only

B I and II only

C I, II, and III only

D I, II, III and IV

Answer B

References SH.OP-AP.ZZ-0008, OPERATIONS TROUBLESHOOTING AND EVOLUTIONS PLAN DEVELOPMENT, p. 6 and 9

Justification

References during Exam None

SH.OP-AP.ZZ-0008, OPERATIONS TROUBLESHOOTING AND EVOLUTIONS PLAN DEVELOPMENT states that Field Engineering SHALL review a troubleshooting plan if the plan is determined to be either HIGH RISK or VERY HIGH Risk. The 4 conditions presented are the 4 conditions outlined in SHOP-8, I= Very High Risk, II= High Risk, III= Medium Risk, IV = Low Risk
A - INCORRECT - both High Risk and VERY High Risk must be evaluated
B - CORRECT
C - INCORRECT - Medium Risk does NOT need to be evaluated
D - INCORRECT - Medium Risk and Low Risk do NOT need to be evaluated

Question Source New ☑ **Memory Level** ☐ **Comprehension Level**

Question History:

New 9/9
SXD - confusing - put in High Risk, Very High Risk, etc.
MB - Changed 9/26
SXD - Change back to old way to make more difficult
MB - 10/3 Changed back to old way
SXD - OK
AF - OK
Val - give shop-8
MB - 11/8 can't give SHOP-8, added definitions High Risk, Very High Risk, etc.

Question	97

- [ ] RO
- [x] SRO

Tier # 3 **Group #**

Importance 3.5

Hope Creek SRO Exam - Nov 2005

HC Obj:

2.2.21 Generic

Knowledge of pre- and post-maintenance operability requirements (CFR: 43.2)

Question

You are the CRS of Hope Creek on Saturday night.

Maintenance has just completed adjusting the OPEN indication limit switch on valve BE-HV-F015B CS LOOP B TEST RET VLV.

Before declaring the valve OPERABLE, which of the following Test activities needs to be performed on BE-HV-F015B CS LOOP B TEST RET VLV in accordance with NC.MD-AP.ZZ-0050, Maintenance Testing Program Matrix:

A Valve Interlock Test

B External Leak Check

C Stroke Time Test

D Response Time Test

Answer C

References NC.NA-AP.ZZ-0050, Station Post Maintenace Testing, P.8
NC.MD-AP.ZZ-0050, Maintenance Testing Program Matrix, p 86-89

Justification

References during Exam NC.MD-AP.ZZ-0050

A - INCORRECT - Valve Interlock test is only applicable to A & B RHR Shutdown cooling valves
B - INCORRECT - External Leak Check (p90) is to be performed on an Air-Operated Valve for PMT for packing adjustment.
C. CORRECT - for Limit Switch adjustment for an MOV in the OPEN direction, the following tests need to be performed:
1. Functional Stroke *
5. Stroke Time Test *
9. Thermal Overload Bypass Surveillance *
D - INCORRECT, Response Time Test is a Test to determine the time interval from when a specified setpoint or condition is reached until a specified activity occurs. This is needed for an RT (Re-test) on a TRIP UNIT, Replacement (p.31)

Question Source New

- [ ] Memory Level
- [x] Comprehension Level

Question History:

New
SXD - No comment 9/23
AF - Not important, check if you have procedure, maybe re-write. Valve is broke what is re-test requirements. Give them procedure.
MB - 10/27 - re-wrote question
MB - 11/8 - changed valve to CS valve

Question 98

☐ **RO**
☑ **SRO**

Tier # 3 **Group #**

Importance 3.1

HC Obj:

2.3.4 Generic

Knowledge of radiation exposure limits and contamination control / including permissible levels in excess of those authorized. (CFR: 43.4 / 45.10)

Question

An NEO has been assigned to enter the Condenser Bay at power to investigate a steam leak. His current radiation history is as follows:

- Annual Exposure to date: 3280 mR TEDE
- Expected dose for this entry: 300 mR
- Highest Expected Dose Rate for the area: 600mR/hr
- NEO will be provided with continuous RP coverage during his entry

Which ONE of the following describes the REQUIRED action needed to complete the steam leak investigation per NAP 24, Radiation Protection Program, if any, based upon the above conditions:

A Dose Level Extension must be obtained prior to entry.

B Planned Special Exposure must be obtained.

C A Special RWP must be written.

D NO additional action required.

Answer D

References INPO Question 19298
NC.NA-AP.ZZ-0024, RADIATION PROTECTION PROGRAM - p. 27

Justification

References during Exam None

A - INCORRECT - since operator has already exceeded 3000 TEDE, the next extension is NOT required until he will exceed 4000 TEDE
B - INCORRECT - Planned Special Exposure is only required if dose is to exceed 10CFR20 limits, which this will NOT
C - INCORRECT - Special RWP is only required to be written for entry into a VHRA (>500 rads/hr), per 5.8.1. In addition, Section 5.11.3 of NAP-24, For work situations requiring immediate access, RP may substitute continuous coverage in lieu of an RWP..
D. - CORRECT - Already extended.

Question Source New ☐ **Memory Level** ☑ **Comprehension Level**

Question History:

SXD review 7/27 - too easy - LOD 1
AF - OK
SXD - Beef up
MB - 10/3 - Wrote new question
SXD - OK
AF - OK
MB - OK 11/8

Question 99

☐ **RO**
☑ **SRO**

Tier # 3 **Group #**

Importance 4

HC Obj:

2.4.22 Generic

Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations (CFR: 43.5 / 45.12)

Question

Hope Creek is experiencing an ATWS.

