Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	205000 K1.04	
	Importance Rating	2.7	

Knowledge of the physical connections and/or cause- effect relationships between SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) and the following: Fuel pool cooling assist: Plant-Specific

Proposed Question: Common 1

Which ONE of the following describes the Fuel Pool Cooling Assist flowpath through RHR and the effect of a lowering Skimmer Surge Tank level.

A. Flow is through any RHR Loop returning to the spent fuel pool.
As Skimmer Surge Tank level lowers, NPSH would be diminished for the RHR pump.

B. Flow is through ONLY RHR Loops A or B returning to the spent fuel pool.
As Skimmer Surge Tank level lowers, NPSH would be diminished for the RHR pump.

C. Flow is through ONLY RHR Loops A or B returning to the spent fuel pool.
As Skimmer Surge Tank level lowers, RHR pump NPSH would be unaffected.

D. Flow is through any RHR Loop returning to the spent fuel pool.
As Skimmer Surge Tank level lowers, RHR pump NPSH would be unaffected.

Proposed Answer: B

Explanation (Optional): B. Correct

Per FPCCS Lesson Plan, Page 15, Section III.B.7
Skimmer Surge Tanks Provide net positive suction head (NPSH) for the FPCCS pumps and a RHR pump when operating in the augmented fuel pool cooling mode.

Per FPCCS Lesson Plan, Page 24, Section III.C.1.c.2).
Flow is from the skimmer surge tank outlet to the RHR System (either Loop A or B). Return from the RHR System is to the spent fuel storage pool only via a set of dedicated diffusers.

B. Correct.

A. Incorrect. Only RHR Loop A & B is used
C. Incorrect. RHR pump NPSH is also affected
D. Incorrect. Only RHR Loop A & B may be used. RHR pump NPSH is also affected

Technical Reference(s) LP- NOH01FPCCOO-05 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: RHRSYSE003 (As available)

Question Source:
Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam ____

Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 X

Comments:

SOER 85-1

2) Leak detection is provided by a drain line located at a low point outside of the seal. The leakage would fill a level/flow switch identical to the type used on the spent fuel storage pool gates. The switches will then overflow to the drywell sump; which adds to the IDENTIFIED leak rate. The level switch causes the same overhead alarm as spent fuel storage pool gate leakage and drywell to reactor building seal leakage (Window D1-B5). The level switch also alarms an individual computer point.

OBJ. 3d

7. Skimmer Surge Tanks

a. Purpose

1) Provide net positive suction head (NPSH) for the FPCCS pumps and a RHR pump when operating in the augmented fuel pool cooling mode.

2) The skimmer surge tanks also act as a surge volume to accommodate water displaced by the largest piece of equipment that will be set in or removed from the storage pools.

b. Characteristics

There are two skimmer surge tanks, each having a capacity of 3750 gallons. The tanks are connected by an equalizing line which ensures the levels are maintained the same in both tanks. Each tank has a removable debris screen which prevents large objects from entering the FPCCS. The tanks are sealed in individual concrete rooms with access via plugs on the Refuel Floor.

OBJ. 8a, b

c. Normal make-up to the spent fuel storage pool is via a level control valve (LV-4660) from the Condensate Storage and Transfer System (AP). In the event of an emergency, make-up to the spent fuel storage pool can be supplied by the SSWS, Fire Water System, or RHR System.

There are service boxes on the Refuel Floor for demineralized water and from the Condensate Storage and Transfer System that can also be used as sources for make-up.

- Spent Fuel Storage Pool to Fuel Cask Storage Pit gate area
- Fuel Cask Storage Pit

b) Discharge connections

- Condensate Storage and Transfer System
- Reactor Cask Storage Pit
- Fuel Cask Storage Pit

c. RHR System Augmented Fuel Pool Cooling Mode

M-51, Sheet 1, Note 25

1) Whenever the RHR System is operated in the augmented fuel pool cooling mode, flow element FE-4462A(B) (containment spray flow) must be removed and replaced with a special flow restricting orifice specially designed for this mode of operation. This is due to the design capacity of the skimmer surge tank weirs being less than rated RHR pump flow.

2) ~~Flow~~ is from the skimmer surge tank outlet to the ~~RHR System (either Loop A or B)~~. ~~Return~~ from the RHR System is to the spent fuel storage pool only via a set of dedicated diffusers. These diffusers are equipped with vacuum breakers.

OBJ. 7b

2. Torus Water Cleanup

The Torus Water Cleanup pump takes a suction from the torus through a strainer to keep large material from entering the pump suction. The water then passes through two containment isolation valves (HV-4680 and HV-4681) to the pump. The pump discharge is directed to the filter-demineralizers through Reactor Building isolation valve HV-4656. From the filter-demineralizers, water flows back into the Reactor Building and through isolation valve HV-4663. Next, flow is directed back to the torus through two containment isolation valves (HV-4679 and HV-4652).

OBJ. 8a, b

3. Spent Fuel Storage Pool Level

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	218000 K1.02	
	Importance Rating	4.0	

Knowledge of the physical connections and/or cause-effect relationships between AUTOMATIC DEPRESSURIZATION SYSTEM and the following: Low pressure Core Spray

Proposed Question: Common 2

Which one of the following describes the ADS response to the ECCS pump discharge pressure permissive contact opening during the period of time the 105 second Delay Timer is timing out? (Assume all other permissives continue to be met and NO operator actions are taken).

The Delay Timer:

A. stops and ADS actuation will NOT occur even if the ECCS pump discharge pressure permissive contact is closed.

B. continues to time out and the ADS actuation will immediately occur when the ECCS pump discharge pressure permissive contact is closed.

C. de-energizes, resets to zero, then when the ECCS pump discharge pressure permissive is met, the Delay Timer starts a second cycle.

D. stops until the ECCS pump discharge pressure permissive contact is closed at which time ADS will initiate after the Delay Timer completes the cycle.

Proposed Answer: B

Explanation (Optional): IAW SN-0001:

3.3.1. The following signals (both Sub Channels B and F (or D and H) need to be energized) auto initiates ADS
1. Drywell Pressure (1.68 psig) (Seal-In)
OR 5 minute timer times out (for line breaks outside drywell) AND
2. Level 1 (-129 inches) AND
3. Confirmatory Level 3 (12.5 inches) AND
4. 105 second time delay AND
5. Core Spray discharge pressure 145 psig
OR RHR Pumps discharge pressure 125 psig.

Once the 105 second timer times out Core Spray and RHR pump discharge pressure is evaluated. ADS is actuated if the discharge pressure permissives are met.

B. Correct - LP ECCS status is evaluated after the 105 sec timer times out.

A. Incorrect - the 105 sec timer does not stop
C. Incorrect - there is no second cycle for the 105 second timer
D. Incorrect - the 105 sec timer does not stop

Technical Reference(s) HC.OP-SO.SN-0001 section 3.3.1 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ADSSYSE002 (As available)

Question Source: Bank # 53287
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

Comments:

3.2.11. **OBSERVE** the Safety Relief Valve Acoustic Monitor operability requirements IAW T/S 3.4.2.1. ____

3.2.12. **OBSERVE** the ADS requirements (of ECCS) IAW T/S 3.5.1. ____

3.2.13. **OBSERVE** the Safety/Relief Valve and Safety/Relief Valve Low-Low Set Function operability requirements IAW T/S 3.4.2.2. ____

3.2.14. **OBSERVE** the Safety/Relief Valve position indication requirements IAW T/S 3.3.7.4 and 3.3.7.5. ____

3.2.15. SRV Low-Low Set Function setpoints: ____

PSV-F013H	Open at 1047 psig; Close at 905 psig
PSV-F013P	Open at 1047 psig; Close at 935 psig
PSV-F013H	Subsequent opening 1017 psig

3.3 **Interlocks**

3.3.1. The following signals (both Sub Channels B and F (or D and H) need to be energized) auto initiates ADS: ____

A. For Channel B (or D) ____

1. Drywell Pressure (1.68 psig) (Seal-In) OR 5 minute timer times out (for line breaks outside drywell)

   AND

2. Level 1 (-129 inches)

   AND

3. Confirmatory Level 3 (12.5 inches)

   AND

4. 105 second time delay

   AND

5. Core Spray discharge pressure 145 psig OR RHR Pumps discharge pressure 125 psig.

(continued on next page)

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215004 K2.01	
	Importance Rating	2.6	

Knowledge of electrical power supplies to the following: SRM channels/detectors.

Proposed Question: Common 3

Given the following conditions:

- The plant is performing a startup
- All equipment is operable
- The RPS shorting links are installed

Then, the SRM 'A' drawer loses power.

Which one of the following describes what supplies the SRM 'A' drawer and what is the plant response?

A. 125 VDC
A Reactor Scram and rod block will occur.

B. +24VDC.
A Reactor Scram and rod block will occur.

C. +24VDC.
A rod block will occur. A Reactor Scram will NOT occur.

D. 125 VDC
A rod block will occur. A Reactor Scram will NOT occur.

Proposed Answer: C

Explanation (Optional):
IAW RPS Lesson Plan NOH01RPS00C-05 - An SRM INOP condition will not initiate a Scram signal. [CR 960218138]

IAW SRM Lesson Plan – Section VII.B.2.a. - The +24 VDC supplies the detector HVPS. Loss of this power will result in a loss of the HVPS and generate a channel INOPERATIVE trip.

Loss of ±24 VDC - The +24 VDC supplies the detector HVPS. Loss of this power will result in a loss of the HVPS and generate a channel INOPERATIVE trip.

Loss of 1BJ484 - Results in a loss of control and indications for the SRMS on 10C651. The SELECT, POSITION STATUS, POWER ON, DRIVING IN/OUT circuitry is lost.

IAW Section V.A.2.- ± 24 VDC Power System - supplies detector polarizing voltage and SRM logic modules.

C. Correct. – the loss of the 24 VDC will cause a loss of SRM logic modules and an inop trip and a withdraw block. No scram signal will occur

A. Incorrect. - +24 VDC is the power supply to the SRM HVPS ,A scram signal will not occur,
B. Incorrect. – A scram signal will not occur
D. Incorrect. – +24 VDC is the power supply to the SRM HVPS.

Technical Reference(s) LP - NOH01RPS00C-05 (RPS) (Attach if not previously provided)
LP- (SRM)

Proposed references to be provided to applicants during examination: None

Learning Objective: SRMSYSE013 (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

Comments:

#3

b. Fuel loading chambers used are substituted for normal in core SRM detectors.

1) Mounted inside dummy fuel assemblies to facilitate their movement by the refueling bridge grapple.

2) Output signal connected to SRM pre-amplifier in place of normal detector signal.

3) When sufficient fuel assemblies have been loaded into the core to produce a background count rate of at least 3 cps, the normal SRM detectors may be reconnected and the fuel loading chambers removed.

Obj. 11

B. Abnormal/Emergency Operations

1. Review the following procedure relative to SRMS operation:

- OP-AB.IC-0004, Neutron Monitoring

Ensure symptoms (system response) and immediate operator actions are discussed.

2. SRM Malfunctions

Obj. 13

a. Loss of +24 VDC

The +24 VDC supplies the detector HVPS. Loss of this power will result in a loss of the HVPS and generate a channel INOPERATIVE trip.

Refer to Incident Report 86-067 in the Plant and Industry Event section.

b. Loss of 10Y202

The detector cannot be moved because there is no power to the drive motor.

c. Loss of 1BJ484

Results in a loss of control and indications for the SRMS on 10C651. The SELECT, POSITION STATUS, POWER ON, DRIVING IN/OUT circuitry is lost.

The detector cannot be moved.

C. Plant and Industry Events

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NOTE: An SRM INOP condition will not initiate a Scram signal. [CR 960218138]

e) SRM high flux scram setpoint: $2X10^5$ CPS

FIG 11,12

2) Intermediate Range Monitor: Protects against rapid increases in core neutron flux levels between the startup range and the power range during reactor startup and shutdown operations

a) 8 IRM channels (A thru H) provide upscale and inoperative scram inputs

b) IRM channels A, C, E & G input to RPS Trip System "A"; Channels B, D, F & H input to RPS Trip System "B"

c) Logic is 1-out-of-4-twice

- 1-out-of-3-twice is minimum requirement (per technical specifications)
- Allows single IRM input to each RPS trip system to be bypassed

d) Trip inputs and setpoints

- IRM Hi-Hi (120/125 of scale)
- IRM INOP: mode switch not in OPERATE, detector high voltage low, module unplugged

3) Average Power Range Monitor: Initiate a scram when reactor core average power reaches or exceeds a limiting safety system setting.

a) Six APRM channels provide upscale and inoperative scram inputs

b) APRM channels A, C, E input to RPS A. Channels B, D, F input to RPS B

c) Logic is 1-out-of-3-twice

- 1-out-of-2-twice is minimum requirement

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	239002 K2.01	
	Importance Rating	2.8	

Knowledge of electrical power supplies to the following: SRV solenoids.