You are the CRS and you just directed the RO to inhibit the automatic initiation of the Automatic Depressurization System (ADS).

Which of the following is the reason why you directed the RO to inhibit the automatic initiation of ADS?

To prevent ________________

A A power excursion due to low pressure ECCS injection

B Large irregular neutron flux oscillations

C Exceeding 110°F Suppression Pool Temperature before boron injection

D Causing a Brittle fracture of the Reactor Vessel

Answer A

References
INPO Question 24595
HC.OP-EO.ZZ-0101A, ATWS – RPV CONTROL, P. 18

Justification

References during Exam None

A - CORRECT - Per EOP 101A bases - Further, rapid and uncontrolled injection of large amounts of relatively cold, unborated water from low pressure injection systems may occur as RPV pressure decreases to and below the shutoff heads of these pumps. Such an occurrence would quickly dilute in-core boron concentration and reduce reactor coolant temperature. When the reactor is NOT shutdown, or when the shutdown margin is small, sufficient positive reactivity might be added in this way to cause a reactor power excursion large enough to severely damage the core.
B - INCORRECT - ADS initiation would NOT cause flux oscillation but rather a rapid reduction in core power due to voids
C - INCORRECT - This may or may NOT be true but it is NOT the reason for inhibiting ADS
D - INCORRECT - While an ADS actuation will cause a Thermal Shock to the vessel, the vessel will be de-pressurized so you will NOT have a PTS concern

Question Source Mod

☐ **Memory Level** ☑ **Comprehension Level**

Question History:

SXD review 7/27 - OK
AF - on "D" changed to Brittle fracture.
MB - 8/24 - Made changes as requested
RJC - borderline SRO, get Archie to offer suggestions to make more SRO.
MB - 11/8 - OK

Question	100

Hope Creek SRO Exam - Nov 2005

☐ RO
☑ SRO

Tier # 3 **Group #**

Importance 3.9

HC Obj: INTEOPE001

2.4.14 Generic

Knowledge of general guidelines for EOP flowchart use (CFR: 43.5)

Question

Which of the following is correct concerning the use of Hope Creek's Emergency Operating Procedures (EOPs)?

A	If an EOP step cannot be performed, do not continue in the procedure until plant conditions allow completion of that step.
B	While executing an EOP, the operator should wait to see the effectiveness of an action step before continuing in the procedure.
C	If another entry condition for that EOP occurs, the operator should note the new entry condition and continue in the procedure.
D	If an action statement is followed by a decision step, an appropriate amount of time should be allowed to observe the effects of the action.

Answer D

References NOH01INTEOP-00, Introduction to EOP's, p.19 E.5
Bank Question Q56118

Justification

References during Exam None

A - INCORRECT
B - INCORRECT
C - INCORRECT
D - CORRECT per NOH01INTEOP-00

Question Source Bank ☑ **Memory Level** ☐ **Comprehension Level**

Question History:

New 9/9
SXD - add references
MB - 9/26 - Need references from Archie
AF - re-write - NOT linked to job - OSC, epep202, OSC duties
MB - re-wrote questions, need Archie to assist in verifying Question is OK.
MB - swapped out with bank question.
MB - 11/8 - resample, no Operator tasks associated with K/A

HOPE CREEK NRC EXAM - NOV/DEC 2005
RO EXAM KEY

1 B	**26** B
2 D	**27** A
3 D	**28** B
4 A	**29** B
5 B	**30** A
6 D	**31** A
7 B	**32** B
8 D	**33** A
9 B	**34** B
10 A	**35** D
11 B	**36** C
12 B	**37** D
13 D	**38** C
14 D	**39** A
15 A	**40** D
16 A	**41** D
17 A	**42** C
18 A	**43** D
19 A	**44** A
20 D	**45** B
21 B	**46** C
22 C	**47** A
23 A	**48** D
24 A	**49** A
25 B	**50** B

HOPE CREEK NRC EXAM - NOV/DEC 2005
RO EXAM KEY

51	D
52	B
53	D
54	A
55	A
56	C
57	C
58	D
59	C
60	A
61	A
62	C
63	D
64	C
65	C
66	C
67	A
68	B
69	D
70	C
71	A
72	B
73	C
74	C
75	A

HOPE CREEK NRC EXAM - NOV/DEC 2005
SRO EXAM KEY

1	B	**26**	B
2	D	**27**	A
3	D	**28**	B
4	A	**29**	B
5	B	**30**	A
6	D	**31**	A
7	B	**32**	B
8	D	**33**	A
9	B	**34**	B
10	A	**35**	D
11	B	**36**	C
12	B	**37**	D
13	D	**38**	C
14	D	**39**	A
15	A	**40**	D
16	A	**41**	D
17	A	**42**	C
18	A	**43**	D
19	A	**44**	A
20	D	**45**	B
21	B	**46**	C
22	C	**47**	A
23	A	**48**	D
24	A	**49**	A
25	B	**50**	B

HOPE CREEK NRC EXAM - NOV/DEC 2005
SRO EXAM KEY

51	D	**76**	B
52	B	**77**	B
53	D	**78**	A
54	A	**79**	C
55	A	**80**	B
56	C	**81**	D
57	C	**82**	D
58	D	**83**	C
59	C	**84**	C
60	A	**85**	B
61	A	**86**	C
62	C	**87**	B
63	D	**88**	D
64	C	**89**	D
65	C	**90**	B
66	C	**91**	C
67	A	**92**	A
68	B	**93**	C
69	D	**94**	B
70	C	**95**	D
71	A	**96**	B
72	B	**97**	C
73	C	**98**	D
74	C	**99**	A
75	A	**100**	D