Proposed Question: Common 4

The plant is operating at rated power.

Which one of the following describes the effect of a loss of 125 VDC BD417, on the Automatic Depressurization System (ADS), if the required actuation setpoints were reached?

NOTE: Assume all other equipment is operable AND NO operator actions were taken to inhibit ADS

A. "A" ADS SRV pilot solenoids have lost power.
An ADS actuation would occur.

B. "A" ADS SRV pilot solenoids have lost power.
An ADS actuation would NOT occur.

C. "B" ADS SRV pilot solenoids have lost power.
An ADS actuation would occur.

D. "B" ADS SRV pilot solenoids have lost power.
An ADS actuation would NOT occur.

Proposed Answer: A

Explanation (Optional):

A. Correct

IAW ADS Lesson Plan NOH01ADSSYSC-03 Section V.A.3 - 125 VDC Class 1E Distribution System - The 125 VDC Class 1E Distribution System supplies electrical power to the ADS SRV pilot solenoids <u>and</u> the ADS logic channels.

ADS Channel B logic <u>and</u> the A ADS SRV pilot solenoids are powered from 1BD417.

ADS Channel D logic <u>and</u> the B ADS SRV pilot solenoids are powered from 1DD417.

IAW Section III.B.1.a.2)b) - Satisfaction of either ADS Channel B <u>or</u> ADS Channel D will result in the actuation of the ADS and the depressurization of the RPV. This ensures that a single failure will <u>neither</u> initiate <u>nor</u> inhibit the ADS function.

B. Incorrect. – Channel D would cause an actuation
C. Incorrect. - DD417 supplies the B ADS SRV solenoids
D. Incorrect. – DD417 supplies B ADS SRV solenoids, Channel D would cause an actuation

Technical Reference(s) LP NOH01ADSSYSC-03 (ADS) (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: ADSSYSE007 (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

Comments:

#4
NRC

V. SYSTEM INTERRELATIONS

A. Support Systems

OBJ. 7a

1. RHR and Core Spray Systems

   a. Pressure switches located in the RHR and/or Core Spray pump discharge headers provide permissive signals to the ADS (automatic) initiation logic subchannels.

      1) Pumps are determined to be operating as sensed by pump discharge pressure being above the setpoint.

         a) Core Spray pumps: 145 psig

         b) RHR pumps: 125 psig

      2) The pump discharge pressure permissive logic requires a minimum of one Core Spray loop (two operating pumps) <u>or</u> one RHR pump.

   b. Core Spray System logic channel trip units provide RPV level <u>and</u> drywell pressure trip outputs to the ADS initiation logic.

OBJ. 7b

2. PCIG System

   The PCIG System supplies compressed (nitrogen) gas for pnuematic operation of the SRV pilot valves.

OBJ. 7c

3. 125 VDC Class 1E Distribution System

   The 125 VDC Class 1E Distribution System supplies electrical power to the ADS SRV pilot solenoids <u>and</u> the ADS logic channels.

   a. ADS Channel B logic <u>and</u> the A ADS SRV pilot solenoids are powered from 1BD417.

   b. ADS Channel D logic <u>and</u> the B ADS SRV pilot solenoids are powered from 1DD417.

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source of makeup water is available to the RPV prior to depressurization.

4. The ADS energizes the pilot solenoids of 5 of the 14 Nuclear Pressure Relief System Safety Relief Valves (SRVs) to depressurize the RPV to the suppression pool.

Fig. 1
OBJ. 3

a. The SRVs associated with the ADS are PSV-F013A,B,C,D and E.

1) The safety lift setpoint of each ADS SRV is 1130 psig, which is the highest lift setpoint of the 14 SRVs.

2) The other 9 SRVs have a safety lift setpoint of either 1108 psig or 1120 psig.

3) The ADS SRVs have the highest safety lift setpoint to minimize valve actuations, thereby ensuring maximum reliability and availability for the ADS function.

b. When the ADS energizes the SRV pilot solenoids, Primary Containment Instrument Gas (PCIG) pressure is admitted to the pneumatic actuator of the SRV pilot assembly, causing the valve to open.

<u>NOTE</u>: the function, description and operation of the SRVs and the supporting and interfacing components and systems are contained in the Main Steam System lesson plan NOH01MSTEAMC

B. Major Subsystems/Components

1. ADS Logic Design

a. Two divisions (Channels B and D) of ADS logic are provided:

1) Channel B is comprised of subchannels B <u>and</u> F.

2) Channel D is comprised of subchannels D <u>and</u> H

a) Both subchannels (B and F <u>or</u> D and H) within an ADS channel (division) must be satisfied for the associated ADS channel to initiate depressurization of the RPV. This logic arrangement prevents an inadvertent actuation of the ADS.

b) Satisfaction of either ADS Channel B <u>or</u> ADS Channel D will result in the actuation of the

ADS and the depressurization of the RPV. This ensures that a single failure will neither initiate nor inhibit the ADS function.

b. Channels A and C are not used by ADS.

1) Channels A and C are used by the HPCI System initiation logic.

2) The ADS is electrically separated from the HPCI System, including sensors and logic circuitry, thereby ensuring that no single failure affecting the HPCI System will impact the ADS, and vice versa

2. The majority of the ADS logic is mounted in two panels located in the Lower Relay Room: Panel 10C628 (Division 2, subchannels B and F) and Panel 10C631 (Division 4, subchannels D and H).

a. Associated ADS channel initiation logic and power fuses.

b. Low-Low Set relief logic (see Main Steam System Lesson Plan NOH01MSTEAMC).

c. SRV logic and control power fuses.

d. Not included are the process input trip units, which are mounted on the associated ECCS control panels:

1) Panel 10C618 (Division 2, subchannels B and F).

2) Panel 10C640 (Division 4, subchannels D and H).

3. Drywell pressure input

OBJ. 2
Fig. 7

a. Each subchannel (B,F,D,H) contains a single high drywell pressure input, which will energize relay K1(B, F, D, H).

1) Channel B

a) Subchannel B: transmitter N094B, trip unit N694B on Panel 10C618.

b) Subchannel F: transmitter N094F, trip unit N694F on Panel 10C618.

V. SYSTEM INTERRELATIONS

A. Support Systems

OBJ. 7a

1. RHR and Core Spray Systems

   a. Pressure switches located in the RHR and/or Core Spray pump discharge headers provide permissive signals to the ADS (automatic) initiation logic subchannels.

      1) Pumps are determined to be operating as sensed by pump discharge pressure being above the setpoint.

         a) Core Spray pumps: 145 psig

         b) RHR pumps: 125 psig

      2) The pump discharge pressure permissive logic requires a minimum of one Core Spray loop (two operating pumps) <u>or</u> one RHR pump.

   b. Core Spray System logic channel trip units provide RPV level <u>and</u> drywell pressure trip outputs to the ADS initiation logic.

OBJ. 7b

2. PCIG System

   The PCIG System supplies compressed (nitrogen) gas for pnuematic operation of the SRV pilot valves.

OBJ. 7c

3. 125 VDC Class 1E Distribution System

   The 125 VDC Class 1E Distribution System supplies electrical power to the ADS SRV pilot solenoids <u>and</u> the ADS logic channels.

   a. ADS Channel B logic <u>and</u> the A ADS SRV pilot solenoids are powered from 1BD417.

   b. ADS Channel D logic <u>and</u> the B ADS SRV pilot solenoids are powered from 1DD417.

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Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	217000 K3.03	
	Importance Rating	3.5	

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

Proposed Question: Common 5

Following an automatic scram from 100% RTP due to closure of the MSIVs, a steam leak developed on the RCIC steam line in the pipe routing area (pipe chase) and temperature is currently 162 degrees and rising at 2 degrees per minute.

Which one of the following identifies the method(s) available to reduce RPV pressure and commence reactor cooldown 30 minutes from now?

A. SRVs ONLY.

B. RCIC & HPCI ONLY.

C. HPCI and SRVs ONLY.

D. SRVs, RCIC & HPCI.

Proposed Answer: C

Explanation (Optional):
IAW Lesson Plan (RCIC)
Section IV.E.2.a.7) – auto isolation - RCIC Steam Pipe Area High High Temperature; >160°F for 30 minutes.
Section IV.E.5.a.2) – RCIC Trip is caused by any RCIC Isoaltion

C. Correct – HPCI & RCIC steam line supplies are separate. SRVs are unaffected and available. RCIC will trip on isolation signal of >160 degrees for 30 minutes

A. Incorrect – HPCI would also be available
B. Incorrect - SRVs would also be available. RCIC would have tripped on the isolation signal
D. Incorrect – RCIC would have tripped on the isolation signal

Technical Reference(s) RCIC Lesson Plan (Attach if not previously provided)

NOH04RCIC00-05

Proposed references to be provided to applicants during examination: NONE

Learning Objective: RCIC00E021 (As available)

Question Source: Bank # INPO Bank, Susq 2002

Modified Bank # _____ (Note changes or attach parent)

New _____

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge _____

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

Comments:

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4) RCIC Room Ventilation High Differential Temperature; >70°F.

5) Equipment Area High Temperature; >160°F.

6) Torus Room High Temperature; >128°F for 30 minutes.

7) RCIC Steam Pipe Area High High Temperature; >160°F for 30 minutes.

b. At Panel 621 for Channel B (640 for D), switch S5B (S5D) in the bypass position removes all temperature inputs to the RCIC isolation.

c. Pushbutton S1B for Channel B (S1D for D channel) when depressed can bypass the 30 minute time delay. These are normally used for test purposes.

d. Automatic Isolation Logic Train Actuations are arranged as follows:

1) Isolation Logic Train "B", when actuated, performs the following:

a) Closes HV-F008 (Outboard Steam Supply Isolation Valve)

b) Trips closed HV-4282 (Turbine Trip Throttle Valve)

2) Isolation Logic Train "D", when actuated, performs the following:

a) Closes HV-F007 (Inboard Steam Supply Isolation Valve)

b) Closes HV-F076 (Inboard Steam Supply Line Warmup Valve)

c) Trips closed HV-4282 (Turbine Trip Throttle Valve)

e. When the isolation condition is cleared, the logic must be reset by taking the NORM/RESET keylock switch to the reset position. The switches for F007 and F008 must be matched with the valve position before valve movement is allowed.

3. RCIC System Manual Isolation

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a. Manual Isolation of the RCIC System is accomplished by depressing the "Isolation Logic 'B' Trip" Pushbutton

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1) 'B' Logic train is actuated which causes the closure of HV-F008 and the trip of HV-4282.

b. To enable the "Isolation Logic 'B' Trip" pushbutton, a RCIC Initiation signal MUST be sealed in.

c. Controls to manually actuate the "D" Isolation Logic Train are NOT provided.

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4. RCIC System Exhaust Line Vacuum Breaker Automatic Isolation.

a. Exhaust Line vacuum breaker isolation valves HV-F062 and HV-F084 will auto close on the following signals:

1) Low steam supply line pressure 64.5 psig

and

2) High drywell pressure 1.68 psig

b. Isolation Logic Train 'B' actuation closes HV-F062 (Outboard Vacuum Breaker Isolation Valve)

c. Isolation Logic Train 'D' actuation closes HV-F084 (Inboard Vacuum Breaker Isolation Valve)

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Obj. 7.

Fig. 13

5. RCIC Turbine Trip

a. The RCIC Turbine Trip Signals are as follows:

1) Turbine Trip Pushbutton
2) RCIC Isolation (Man or Auto).
3) Pump Suction Low Pressure (15" Hg Vac).
4) Turbine Exhaust High Pressure (≥50 psig).
5) Mechanical Overspeed (121% speed).
6) High Reactor Water Level/Level 8 (+ 54 inches)

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Upon a turbine trip, either the turbine trip throttle valve 4282 closes (items 1-5 above) or the steam supply valve F045 closes (for item 6).

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Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	206000 K3.03	
	Importance Rating	3.4	

Knowledge of the effect that a loss or malfunction of the HIGH PRESSURE COOLANT INJECTION SYSTEM will have on following: Suppression pool level control: BWR-2,3,4.

Proposed Question: Common 6

HPCI is in full flow test for RPV pressure control following a reactor scram. RPV level has recovered to 35 inches and is stable. All other containment parameters are normal.

Then, logic control power is lost to valve BJ-HV-F042 "PMP SUCT FROM SUPP CHB"

Which one of the following describes the effect of the loss?

A. CST level would be adversely affected because the valve would NOT auto-open on a HIGH CST level signal.

B. CST level would be adversely affected because the valve would NOT auto-close on a LOW CST level signal.

C. Suppression Pool level would be adversely affected because the valve would NOT auto-close on a LOW Suppression Pool level signal.

D. Suppression Pool level would be adversely affected because the valve would NOT auto-open on a HIGH Suppression Pool level signal.

Proposed Answer: D

Explanation (Optional):

IAW HC.OP-BJ-0001 - BJ-HV-F042 PMP SUCT FROM SUPP CHB-Auto closes on HPCI Div 1 Isolation signal (K51A). **Auto opens on CST low level OR Suppression Chamber high** level (K42), IF BJ-HV-F042 handswitch is not in AUTO OPEN OVRD AND no HPCI Div 1 Isolation signal (K51A). Opens manually IF no HPCI Div 1 Isolation signal (K51A).

D. Correct – valve would not auto open on SP level high

A. Incorrect – valve auto open on low CST level

B. Incorrect – valve only auto closes on an isolation signal

C. Incorrect – valve does not auto close on low SP level

Technical Reference(s) HC.OP-BJ-0001 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: HPCI00E012 (As available)

Question Source: Bank # 53735
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 X

Comments:

#6

3.3.4. HPCI valve interlocks are as follows:

- FD-HV-F002 INBD HPCI STM ISLN MOV Auto Closes on HPCI Div 3 Isolation signal (K50C). Opens manually if no HPCI Div 3 Isolation signal (K50C). IF Isolation signal was sealed in (K92) with FD-HV-F002 handswitch in OPEN THEN handswitch must be placed in CLOSE THEN OPEN. ____

- FD-HV-F003 OUTB HPCI STM ISLN MOV Auto closes on HPCI Div 1 Isolation signal (K50A). Opens manually if no HPCI Div 1 Isolation signal (K50A). IF Isolation signal was sealed in (K93) with FD-HV-F003 handswitch in OPEN THEN handswitch must be placed in CLOSE THEN OPEN. ____

- FD-HV-F001 HPCI TURB STM SPLY Auto opens on HPCI Initiation (K33), IF FD-HV-F071 TURB EXH ISLN is full open. Opens manually IF FD-HV-F071 TURB EXH ISLN is full open. ____

- FD-HV-F100 HPCI W/U VLV Auto closes on HPCI Div 3 Isolation signal (K50C & K52C). Opens manually IF no HPCI Div 3 Isolation signal (K50C). ____

- BJ-HV-F004 PMP SUCT FROM CST Auto closes IF BJ-HV-F042 PMP SUCT FROM SUPP CHB is full open (K43). Auto opens IF BJ-HV-F042 PMP SUCT FROM SUPP CHB is not full open (K43) AND HPCI Initiation (K33). ____

- BJ-HV-F042 PMP SUCT FROM SUPP CHB Auto closes on HPCI Div 1 Isolation signal (K51A). Auto opens on CST low level OR Suppression Chamber high level (K42), IF BJ-HV-F042 handswitch is not in AUTO OPEN OVRD AND no HPCI Div 1 Isolation signal (K51A). Opens manually IF no HPCI Div 1 Isolation signal (K51A). ____

- BJ-HV-F007 PMP DSCH ISLN Auto closes IF Test Selector switch is in HV-F007 position AND Test Jack installed (K66). Auto opens on HPCI Initiation (K32) AND not in test (K66). ____

- BJ-HV-F006 PMP DSCH TO CS ISLN Auto closes IF FD-HV-F001 HPCI TURB STM SPLY is full closed (K44)
  OR Test Selector switch is in HV-F006/HV-8278 (HV-F105) (AND Test Jack installed) (K65)
  OR FD-FV4880 HPCI TURB STOP VLV is full closed. Auto opens on HPCI Initiation (K90)
  AND FD-FV4880 HPCI TURB STOP VLV is not full closed (K39)
  AND FD-HV-F001 HPCI TURB STM SPLY is not full closed (K44) AND not in test (K65). ____

Continued on next page

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	300000 K4.02	
	Importance Rating	3.0	

Knowledge of (INSTRUMENT AIR SYSTEM) design feature(s) and or interlocks which provide for the following: Cross-over to other air systems

Proposed Question: Common 7

The plant is operating at rated power when the following annunciators are received:

- A2-A1, INST AIR HEADER A PRESSURE LO
- A2-A2, INST AIR HEADER B PRESSURE LO
- A2-B1, COMPRESSED AIR SYSTEM TROUBLE
- A2-B2, COMPRESSED AIR PANEL 00C188

Current air pressures are:

- Service Air pressure is 87 psig.
- Instrument air pressure at the Emergency Instrument Air Receiver is 82 psig.
- Instrument air pressure at the Instrument Air Receivers is 77 psig.

What is the configuration of the Service and Instrument Air System?

A. Instrument Air Dryer 1AF104 Isolation Valve, HV-11416, will be open.
The Standby Service Air Compressor will be running.

B. The Service Air Supply Header Isolation Valve, HV-7595, will be closed.
The Standby Service Air Compressor will be running.

C. Instrument Air Dryer 1AF104 Isolation Valve, HV-11416, will be closed.
The Standby Service Air Compressor will NOT be running.

D. The Service Air Supply Header Isolation Valve, HV-7595, will be closed.
The Standby Service Air Compressor will NOT be running.

Proposed Answer: A

Explanation (Optional):
IAW HC.OP-AB.COMP-0001 & Lesson Plan NOH01INSAIR-02 (Instrument air) – Section IV.C.1.g. – page 57 - As the loss of air event starts, there is very little effect on plant operation. As air pressure begins to decrease, some automatic actions will occur that will attempt to stop the loss of air.

- Standby Service Air Compressor starts (92 psig)
- Emergency Instrument Air Compressor starts (85 psig)
- Instrument Air Dryer 1AF104 isolation valve (HV-11416) opens (85 psig)
- Service Air Header Isolation Valve (HV-7595) closes (70 psig Instrument Air pressure)

A. Correct. The Emergency Instrument Air Compressor starts at 85#, HV-11416 opens at 85#

B. Incorrect - The Service Air Supply Header Isolation Valve, HV-7595 will be open
C. Incorrect - Instrument Air Dryer 1AF104 Isolation Valve, HV-11416 opens at 85#
D. Incorrect - The Service Air Supply Header Isolation Valve, HV-7595 closes at 70#, the Standby Service Air Compressor starts at 92#.

Technical Reference(s) HC.OP-AB.COMP-0001 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: INSAIRE006 (As available)

Question Source: Bank # 56927
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 X

Comments:

IMMEDIATE OPERATOR ACTIONS

CONDITION	ACTION
Trip of the Inservice Service Air Compressor Date/Time: _____	**PLACE** the out of service Service Air Compressor in service. (AB.ZZ-0001)

AUTOMATIC ACTIONS

IF	THEN
< 92 psig Service Air Pressure.	Standby Service Air Compressor Auto Start
≤ 85 psig Emergency Instrument Air Receiver Pressure	• EIAC Auto Start • RACS Demineralizers Isolate
≤ 85 psig Instrument Air Pressure OR Loss of Power to 1A-F-104	1-KBHV-11416 OPENS
≤ 70 psig Instrument Air Pressure	1-KAHV-7595 AUTO CLOSES

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	262001 K4.05	
	Importance Rating	3.4	

Knowledge of A.C. ELECTRICAL DISTRIBUTION design feature(s) and/or interlocks which provide for the following: Paralleling of A.C. sources (synchroscope).

Proposed Question: Common 8

A unit startup is in progress, the main generator is being synchronized to the grid. The following indications are observed:

- Keylocked SYNC SCOPE switch in the ON position
- Sync Scope and voltages indicate the first main generator output breaker is ready for closure

The operator depresses the BS2-6 CLOSE pushbutton, but the breaker fails to close.

Which condition prevented breaker closure?

A. The 52x60 Generator Disconnect is open.

B. The main generator exciter field breaker is open.

C. The SYNC CHECK ON pushbutton was NOT held depressed before depressing the CLOSE pushbutton.

D. The SYNC CHECK OFF pushbutton was NOT held depressed before depressing the CLOSE pushbutton.

Proposed Answer: D

Reference: HC.OP-SO.MA-0001 – Section 5.2.16. **PERFORM** the following to synchronize the Main Generator using Manual Load Control:

D. **PERFORM** the following (with Steps 2 thru 5 being performed in rapid succession):
1. WHEN SYNCHROSCOPE Pointer is at 5 minutes before 12 O'clock position,
PRESS AND **HOLD** SYNCH CHECK OFF push-button. _____
2. WHEN SYNCHROSCOPE Pointer is at 2 minutes before 12 O'clock position,
CLOSE BS 6-5 (BS-2-6) Breaker. ____

3. **RELEASE** the SYNCH CHECK OFF push-button.

D. Correct - The SYNC CHECK OFF pushbutton was not held depressed before depressing the CLOSE pushbutton. The OFF button is required to be held IAW HC.OP-SO.MA-0001.

A. Incorrect - The 52x60 Generator Disconnect is open. For the sync scope and voltages to indicate that the machine is properly synchronized the exciter field breaker and 52x60 must both be closed.

B. Incorrect - The SYNC CHECK ON pushbutton was not held depressed before depressing the CLOSE pushbutton. The OFF button is required to be held IAW HC.OP-SO.MA-0001.

C. Incorrect - The main generator exciter field breaker is open. For the sync scope and voltages to indicate that the machine is properly synchronized the exciter field breaker and 52x60 must both be closed.

Technical Reference(s) HC.OP-SO.MA-0001 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: MNPWR0E016 (As available)

Question Source: Bank # 56833
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 X

Comments:

CAUTION

After synchronization and closing a Generator Breaker, Load Setpoint should be increased to 30% IMMEDIATELY. If the following steps are not completed or expected indications received, the CRS should consider tripping the Main Turbine and ensuring the Generator is isolated from the 500 Kv System and the Turbine is coasting down.

5.2.16. **PERFORM** the following to synchronize the Main Generator using Manual Load Control:

A. **SELECT** [Control], [Speed-Load] ____

NOTE

Ramp Rate will indicate 20 %/min until breaker is closed.

Upon breaker closure, ramp rate will be 60%/min for 2 seconds (2 % load).

Ramp Rate will then go to setpoint entered.

B. **SELECT** Load Set, [Ramp Rate] AND **ENTER** 10 %/min ____

C. **SELECT** Load Set, Manual Adj.: [Raise] OR [Lower] until pointer on SYNCHROSCOPE is moving slowly in the FAST direction. (10C651D) ____

Continued on next page

5.2.16 (continued)

D. **PERFORM** the following (with Steps 2 thru 5 being performed in rapid succession):

1. WHEN SYNCHROSCOPE Pointer is at 5 minutes before 12 O'clock position, **PRESS** AND **HOLD** SYNCH CHECK OFF push-button. ____

2. WHEN SYNCHROSCOPE Pointer is at 2 minutes before 12 O'clock position, **CLOSE** BS 6-5 (BS-2-6) Breaker. ____

3. **RELEASE** the SYNCH CHECK OFF push-button. ____

4. IMMEDIATELY **SELECT** Load Set, : Setpoint AND **ENTER** 30 %. ____

5. **OBSERVE** Bypass Valves close as Generator loads to 30 % Load Set. ____

6. IMMEDIATELY **OBSERVE** the following:

   - Generator phase current increases ____
   - MW load and MVAR load increases ____
   - Synchroscope pointer steady at 12 o'clock position ____

7. **ADJUST** Generator MVARs to within the limits of Excitation Limit Curve of Attachment 1. ____

5.2.17. **PLACE** the 500 KV BUS BS 6-5 (BS-2-6) SYNCHROSCOPE in OFF using Control Room Bezel Key. ____

5.2.18. **PLACE** the 500 KV BUS BS 2-6 (BS 6-5) SYNCHROSCOPE in ON using Control Room Bezel Key. ____

5.2.19. **CHECK** SYNCHROSCOPE Pointer remains steady at 12 o'clock position. ____

5.2.20. **PRESS** AND **HOLD** BS 2-6 (BS 6-5) SYNCH CHECK OFF push-button. ____

5.2.21. **CLOSE** BS 2-6 (BS 6-5) AND **RELEASE** SYNCH CHECK OFF push-button. ____

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	264000 K5.06	
	Importance Rating	3.4	

Knowledge of the operational implications of the following concepts as they apply to EMERGENCY GENERATORS (DIESEL/JET) : Load Sequencing.

Proposed Question: Common 9

The plant experienced a loss of 10A401.

The bus was subsequently restored to the normal lineup.

However, before the load sequencer was reset, all power was again lost to 10A401.

How will the EDG and electrical distribution system respond to this event?

A The EDG will automatically start. The loads on bus 10A401 will sequence on after the output breaker is closed.

B. The EDG will automatically start. The loads on bus 10A401 will NOT sequence on after the output breaker is closed.

C. The EDG will require a manual start. The loads on bus 10A401 will sequence on after the output breaker is closed.

D. The EDG will require a manual start. The loads on bus 10A401 will NOT sequence on after the output breaker is closed.

Proposed Answer: B

Explanation (Optional):

B. Correct – The EDG will start on the loss of power to the bus. Without the sequencer being reset, no loads will sequence onto the bus.

A. Incorrect – The loads will not sequence on.
C. Incorrect – The EDG will auto start when the bus loses power.
D. Incorrect – The EDG will auto start when the bus loses power. Without the sequencer being reset, no loads will sequence onto the bus.

Technical Reference(s) AB-ZZ-135 (Attach if not previously provided)

1EAC00E007

Proposed references to be provided to applicants during examination: None

Learning Objective: _____ (As available)

Question Source: Bank # INPO Bank 25444

Modified Bank # _____ (Note changes or attach parent)

New _____

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge _____

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

Comments:

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#9 NRC

The LOP train continues its output sequence of start signals until the last step has been completed. At this time, the system can be reset. The BLOCK relays have been programmed to remain active for a full thirty (30) minutes once activated and reset of the system does not release these relays.

The specific times for the various LOP sequences for the four (4) Safety Features Sequencer Systems are presented in Table 2-1.

Upon receiving a Loss of Coolant Accident (LOCA) signal, the Emergency Load Sequencer will automatically provide:

- a set of Process Start Inhibit Signals (PSIS) to various LOCA loads via hermetically sealed relays;
- a train of start signals to selected LOCA loads via hermetically sealed relays; and
- an energize signal to the thirty (30) BLOCK relays.

The LOCA train continues its output sequence of start signals until the last step has been completed. At this time, the system can be reset. The BLOCK relays have been programmed to remain active for a full thirty (30) minutes once activated and reset of the system does not release these relays.

The specific times for the various LOCA sequences for the four (4) Safety Features Sequencer Systems are presented in Table 2-2.

As stated, the corrective action required from the Emergency Load Sequencer System is automatic upon initiation by a LOCA or LOP Signal. Upon completion of the step sequence, the system must be reset. This can be accomplished via the keylock switch on the Manual Test Panel of the Logic Cabinet or remotely in the Main Control Room (future feature).

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#9 ~RC

D. LOP during LOCA Sequencing - Should an LOP signal arrive when the LOCA is sequencing, the LOCA sequencing stops and resets to time zero. When bus power returns via the Standby Diesel Generator, the LOCA signal overrides the LOP sequencer and the LOCA sequencer restarts from the beginning of the sequence. The Emergency Load Sequencer must be manually reset at the end of the sequence.

E. LOCA after LOP Sequencing is Completed - The LOCA signal sheds the non-1E loads (not part of this system) and initiates sequenced signals to loads required for the safe shutdown of the reactor. The Emergency Load Sequencer must be reset at the end of the sequence.

F. LOP after LOCA Sequencing is Completed - If the LOCA signal is still present at the system, the LOCA train is reinitiated as soon as the Standby Diesel Generator Breaker closes, overrides the LOP sequence and initiates sequenced signals to loads required for the safe shutdown of the reactor. The Emergency Load Sequencer must be manually reset at the end of the sequence.

G. LOCA with Simultaneous LOP - The LOCA signal takes precedence. As soon as the Standby Diesel Generator Breaker closes, the LOCA signal overrides the LOP sequence, initiates the LOCA train and provides sequenced signals to loads required for the safe shutdown of the reactor. The Emergency Load Sequencer must be manually reset at the end of the sequence.

Note that in all of the situations listed above, the PSIS (Process Start Inhibit Signal) signal is generated to prevent automatic process-related or maintained contact starts of Emergency Load Sequencer System controlled equipment before receiving the programmed train start signal. The PSIS remains effective until the sequencing is completed. Upon completion of the sequencing cycle, the PSIS is reset when the system sequence is manually reset.

H. Manual Test Operation - The various test features of this system are fully detailed in Section 2.3.

I. LOCA during Manual Test - A LOCA occurring during a manual test start sequence takes precedence and automatic LOCA step train sequencing commences to provide signals to loads required for the safe shutdown of the reactor. The Emergency Load Sequencer must be manually reset at the end of the sequence.

#10
NRC

4.24 WHEN HPCI/RCIC Room temperatures return to normal THEN **PERFORM** the following: [**CD-675F**]

4.24.1. **ENSURE** the following the HPCI Room Fire Dampers are open:

- 1FP-GUD292 ____
- 1FP-GUD293 ____

4.24.2. **PLACE** HPCI AND RCIC high temperature isolation trip switches to NORMAL at the following panels: (Key 172, PA2235 in Work Control Key Cabinet for all switches) ____

A. HPCI at Panel P-620 Switch B21B-S6A ____

Panel P-641 Switch B21B-S6C ____

B. RCIC at Panel P-621 Switch B21B-S5B ____

Panel P-640 Switch B21B-S5D ____

4.25 When directed by the CRS, **PERFORM** the following:

4.25.1. **ENSURE** all Load Sequencer breakers A (B, C, D) C428 are closed. ____

NOTE

Emergency load sequencer keys in Work Control Key Cabinet:
- Key 86, GEK594 or EB0001 (Sequencer A)
- Key 87, CAT60 or EB0001 (Sequencer B)
- Key 88, GEK592 or EB0001 (Sequencer C)
- Key 89, GEK591 or EB0001 (Sequencer D)

4.25.2. **RESET** LOP Emergency Load sequencers using WCC keys. ____

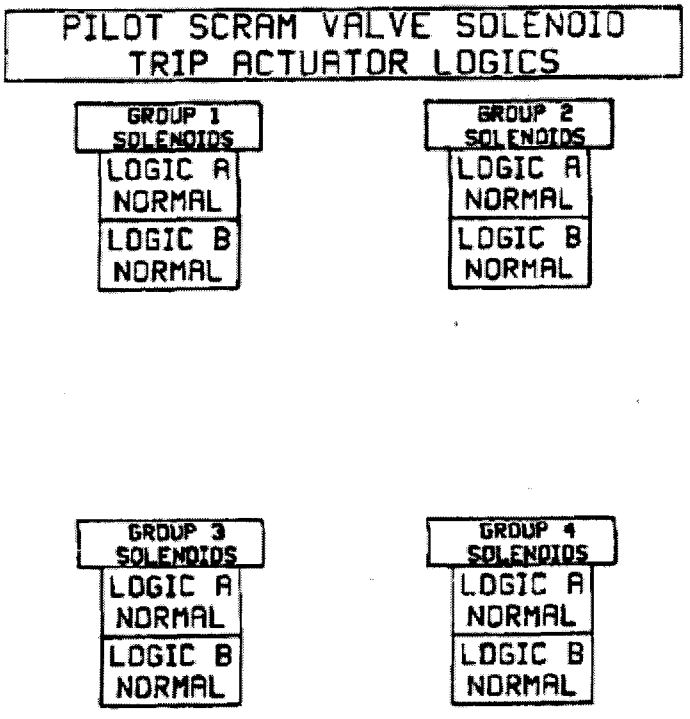
4.25.3. **RESET** LOCA sequencers and **VERIFY** "Logic A (B, C, D) Initiation" lamps are extinguished. ____

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	212000 K5.02	
	Importance Rating	3.3	

Knowledge of the operational implications of the following concepts as they apply to REACTOR PROTECTION SYSTEM : Specific logic arrangements.

Proposed Question: Common 10

The reactor is operating at 100% power. The operator observes the Logic A Normal light under the Group 1 Solenoids is NOT illuminated. The cause is NOT a bad light bulb.



Given this condition, if a half-scram condition occurs on the:
(Chose answer based on logic function ONLY)

A. 'B1' logic; 1/2 of the control rods will scram.

B. 'A2' logic; 1/2 of the control rods will scram.

C. 'A1' logic; 1/4 of the control rods will scram.

D. 'B2' logic; 1/4 of the control rods will scram

Proposed Answer: D

a. Explanation (Optional): IAW RPS Lesson Plan NOH01RPS00C-05 Section II.C.4.c. - Energized trip systems maintain the scram pilot valve solenoids energized
   1) Two contacts in series are maintained closed to energize the A (B) solenoid
   a) One contact will open with a scram signal in trip channel A1 (B1)
   b) The other contact will open with a scram signal in trip channel A2 (B2)
   2) Either A1 or A2 trip channel will de-energize the A solenoids for a half-scram.
   3) Either B1 or B2 trip channel will de-energize the B solenoids for a half-scram.

D. Correct. ANY 'B' side RPS trip will de-energize the 'B' scram pilot solenoid valves for the GP1 rods, resulting in their scramming in. Each group comprises approximately 1/4 of the rods.

A. Incorrect. The 'A' scram pilot solenoid valves for the GP1 rods are already de-energized.
B. Incorrect. The 'A' scram pilot solenoid valves for the GP1 rods are already de-energized
C. Incorrect. ANY 'B' side RPS trip will de-energize the 'B' scram pilot solenoid valves for the GP1 rods, resulting in their scramming in. Each group comprises approximately 1/4 of the rods.

Technical Reference(s): Lesson Plan NOH01RPS00C-05 (Attach if not previously provided)
Prints PN1-C71-1020-006 Sheets 7, 13 ,14

Proposed references to be provided to applicants during examination: None

Learning Objective: RPS000E017 (As available)

Question Source: Bank # 62627
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

Comments:

#10

1) Each trip channel has dedicated sensors, trip units, relays, and manual switches.

2) Trip channels measuring parameters at scram setpoints or beyond de-energize its trip system.

3) Trip channels and trip systems are powered from associated RPS Bus

TABLE 1

4) See Table 1 for scram parameters

FIG 5,6

c. Energized trip systems maintain the scram pilot valve solenoids energized

1) Two contacts in series are maintained closed to energize the A (B) solenoid

a) One contact will open with a scram signal in trip channel A1 (B1)

b) The other contact will open with a scram signal in trip channel A2 (B2)

2) Either A1 or A2 trip channel will de-energize the A solenoids for a half-scram.

3) Either B1 or B2 trip channel will de-energize the B solenoids for a half-scram.

OBJ 14

4) (Either A1 or A2) and (either B1 or B2) will produce a full scram; considered a 1-out-of-2-taken-twice logic.

5) The de-deenergization of both solenoids allows the air to be vented off the operators of the scram inlet and outlet valves failing them open. This in turn admits pressurized water from the HCU to the under piston side of the control ride drive mechanism resulting in the rapid insertion (scram) of the control rod.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	263000 K6.01	
	Importance Rating	3.2	

Knowledge of the effect that a loss or malfunction of the following will have on the D.C. ELECTRICAL DISTRIBUTION: A.C. electrical distribution.

Proposed Question: Common 11

The plant is operating at full power when a LOCA and a loss of offsite power (LOP) occur. Emergency diesel generators respond as designed.

WHICH one of the following describes the status of the 1E and Non-1E 125 VDC Battery Chargers 30 seconds after the event?

A. 1E battery chargers are in service and the Non-1E battery chargers are load shed and CANNOT be returned to service.

B. 1E and Non-1E battery chargers are load shed and both are automatically restored at the same time by load sequencing.

C. 1E battery chargers are in service and the Non-1E battery chargers are load shed and can be manually restored by overriding the load shed and re-energizing the MCC's.

D. 1E and Non-1E battery chargers are load shed; the 1E battery chargers are automatically restored by load sequencing and the Non-1E battery chargers will be restored 2 minutes after the sequencer starts.

Proposed Answer: C

Explanation (Optional):

IAW DC Electrical Lesson Plan section X.C.1.b. - Upon a LOCA, the MCCs that supply the battery chargers (excluding the guardhouse battery charger 10D514) are shed from the Class 1E 480 VAC Unit Substations that normally supply their power. Shedding of the MCCs places the 125 VDC (non- 1E) power requirements on the respective batteries.

The LOCA signal for the MCC feeder breakers can be overridden in the control room at 10C650

C. Correct - The 1E battery chargers supply breakers are not load shed. The Non-1E battery chargers can be restored manually.

A. Incorrect - The Non 1E chargers are not automatically restored after a LOCA.
B. Incorrect – The 1E chargers are not load shed. Non-1E battery chargers are not automatically restored.
D. Incorrect - The 1E battery chargers are not load shed. Non-1E battery chargers are not automatically restored

Technical Reference(s) OP-SO-SM-0001, Table SM-020 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: DCELECE015 (As available)

Question Source: Bank # 54243
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

Comments:

#11

2) The MCC for the guardhouse 125 VDC switchgear panel has a standby diesel generator (00G503) as its backup power source.

b. Loss of Coolant Accident (LOCA)

Upon a LOCA, the MCCs that supply the battery chargers (excluding the guardhouse battery charger 10D514) are shed from the Class 1E 480 VAC Unit Substations that normally supply their power.

1) Shedding of the MCCs places the 125 VDC (non- 1E) power requirements on the respective batteries.

2) The LOCA signal for the MCC feeder breakers can be overridden in the control room at 10C650 (this will be covered in detail during the 1E AC Distribution System lecture).

Obj. 17&21

c. Using the below listed procedures discuss the following:

1) HC.OP-AB.ZZ-0147, DC System Grounds
   a) Symptoms
   b) Subsequent operator actions:
   c) Discussion section
2) HC.OP-AB.ZZ-0149, 250 VDC Malfunction
   a) Symptoms
   b) Subsequent operator actions:
   c) Discussion section
3) HC.OP-AB.ZZ-0150, 125 VDC Malfunction
   a) Symptoms
   b) Subsequent operator actions:
   c) Discussion section
4) HC.OP-AB.ZZ-0151, +24 VDC Malfunction
   a) Symptoms
   b) Automatic Actions
   c) Subsequent operator actions:
   d) Discussion section

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215003 K6.05	
	Importance Rating	3.1	

Knowledge of the effect that a loss or malfunction of the following will have on the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM: Trip units.

Proposed Question: Common 12

Given the following conditions:

- A plant startup is in progress.
- IRM "G" is inoperable and bypassed
- All other IRMs are indicating on range 8

Then, a power failure occurs on the trip unit for IRM "E".

Which one of the following describes the effects of this power failure on the IRM trip units?

A. With the mode switch in RUN, a half scram and rod block would occur.

B. With the mode switch in RUN, ONLY a rod block would occur.

C. With the mode switch in STARTUP, a half scram and rod block would occur.

D. With the mode switch in STARTUP, ONLY a rod block would occur.

Proposed Answer: C

Explanation (Optional): IAW LP NOH01IRMSYS-02 page 17 & 18

C. Correct. A loss of power

A. Incorrect. All scrams and rod blocks are bypassed with mode switch in RUN
B. Incorrect. All scrams and rod blocks are bypassed with mode switch in RUN to the trip units would cause a rod block and half scram with the mode switch in Startup
D. Incorrect. A half scram would also occur

Technical Reference(s) NOH01IRMSYS-02 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: IRMSYSE012 (As available)

Question Source: Bank # _____
Modified Bank # _____ (Note changes or attach parent)
New X

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

Comments:

#12

3) IRM Inoperative Rod Block

a) Certain parameters are monitored which provide an indication of IRM channel inoperability. These parameters are:

- Detector high voltage power supply (LOW)
- Drawer module unplugged
- Channel mode switch (not in OPERATE)

OBJ 5

b) This rod block is bypassed if either:

- The associated IRM is bypassed

  OR

- The reactor mode switch is in RUN

4) IRM Downscale

The downscale rod block setpoint of 5/125 of scale indicates a possible malfunction within the IRM channel. A control rod withdrawal block is initiated to inhibit reactor power escalation under conditions of IRM inoperability.

This rod block is bypassed if either:

- The associated IRM is bypassed

  OR

- The reactor mode switch is in RUN

  OR

- Associated IRM range switch selected to Range 1 (enables reactor startup)

FIG 8

4. Reactor Scram (Inputs into the reactor protection system - RPS)

a. The IRMS provide inputs into the RPS under conditions which indicate excessive reactor power for existing conditions and/or IRM channel inoperability.

b. Loss of power will result in the scram signal being generated by the IRMS (fail-safe). The assignment of IRMS to the RPS logic is as follows:

IRMS	RPS TRIP SYSTEM
A&E	A (Logic A1)

#12

* These switches also serve similar functions for the SRM detectors.

c. Detector select pushbuttons (A,B,C,D,E,F,G,H)

This pushbutton control is used to enable/disable the detector positioning circuitry for the selected detector. (i.e. it selects/deselects the detector for movement.)

FIG 7

3. Rod Blocks (Inputs into the reactor manual control system).

a. The IRMS provide inputs into the reactor manual control system (RMCS) to inhibit control rod withdrawal under conditions which could result in fuel cladding damage (overpower condition) or which indicate IRM channel inoperability.

OBJ 5 FIG 7

b. Loss of power will result in a rod block being generated by the IRMS (fail safe). There are two trip systems within the RMCS with four IRMs assigned to each trip system. A single IRM input into either RMCS trip system will initiate a rod block.

1) Detector Not Full In Rod Block.

Accurate core power monitoring cannot be achieved during startup if the IRM detectors are not fully inserted. Therefore, it would be imprudent to allow rod withdrawal under conditions when the IRM is required to be functional.

This rod block is bypassed if either:

- The associated IRM is bypassed

OR

- The reactor mode switch is in RUN

2) IRM Upscale Rod Block

The upscale rod block setpoint of 108/125 of scale indicates reactor power is high for existing conditions. The control rod withdrawal block prevents further power escalation with control rods.

This rod block is bypassed if either:

- The associated IRM is bypassed

OR

- The reactor mode switch is in RUN

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	203000 A1.01	
	Importance Rating	4.2	

Ability to predict and/or monitor changes in parameters associated with operating the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) controls including: Reactor water level

Proposed Question: Common 13

Given the following conditions:

- The reactor is shutdown and is being cooled down
- RPV level is being held relatively constant at (-30) inches using Startup Level Control
- Reactor pressure is about 390 psig
- All rods are fully inserted

Then, 'C' RHR Loop initiation logic is inadvertently initiated in the LPCI mode, RHR LPCI Injection Valve BC-HV-F017C opens.

Which of the following describes the operational effect of this condition?

A. RPV level will rise and torus level will lower.

B. RPV level will lower and torus level will rise.

C. RPV level will rise and torus level will rise.

D. RPV level and torus level will remain relatively constant.

Proposed Answer: D

Explanation (Optional): IAW RHR LP NOH01RHRSYSC-06

D. Correct - The shut-off head of the RHR pumps is about 366 psig, the min flow valve will remain open. RPV level will remain relatively constant.

A. Incorrect – RHR will not be injecting so levels will remain relatively constant

B. Incorrect- Torus level will not be significantly affected when the RHR pump is at shutoff head

C. Incorrect- The shut-off head of the RHR pumps is about 366 psig. Levels should remain relatively unchanged because the RHR pump will not inject at shutoff head.

Technical Reference(s) RHR LP NOH01RHRSYSC-06 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: RHRSYSE011 (As available)

Question Source: Bank # 56412
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

Comments:

#13-nrc

Obj. E003e

6. Fuel Pool Cooling Assist

   RHR Loops A and/or B can be manually aligned to provide cooling for the spent fuel pool if additional cooling capacity is required.

Obj. E003h

7. Alternate Injection

   IF other systems/sources are not available for RPV level restoration or flooding, SSW can be supplied to the RPV as a last resort. Key lock switches on 10C650A and 10C651A allow positioning of isolation valves to connect SSW and RHR Loop B.

B. Major Subsystems/Components

Obj. E007a

1. RHR Pumps (AP-DP202)

   a. Purpose

      The RHR pumps develop the required discharge head and flow rate to support all modes of RHR System operation.

   b. Characteristics

      1) Each pump is located in individual pump rooms on the 54 ft elev, west side of the Rx Bldg. The pumps are four stage, centrifugal, deep well pumps and each can provide a flow rate of 10,000 gpm at a discharge head of approx. 171 psig (Note Technical Specifications requires LPCI mode flow of at least 10,000 gpm against a test line pressure corresponding to a reactor vessel to containment differential pressure of >20 psid.) Pump shutoff head is approx. 366 psig (875 ft).

Seal flow approx. 3-7 gpm through separator

      2) A small portion of pump discharge flow is routed through a cyclone separator and heliflow HX to provide flushing and cooling of the pump mechanical seal. The cyclone separator ensures particle free water (centrifugal force separates suspended solids) for flushing the seal areas. SACS flow through the heliflow HX provides cooling for the seal water.

      The discharge of both the cyclone separator and the pump mechanical seal is routed back to the RHR pump suction.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	261000 A1.04	
	Importance Rating	3.0	

Ability to predict and/or monitor changes in parameters associated with operating the STANDBY GAS TREATMENT SYSTEM controls including: Secondary containment differential pressure

Proposed Question: Common 14

The plant is at full power. All systems are operable.

Reactor Building Ventilation (RBVS) is operating in a normal alignment

Which one the following describes the actions required when swapping over to FRVS IAW HC.OP-SO.GU-0001 "FRVS Operation" and their affect on Secondary Containment D/P throughout the evolution?
(Actions are listed in the order performed)

A. Remove the RBVS exhaust fan from service
Remove the RBVS supply fan from service
Start an FRVS vent fan
Start the FRVS recirc fans
D/P will remain negative

B. Remove the RBVS exhaust fan from service
Remove the RBVS supply fan from service
Start an FRVS vent fan
Start the FRVS recirc fans
Initially D/P will go positive but then return to negative

C. Start an FRVS vent fan
Remove the RBVS supply fan from service
Remove the RBVS exhaust fan from service
Start the FRVS recirc fans
D/P will remain negative

D. Start an FRVS vent fan
Remove the RBVS supply fan from service
Remove the RBVS exhaust fan from service
Start the FRVS recirc fans
Initially D/P will go positive but then return to negative

Proposed Answer: C

Explanation (Optional): IAW HC.OP-SO.GU-0001, Section 5.3.4 – The FRVS vent fan is first

action taken in the sequence. This will ensure a negative pressure in the RB when the RBVS exhaust fans are removed from service in the next sequenced step

C. Correct.

A. Incorrect. The FRVS vent fan must be started first
B. Incorrect. The FRVS vent fan must be started first
D. Incorrect – D/P will remain negative

Technical Reference(s) HC.OP-SO.GU-0001 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: RBVENTE005 (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

Comments:

NOTE 5.3.3

Normal operating FRVS Vent System lineup is with one FRVS Vent Fan operating and the other FRVS Vent Fan in AUTO.

5.3.3 **START** 1 FRVS Vent Fan as follows: _____

(1) FRVS VENT

A. Momentarily **PRESS** the MAN push-button for A(B)V206 FRVS Vent Fan. _____

B. **OBSERVE** alarm BOP SAFETY SYS OUT OF SVCE. _____

C. Momentarily **PRESS** the A(B)V206 START push-button for A(B)V206 FRVS Vent Fan. _____

D. **OBSERVE** an indicated flow of approximately 9000 cfm on FR-9426A(B) for the running FRVS Vent Fan. _____

E. **PRESS** the AUTO LEAD PB for the inservice FRVS Vent Fan AND **INITIAL** Attachment 1. _____

F. **PRESS** the AUTO PB for the out of service FRVS Vent Fan AND **INITIAL** Attachment 1. _____

NOTE 5.3.4

Step 5.3.4 is performed from Aux Bldg HVAC Panel 10C382 unless otherwise specified.

Performance of Step 5.3.4 will result in the Reactor Building Ventilation System being unavailable in the event of a LOP. In the event of a LOP, manual restoration of Reactor Building Ventilation will be required OR an FRVS Vent Fan placed in service to maintain a negative pressure in the Reactor Building.

5.3.4 **PERFORM** the following to remove the Reactor Building Ventilation System from service: _____

A. **PERFORM** the following at Aux Building HVAC Panel 10C382: _____

2

RBVS SUPPLY

1. **PLACE** Control Switch for the idle Reactor Bldg Supply Fan C(A,B)VH300 in STOP. _____
2. **STOP** B(C,A)VH-300, Reactor Bldg Supply Fan. _____
3. **STOP** A(B,C)VH-300, Reactor Bldg Supply Fan. _____

3

RBVS EXHAUST

4. **PLACE** Control Switch for the idle Reactor Bldg Exhaust Fan C(A,B)V301 in STOP. _____
5. **STOP** B(C,A)V301, Reactor Bldg Exhaust Fan. _____
6. **STOP** A(B,C)V301, Reactor Bldg Exhaust Fan. _____

B. **CLOSE** the following dampers at Panel 10C651E: _____

4

- HD9370A, Reactor Bldg Outbd Sply _____
- HD9414A, Reactor Bldg Inbd Exch _____
- HD9370B, Reactor Bldg Inbd Sply _____
- HD9414B, Reactor Bldg Outbd Exch _____

5.3.5 **START** 4 FRVS Recirculation Fans as follows: _____

(4) FRVS RECIRC

A. **MOMENTARILY PRESS** A(B,C,D,E,F)V213 START PB for the desired fans. _____

B. **OBSERVE** an indicated flow of approximately 30,000 cfm on FR-9377 for each of the running fans. _____

C. **MONITOR** the following: _____

- PDR-9377 FRVS DIFF PRESS for indication of ≤ 6.0" W.G. for each operating Air Filter Unit _____
- PDR-9425A(B) FRVS-VENT System Filter D/P ≤ 3.0" W.G. _____

5.3.6 **PERFORM** independent verification that the system is aligned IAW Attachment 1. _____

5.4 Removing FRVS from Service

NOTE 5.4

Observe the Filtration, Recirculation and Ventilation System operability requirements of Technical Specification 3.6.5.3.

5.4.1 **ENSURE** all Prerequisites have been satisfied IAW Section 2.4. _____

5.4.2 **REMOVE** running FRVS Recirc Fans from service as follows: _____

A. IF restoring from an AUTO START: _____

1. **MOMENTARILY PRESS** the STOP PB for: _____

- EV213 FRVS RECIRC FAN _____
- FV213 FRVS RECIRC FAN _____

2. **PRESS** the AUTO/LOCKOUT PB to select Lockout for: _____

- EV213 FRVS RECIRC FAN _____
- FV213 FRVS RECIRC FAN _____

Continued next page

Proposed Answer: B

Explanation (Optional):
IAW ARP C1-F1 & HC.OP-SO.SA-0001

B. Correct. IAW with ARP, If both pumps are not running with a valid signal present, turn the key lock on and start the pumps

A. Incorrect. Both pumps should be running with a manual initiation and the 230 second timer timed out
C. Incorrect. Both pumps should be running with a manual initiation and the 230 second timer timed out
D. Incorrect. Key lock must be turned on prior to starting pump

Technical Reference(s) ARP C1-F1 & HC.OP-SO.SA-0001 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: RRCS00E005 (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

Comments:

H15-NRC

3.3.4 RRCS - Standby Liquid Control System injection will initiate after receipt of the following signals:

A. High Reactor Steam Dome Pressure (1071 PSIG) sealed in

or

B. Low Low Reactor Water Level (Level 2, -38") not sealed in

or

C. Manual Initiation RRCS

and

D. 230 second time delay

and

E. Reactor Power not downscale (> 4% or APRM INOP)

3.3.5 The following Reactor Water Clean-up System Valves will isolate upon a SBLC System initiation:

BG-HV-F001
BG-HV-F004

3.3.6 RRCS - ARI can be reset 30 seconds after initiation, if the initiating condition is clear.

3.3.7 RRCS - Recirc Trip can be reset 13 minutes 50 seconds after initiation.

3.3.8 RRCS - Standby Liquid Control System actuation and RWCU System isolation can be reset 10 minutes after initiation.

4.0 EQUIPMENT REQUIRED

None

H15-NRC

ATTACHMENT F1

SLC/RRCS INITIATION FAILURE

Window Location C1-F1

OPERATOR ACTION:

1. **DETERMINE** IF a valid RRCS initiation signal is present,
   IF signal is valid,
   VERIFY that both SLC Pumps are running.
2. IF a valid RRCS initiation signal is present
   AND a SLC Pump is not running:
   a. **TURN** the non-running SLC Pump KEY-LOCK Switch to ON.
   b. **PRESS** the START pushbutton for the failed pump.
3. **REQUEST** the CRS to initiate corrective action.

INPUTS

Digital Point/ Indication	Nomenclature/Condition	Automatic Action
D2600	SLC/RRCS A INITIATION FAILURE	Alarm only
D2601	SLC/RRCS B INITIATION FAILURE	Alarm only

REFERENCES: J-48-0, Sht. 5
E-6768-0, Sht. 2

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	223002 K1.03	
	Importance Rating	3.0	

Knowledge of the physical connections and/or cause effect relationships between PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF and the following: Plant Ventilation

Proposed Question: Common 16

The plant is in OPCON 4 with the following conditions:

- Primary Containment has been de-inerted
- CACS is aligned to purge the drywell and suppression chamber
- The 'B' and 'C' Reactor Building Ventilation Supply and Exhaust Fans are running
- The 'A' Reactor Building Ventilation Supply and Exhaust Fans are in AUTO

Then, an operator arms and depresses the 'D' Channel PCIS Manual Initiation pushbutton on 10C651C for a surveillance test.

Two minutes later plant condition(s) stabilize.

Which one of the following describes the final status of the containment purge lineup and the Reactor Building Ventilation System (RBVS)?

A. The containment purge lineup will isolate. The Reactor Building Ventilation Fans will be unaffected.

B. The containment purge lineup will isolate. NO Reactor Building Ventilation fans will be running.

C. The containment purge lineup will NOT be affected. The Reactor Building Ventilation Fans will be unaffected.

D. The containment purge lineup will NOT be affected. NO Reactor Building Ventilation fans will be running.

Proposed Answer: B

Explanation (Optional):

B. Correct. Manual initiation of the 'D' Channel PCIS closes the GS-HV-4950, 4962, 4979, and 4980. These valves isolate the purge supply and exhaust lines. While the 'D' channel does not directly trip the running RBVS supply and exhaust fans, it will close the GU-HD-9414B and 9370B. These valves isolate the Reactor Building Ventilation System supply and exhaust lines, which will result in all running fans tripping on low flow after a 90 second time delay. The 'A' RBVS supply and exhaust fans are directly tripped (load shed) by the 'D' Channel PCIS signal.

A Incorrect – NO RBVS fans will be running

C. Incorrect – containment purge supply and exhaust will be isolated. NO RBVS fans will be running.

D. Incorrect – containment purge supply and exhaust will be isolated

Technical Reference(s) HC-OP.SO.SM-0001 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: INERT0E012 (As available)

Question Source: Bank # 62574
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

Comments:

#16-MC

OBSERVE the Group 12 Valves listed in Table **SM-012** have closed under the Manual or Automatic Isolation Signals specified.

TABLE SM-012

VALVE NO.	NOMENCLATURE	MANUAL ISOLATION	AUTO ISOLATIONS								
			A	B	C	D	E	F	G	H	I
"GS-HV4956	CONT ATMOSPHERE CPCS RTN TO CNTMT DPMR	*A CNTMT	X	X	X						
"GS-HV4979	CONT ATMOSPHERE CPCS RTN TO CNTMT DMPR	*D CNTMT	X	X	X						
"GS-HV4951	CONT ATMOSPHERE CNTMT TO CPCS BYPASS	*D CNTMT	X	X	X						
"GS-HV4950	CONT ATMOS CNTMT TO CPCS OUTBD DPMR	*D CNTMT	X	X	X						
"GS-HV4952	CONT ATMOS CNTMT TO CPCS INBD DMPR	*A CNTMT	X	X	X						
"GS-HV4980	CONT ATMOS CPCS RTN TO SUPP CHMBR	*D CNTMT	X	X	X						
"GS-HV4958	CONT ATMOSPHERE CPCS RTN TO SUPP CHMBR	*A CNTMT	X	X	X						
"GS-HV4963	CONT ATMOSPHERE SUPP CHMBR N_2 M/U EXH	*D CNTMT	X	X	X						
"GS-HV4962	CONT ATMOS SUPP CHMBR TO CPCS OUTBD EXH	*D CNTMT	X	X	X						
"GS-HV4964	CONT ATMOSPHERE SUPP CHMBR TO CPCS INBD EXH	*A CNTMT	X	X	X						
"GS-HV4974	CONT ATMOSPHERE N_2 M/U SPLY ISLN	*D CNTMT	X	X	X						
"GS-HV4978	CONT ATMOSPHERE N_2 SPLY HDR ISLN	*D CNTMT	X	X	X						

ISOLATION	SETPOINT
A - REACTOR VESSEL WATER LEVEL 2	-38"
B - DRYWELL PRESSURE - HIGH	1.68 psig
C - REACTOR BUILDING EXHAUST RADIATION - HIGH	1×10^{-3} uCi/cc

* - Can receive a Half Isolation from the corresponding NSSSS Manual Isolation
" - Isolation can be bypassed by ISLN OVRD PB

#16-NAC

OBSERVE the Group 19 Dampers listed in Table **SM-019** have closed under the Manual or Automatic Isolation Signals and other Actions have occurred for Equipment listed as specified.

TABLE SM-019 (1 of 3)

EQUIPMENT NUMBER	NOMENCLATURE	ACTION	MANUAL ISOLATION	AUTO ISOLATIONS			
				A	B	C	D
#GR-HD9414A	REACTOR BLDG SPLY/EXH ISLN INBD EXH	CLOSE	*C CNTMT	X	X	X	X
#GR-HD9414B	REACTOR BLDG SPLY/EXH ISLN OUTBD EXH	CLOSE	*D CNTMT	X	X	X	X
#GU-HD9370A	REACTOR BLDG SPLY/EXH ISLN OUTBD SPLY	CLOSE	*C CNTMT	X	X	X	X
#GU-HD9370B	REACTOR BLDG SPLY/EXH ISLN INBD SPLY	CLOSE	*D CNTMT	X	X	X	X
GT-HD9372A	DRYWELL PURGE EXH DRYWELL VENT	CLOSE	*A CNTMT	X	X	X	X
GT-HD9372C	DRYWELL PURGE SPLY	CLOSE	*A CNTMT	X	X	X	X
GU-HD9395A	FRVS RECIRC BYPASS	CLOSE	*A CNTMT	X	X	X	X
GU-HD9395B	FRVS RECIRC BYPASS	CLOSE	*B CNTMT	X	X	X	X
52-44024	RBVS SUPPLY FAN AVH300	TRIP	*D CNTMT	X	X	X	X
52-41024	RBVS SUPPLY FAN BVH300	TRIP	*A CNTMT	X	X	X	X
52-43024	RBVS SUPPLY FAN CVH300	TRIP	*C CNTMT	X	X	X	X
52-48024	RBVS EXHAUST FAN AV301	TRIP	*D CNTMT	X	X	X	X
52-42024	RBVS EXHAUST FAN BV301	TRIP	*B CNTMT	X	X	X	X
52-45034	RBVS EXHAUST FAN CV301	TRIP	*A CNTMT	X	X	X	X

ISOLATION	SETPOINT
A - REACTOR VESSEL WATER LEVEL 2	-38"
B - DRYWELL PRESSURE - HIGH	1.68 psig
C - REFUEL FLOOR EXHAUST RADIATION - HIGH	2 X 10^{-3} uCi/cc
D - REACTOR BUILDING EXHAUST RADIATION - HIGH	1 X 10^{-3} uCi/cc

- Group 19 Dampers
* - Can receive a Half Isolation Signal from the corresponding NSSSS Manual Isolation

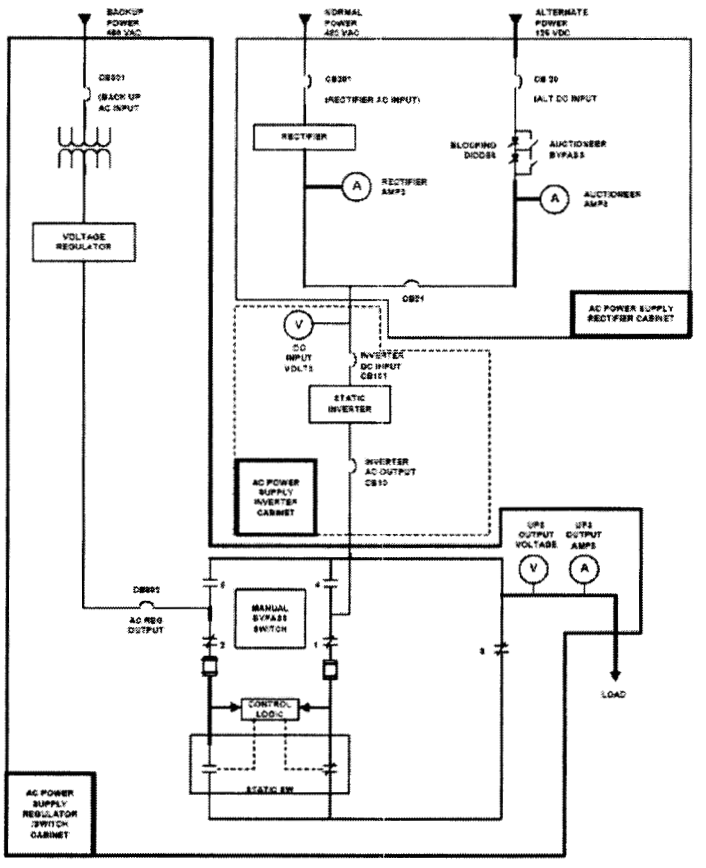
Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	262002 A3.01	
	Importance Rating	2.8	

Ability to monitor automatic operations of the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) including: Transfer from preferred to alternate source.

Proposed Question: Common 17

The Manual Bypass Control Switch on a NON 1E inverter has been placed in the "Bypass-to Alternate" position for testing on a faulty Static Switch.

EXHIBIT 2
Page 1 of 1
UPS POWER CONTROL (TYPICAL)



Which of the following describes the design response if a LOP occurs?

The input to the Static Inverter section will be from…

A. 125 VDC but supply to the system loads will be lost.

B. 125 VDC and maintaining system loads.

C. 480 VAC and maintaining system loads.

D. 480 VAC but supply to the system loads will be lost.

Proposed Answer: A

Explanation (Optional): See attached diagram from HC.OP-SO.PN-0001(Q) Exhibit 3.

A. Correct. The input of the static inverter will be supplied by 125 VDC power but supply to the system loads will be lost - Placing the switch in Bypass to Alternate closes contacts 1,2 and 5 meaning supply to system loads is off the Alternate/Backup supply which is not 1E supplied so loads are lost.

B. Incorrect. The input of the static inverter will be supplied by 125 VDC power and maintaining system loads - contacts 1,2 and 5 meaning supply to system loads is off the Alternate/Backup supply

C. Incorrect. The input of the static inverter will be supplied by 480 VAC power and maintaining system loads - Alternate/Backup supply which is not 1E supplied so loads are lost.

D. Incorrect. The input of the static inverter will be supplied by 480 VAC power but supply to the system loads will be lost - Placing the switch in Bypass to Alternate closes contactsd 1,2 and 5 meaning supply to system loads is off the Alternate/Backup supply.

Technical Reference(s) HC.OP-SO.PN-0001(Q) (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: NON1E0E003 (As available)

Question Source: Bank # 56822
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam _____

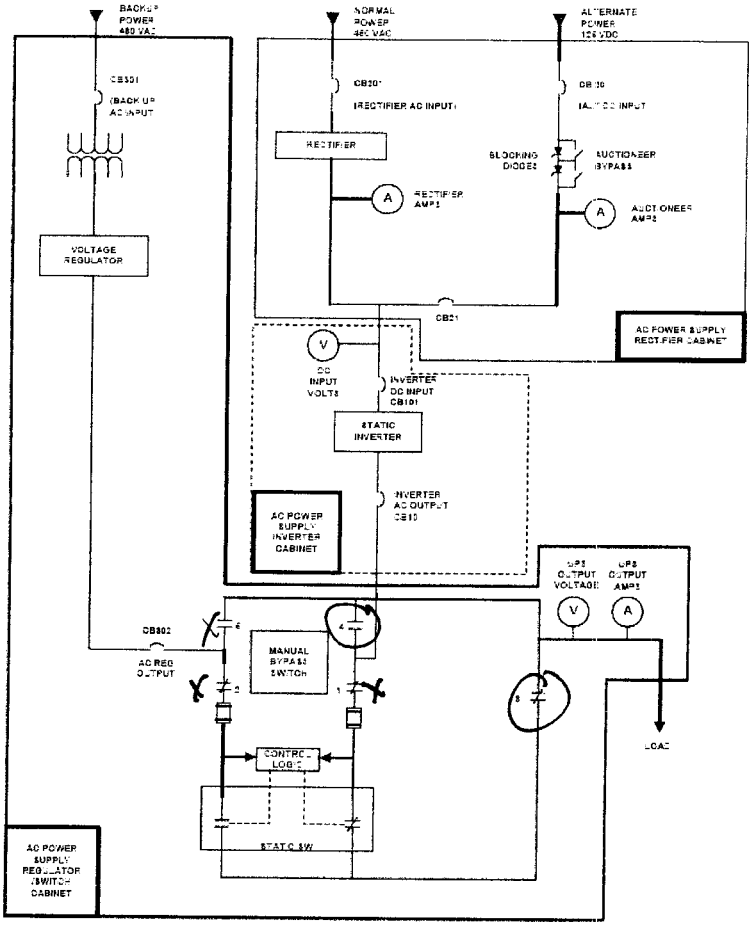
Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

Comments:

#17

EXHIBIT 2
Page 1 of 1
UPS POWER CONTROL (TYPICAL)



Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	209001 A3.03	
	Importance Rating	3.5	

Ability to monitor automatic operations of the LOW PRESSURE CORE SPRAY SYSTEM including: System pressure.

Proposed Question: Common 18

An event has occurred at the plant.

All Core Spray Loops are injecting to the RPV when the Core Spray Min Flow Valve BE-HV-F031A inadvertently strokes open due to a malfunction of FISH-N651A.

How will this affect Core Spray discharge pressure and total indicated flow in the "A" Loop in the control room?

A. Discharge pressure will increase.
Indicated flow will decrease.

B. Discharge pressure will increase.
Indicated flow will increase.

C. Discharge pressure will decrease.
Indicated flow will decrease.

D. Discharge pressure will decrease.
Indicated flow will increase.

Proposed Answer: C

Explanation (Optional):

C. Correct. Due to where the Flow indicator and min flow taps off in the system

A. Incorrect. Discharge pressure will decrease
B. Incorrect. Discharge pressure will decrease, indicated flow will decrease
D. Incorrect. indicated flow will decrease

Technical Reference(s) M-52 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: M-52

Learning Objective: CSSYSE004 (As available)

Question Source: Bank # _____
Modified Bank # _____ (Note changes or attach parent)
New X

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis X

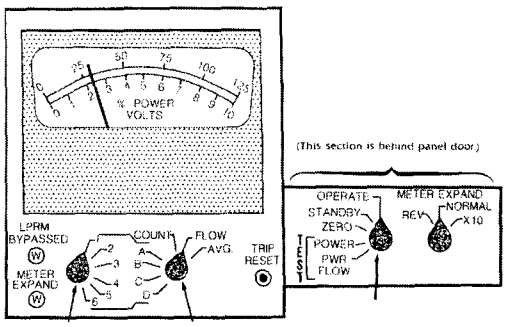
10 CFR Part 55 Content: 55.41 X
_____

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215005 A4.03	
	Importance Rating	3.2	

Ability to manually operate and/or monitor in the control room: APRM back panel switches, meters and indicating lights.

Proposed Question: Common 19



The plant is operating at rated power. APRM 'E' has 5 LPRMs bypassed, APRM 'F' has 5 LPRMs bypassed. All other LPRM inputs are operable.

Which one of the following describes how to use the APRM back panel drawer controls to determine the number of non-bypassed LPRMs and what would their respective meters indicate?

A. Place the respective APRM Meter Function Switch to COUNT.
APRM 'E' meter will indicate 80 on the 0-125% scale.
APRM 'F' meter will indicate 85 on the 0-125% scale.

B. Place the respective APRM Meter Function Switch to COUNT.
APRM 'E' meter will indicate 85 on the 0-125% scale.
APRM 'F' meter will indicate 85 on the 0-125% scale.

C. Place the respective APRM Mode Switch to STANDBY then Place the Meter Function Switch to COUNT.
APRM 'E' meter will indicate 85 on the 0-125% scale.
APRM 'F' meter will indicate 85 on the 0-125% scale.

D. Place the respective APRM Mode Switch to STANDBY then Place the Meter Function Switch to COUNT.
APRM 'E' meter will indicate 85 on the 0-125% scale.
APRM 'F' meter will indicate 80 on the 0-125% scale.

Proposed Answer: A

Explanation (Optional): IAW APRM LP NOH04APRM00C-04

To determine the # of nonbypassed LPRMs on the meter function switch is moved to count

APRM 'E' has 21 LPRM inputs, if 5 are bypassed then 16 remain. Each nonbypassed LPRM is = to 5% on the meter when it is taken to count. Therefore the meter would indicate 80 for APRM 'E'.
APRM 'F' has 22 LPRM inputs, if 5 are bypassed then 17 remain. Each nonbypassed LPRM is = to 5% on the meter when it is taken to count. Therefore the meter would indicate 85 for APRM 'F'.

A. Correct.

B. Incorrect. E would indicate 80, F-85.
C. Incorrect. Only the meter function switch must be moved
D. Incorrect. Only the meter function switch must be moved. E would indicate 80, F-85.

Technical Reference(s) NOH04APRM00C-04 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: LPRM00E005 (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

Comments:

#19-rec

2) That flow unit is removed from the comparator circuit.

NOTE: Bypassing the flow unit does not remove its output from the APRM or RBM channels that receive it. For example: If a flow unit fails low while at power, the flow biased reference signals developed will be very low and will result in trips of those APRMs & RBM channels. Bypassing the flow unit will only allow clearing the rod blocks generated by the flow unit comparator trips.

OBJ. 2a; b; c; d

2. 10C608 Controls

a. Meter Function Switch

FIG. 12(AV1997m), FIG. 13(AV1997k)

1) The meter function switch is used to control the parameters indicated on the associated meter. The following positions are available:

a) AVERAGE

The AVERAGE position enables the % power reading from the associated APRM channel (0 125% scale), to be displayed.

b) FLOW

The FLOW position enables the % flow signal from the associated APRM flow unit to be displayed (0-125% scale).

c) COUNT

The COUNT position enables the meter indication for the number of LPRM inputs into the associated APRM (or LPRM group) channel. The 0-125% scale is used and the reading is divided by 5%. The resultant number indicates the number of "non-bypassed" LPRM inputs (i.e. all LPRMs that have their mode switch in the OPERATE position)

Ex: A reading of 105% = 105÷ 5 = 21 LPRM inputs.

d) A, B, C, D

The alphabetical positions are used in conjunction with the LPRM selector switch to monitor a specific LPRM output. The A, B, C, D position indicates the associated detector level. The x-y detector coordinate is determined from the operator aid located at the top of the local panel.

14/19-PRC

Neutron monitoring system power range panel 10C608 is a control room back panel consisting of five (5) bays. Together the five bays house the electronic components associated with the six (6) APRM and two (2) RBM System channels, the two (2) LPRM groups, the four (4) Recirc flow units and the four (4) power range monitoring system power supply modules.

FIG. 3(AV1563)

2. APRM Averaging Amplifier

   a. Purpose

   The amplifier produces an output signal that is equivalent to the average of all the LPRM inputs it receives. This output signal represents average core power.

FIG. 4(AV2666c)

   b. The LPRM detector outputs are appropriately assigned such that each APRM channel produces an output that is representative of average core power. In order to accomplish this, LPRM detectors are chosen from appropriate radial and axial core locations such that APRM channels A, C & E receive 21 inputs and APRM channels B, D & F receive 22 inputs (typically only one detector per string and 4-6 detectors per level will provide an input to each channel).

FIG. 3(AV1563)

   c. The averaging amplifier will produce an output signal of 0 to +10VDC. This DC signal corresponds to a core thermal power of 0 to 125% of rated. Each LPRM will provide an input to the assigned averaging amp if the individual LPRM amplifier card mode switch is in the operate position. With the switch in any other position, that detector will be removed as an input and the averaging amp will produce an output signal based on the remaining LPRM inputs. The averaging amp output is representative of the average of all LPRM inputs regardless of how many LPRMs are bypassed.

FIG. 5(AV0247d)

3. Recirc Flow Units

   a. Purpose

   Produce a signal representing total recirc system flow for use in determining the flow biased rod block and upscale thermal power scram trip functions.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	400000 A4.01	
	Importance Rating	3.1	

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

Proposed Question: Common 20

Plant conditions are as follows:

- Power is 87%
- "A" SACS Loop is supplying TACS.
- "D" SACS pump is operating in the "B" SACS Loop.
- "B" SACS pump is in AUTO.
- "A", "C" and "D" Service Water pumps are running.
- "B" Service Water pump is in AUTO.

I&C testing causes an inadvertent LOCA signal generation on "C" Core Spray logic. At the same time, an infeed undervoltage trip of the 40308 breaker occurs. All systems responded as designed.

With NO operator action, what will be the final alignment of SACS, TACS and Service Water?

A. "A" SACS Loop is supplying TACS.
"A", "C" and "D" Service Water pumps are running.

B. "A" SACS Loop is supplying TACS.
"A", "B" and "D" Service Water pumps are running.

C. "B" SACS Loop is supplying TACS.
"A", "C" and "D" Service Water pumps are running.

D. "B" SACS Loop is supplying TACS.
"A", "B" and "D" Service Water pumps are running.

Proposed Answer: C

Explanation (Optional): IAW HC.OP-SO.EG-0001 section 3.3.8, HC.OP-SO.EA-0001 section 3.3.1, Loss of Normal supply to 10A403 bus requires transfer to alternate supply 40301 from 1BX501 transformer, this is a dead bus transfer "C" SSW will trip and will start on the LOCA sequencer.

□

C. Correct - Loss of Normal supply to 10A403 bus requires transfer to alternate supply 40301 from 1BX501 transformer, this is a dead bus transfer "C" SSW will trip and will re-start on the LOCA sequencer. Loss of "C" SACS pump will cause a swap of TACS to the B loop. HC.OP-SO.EG-0001, section 3.3.8 "C" channel TACS isolations 2522C/2496C close on LOCA Level 1 signal. This causes a swap to B loop for supply to TACS on low flow.

A. Incorrect - TACS isolations 2522C/2496C close on LOCA Level 1 signal. This causes a swap to B loop for supply to TACS on low flow.

B. Incorrect - TACS isolations 2522C/2496C close on LOCA Level 1 signal. This causes a swap to opposite loop for supply to TACS on low flow. The "B" SW pump will NOT start except for a low flow signal in the associated loop, which did NOT occur.

D. Incorrect - The "B" SW pump will not start except for a low flow signal in the associated loop, which did not occur.

Technical Reference(s): HC.OP-SO.EG-0001 section 3.3.8, HC.OP-SO.EA-0001 section 3.3.1 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: SERWATE006 (As available)

Question Source: Bank # 56921
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

Comments:

H20

3.2.14. The lateral sluice gates may be open or closed as system demand dictates. ____

3.2.15. When placing a Standby SSWS Pump in service, it may be necessary to secure from Standby SACS Loop Temperature Control prior to starting the Service Water Pump. A momentary low flow auto start signal could be generated in the opposite loop when the standby pump is placed in service. Since starting the standby pump results in two SSWS pumps discharging thru one Hx, more flow is diverted thru the RACS cross-tie causing lower flow in the opposite loop. ____

3.3 **Interlocks**

3.3.1. When in AUTO with NO Process Start Inhibit signal, the Station Service Water Pumps auto start on any of the following signals (B AND D Pumps will NOT auto start if control is transferred to the Remote Shutdown Panel):

NO AUTO START SIGNAL RECEIVED

- Associated loop low flow < 13,475 gpm (< 1.0 psid across pump strainer). ____
- Reactor Water low level (< -38 inches) ____
- Drywell high pressure (> 1.68 psig) ____
- Reactor Building high Radiation (> 1X10-3 uCi/cc) ____
- Refueling Floor high Radiation (> 2X10-3 uCi/cc) ____
- Containment Manual Initiation ____

3.3.2. Station Service Water Pumps auto start on Sequence Timer initiated by LOCA or LOP whether in AUTO or MAN. ____

3.3.3. When in AUTO, HV-2371A(B), SACS HX OUTLET VALVE, or HV-2355A(B), SACS HX OUTLET VALVE, will open 10 seconds after the associated Service Water Pump starts and will CLOSE when the associated pump is stopped. The Valve will OPEN on LOCA or LOP, whether in AUTO or MAN, if associated pump has been running at least 10 seconds (HV-2371B and HV-2355B will NOT auto open or close if control is transferred to the RSP). ____

H20

3.3.8. HV-2522A-D, TACS INBD(OUTBD) SPLY/RTN VLVs, AND HV-2496A-D, TACS INBD(OUTBD) SPLY/RTN VLVs, AUTO CLOSE on the following signals: ____

- LOCA Level 1
- LOP
- Lo-Lo-Lo Expansion Tank level
- Respective Pump trip*
  * This can be overridden by placing the respective pump STOP INPUT OVERRIDE switch in the ON position.

3.3.9. HV-2522E-F, SACS/TACS OUTBD (INBD) SPLY ISOL VLVs, AUTO CLOSE on the following signals: ____

- Lo-Lo pressure (2/3 logic) in the TACS Supply Accumulator
- Lo-Lo pressure (2/3 logic) in the TACS Return Accumulator
- Loss of Power

3.3.10. EG-HV-2457A/B are tripped shut at 88°combined Hx outlet temperature by TSH-2457A/B. EG-HV-2517A/B are tripped shut at 88°F combined Hx outlet temperature by TSH-2535A/B and will not respond to TIC-2517A/B until reset. IF one of these valves is known or suspected to have tripped shut on high temperature, the valve(s) can be reset when temperature has returned to < 88 °F in accordance with Section 5.14. ____

4.0 EQUIPMENT REQUIRED

Wrench for removing pipe caps.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	218000 G2.2.25	
Importance Rating	3.2	

Equipment Control: Knowledge of bases in technical specifications for limiting conditions for operations and safety limits. (ADS)

Proposed Question: Common 21

The ADS Technical Specification LCO is based on ensuring the ability of ADS to provide a backup to __(1)__ during a __(2)__ break LOCA.

A. (1) RCIC
(2) Small

B. (1) HPCI
(2) Small

C. (1) RCIC
(2) Large

D. (1) HPCI
(2) Large

Proposed Answer: B

Explanation (Optional):

B. Correct – TS bases 3.5.F. ADS serves as a backup to HPCI during a small break LOCA accident.

A. Incorrect – HPCI not RCIC.
C. Incorrect – HPCI not RCIC, Small not Large.
D. Incorrect – Small not Large Break LOCA.

Technical Reference(s) TS Bases 3.5.1 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: ADSSYSE006 (As available)

Question Source: Bank # WTS 2006
Modified Bank # _____ (Note changes or attach parent)

New _____

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 X
_____

Comments:

#21 NRC

3/4.5.1 and 3/4.5.2 ECCS - OPERATING and SHUTDOWN

The core spray system (CSS), together with the LPCI mode of the RHR system, is provided to assure that the core is adequately cooled following a loss-of-coolant accident and provides adequate core cooling capacity for all break sizes up to and including the double-ended reactor recirculation line break, and for smaller breaks following depressurization by the ADS.

The CSS is a primary source of emergency core cooling after the reactor vessel is depressurized and a source for flooding of the core in case of accidental draining.

The surveillance requirements provide adequate assurance that the CSS will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage to piping and to start cooling at the earliest moment.

The low pressure coolant injection (LPCI) mode of the RHR system is provided to assure that the core is adequately cooled following a loss-of-coolant accident. Four subsystems, each with one pump, provide adequate core flooding for all break sizes up to and including the double-ended reactor recirculation line break, and for small breaks following depressurization by the ADS.

The surveillance requirements provide adequate assurance that the LPCI system will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage to piping and to start cooling at the earliest moment.

Verification every 31 days that each RHR System cross tie valve on the discharge side of the RHR pumps is closed and power to its operator, if any, is disconnected ensures that each LPCI subsystem remains independent and a failure in the flow path in one subsystem will not affect the flow path of the other LPCI subsystem. Acceptable methods of removing power to the operator include de-energizing breaker control power or racking out or removing the breaker. For the valves in high radiation areas, verification may consist of verifying that no work activity was performed in the area of the valve since the last verification was performed. If one of the RHR System cross tie valves is open or power has not been removed from the valve operator, both associated LPCI subsystems must be considered inoperable. The 31 day frequency is acceptable, considering that these valves are under strict administrative controls that will ensure that the valves continue to remain closed with either control or motive power removed.

The high pressure coolant injection (HPCI) system is provided to assure that the reactor core is adequately cooled to limit fuel clad temperature in the event of a small break in the reactor coolant system and loss of coolant which does not result in rapid depressurization of the reactor vessel. The HPCI system permits the reactor to be shut down while maintaining sufficient reactor vessel water level inventory until the vessel is depressurized. The HPCI system continues to operate until reactor vessel pressure is below the pressure at which CSS operation or LPCI mode of the RHR system operation maintains core cooling.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	239002 G2.4.31	
	Importance Rating	4.2	

Emergency Procedures / Plan: Knowledge of annunciator alarms, indications, or response procedures. (SRV)

Proposed Question: Common 22

The plant is operating at 80% power, with the following:

- Reactor water level is +35 inches
- Then, an SRV inadvertently opens

Which one of the following describes the initial Reactor Water level response and what actions are required IAW AB.RPV-0006 "Safety Relief Valve?

A. RPV level will initially rise.
If the SRV CANNOT be closed, within two (2) minutes trip the recirculation pumps and lock the mode switch in Shutdown.

B. RPV level will initially rise.
If the SRV CANNOT be closed, within two (2) minutes reduce the recirculation pumps to minimum and lock the mode switch in Shutdown.

C. RPV level will initially lower.
If the SRV CANNOT be closed, within two (2) minutes trip the recirculation pumps and lock the mode switch in Shutdown.

D. RPV level will initially lower.
If the SRV CANNOT be closed, within two (2) minutes reduce the recirculation pumps to minimum and lock the mode switch in Shutdown.

Proposed Answer: B

Explanation (Optional):

B. Correct – IAW AB.RPV-0006 – IF within 2 minutes the SRV fails to close, reduce the recirculation pumps to minimum and lock the mode switch in Shutdown. RPV Swells up on the RPV pressure reduction when the SRV initially opens.

A. Incorrect. The recirc pumps are to be reduced to minimum, not tripped.

C. Incorrect. RPV Level will rise. The recirc pumps are to be reduced to minimum, not tripped.

D. Incorrect. RPV Level will rise

Technical Reference(s) AB.RPV-0006 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: ABRPV4E004 (As available)

Question Source: Bank # _____
Modified Bank # 22077 (Note changes or attach parent)
New _____

Question History: Last NRC Exam 2005

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

Comments:

RETAINMENT OVERRIDE	
CONDITION	**ACTION**
I. Open SRV Fails to Close Time: _____	WITHIN two minutes, perform the following: ❑ a. **REDUCE** recirc pumps to minimum speed. ❑ b. **LOCK** the Mode Switch in SHUTDOWN. **[T/S 3.4.2.1.b, CD-912X, CD-220C]**

#22 BANK 22077

The plant is operating at 90% power, with the following:

- Reactor water level is 35 inches
- Then, 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. rise and then return to 35 inches

C. rise and remain above 35 inches

D. lower and remain below 35 inches

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	203000 A4.02	
	Importance Rating	4.1	

Ability to manually operate and/or monitor in the control room (RHR/LPCI Injection Mode): System valves.

Proposed Question: Common 23

Given the following conditions:

- Reactor power 85%
- "A" RHR loop in full flow test mode at 10450 gpm flow

Then, a LOCA occurs resulting in the following:

- Drywell pressure is 4.5 psig increasing
- 5 minutes has elapsed since the LOCA
- Reactor pressure is now 500 psig and lowering

Assuming NO operator action, what is the current status of the following "A" RHR Loop valves?

- LPCI injection valve (BC-HV-F017A)
- Test Valve (BC-HV-F024A)
- HX Bypass valve (BC-HV-F048A)

A. (BC-HV-F017A) - open
(BC-HV-F024A) – open
(BC-HV-F048A) – closed

B. (BC-HV-F017A) - open
(BC-HV-F024A) – closed
(BC-HV-F048A) – closed

C. (BC-HV-F017A) - closed
(BC-HV-F024A) – open
(BC-HV-F048A) – open

D. (BC-HV-F017A) - closed
(BC-HV-F024A) – closed
(BC-HV-F048A) – open

Proposed Answer: D

Explanation (Optional): IAW HC.OP-SO.BC-0001

When a LPCI signal is received, the system aligns for RPV injection.

F048 - Receives an open signal for 3 min
F017 - Opens when RPV pressure drops below 450 psig
F024 - Receives a close signal but can be overridden
D. Correct.

A. Incorrect. F017A will still be closed, F024A will be closed, F048A will be open
B. Incorrect. F017A will still be closed, F048A will be open
C. Incorrect. F024A will be closed

Technical Reference(s) HC.OP-SO.BC-0001 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: RHRSYSE014 (As available)

Question Source: Bank # 56220
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

Comments:

NRC 423

TABLE 1
VALVE TABLE

Valve	Power	Auto Actuations	Interlocks
F017A B C D (LPCI Injection)	10B212 10B222 10B232 10B242	Auto open on LPCI init. if following condition exist: (1) LPCI init. present in respective RHR loop logic (2) Power is avail. on respective pump bus (3) Reactor press. less than 450 psig (may be overridden by "AUTO OPEN OVRD")	Rx press. must be less than 450 psig to open valve either MAN or AUTO. Placing Ch. B RSP to EMERG will close F017B & inhibit all associated automatic & OVLD protection Features. The LPCI injection valve must be 100% closed (in the respective loop with a LPCI initiation signal present) to OPEN: F027A(B), F024A(B), F016, F021
F047A B (HX Inlet)	10B212 10B222	None	Placing Ch. B RSP to EMERG transfers F047B to RSP
F003 A B (HX Inlet)	10B212 10B222	Throttle Valve	Placing Ch. B RSP to EMERG transfers F003B to RSP
F048A B (HX Bypass)	10B212 10B222	OPEN upon receipt of a LPCI initiation signal	Valve operation is inhibited for 3 minutes upon receipt of LPCI init. signal, valve interlocked open Ch. B RSP to EMERG transfers F048B to the RSP. OVLD protection can be bypassed in the OPEN direction (BYP IN OPEN)

NRC 4-23

TABLE 1
VALVE TABLE

Valve	Power	Auto Actuations	Interlocks
F015A,B (S/D cooling return F008 (S/D cooling suction)	10B481 10B242 10B242	Close on NS^4 signals 12.5" or 82 psig reactor pressure. These valves off NS^4 channels C and D manual initiation off Channel D.	NS^4 logic (D) must be reset following isolation. First 3 are throttle valves, Channel D RSP transfer SW to EMERG > F015B, and F008 Overload B/P with "OVLD BYP in OPEN (CLOSE)" Switch. F008 can be opened remotely only if MCC breaker Key Lock switch is in ARM position.
F010A B	10B232 10B242	Close on either auto <u>or</u> man LPCI init. signal Loop "C" logic for 10A Loop "D" logic 10B	(1) LPCI initiation logic must be reset to allow operation. Placing Channel D RSP to EMERG will close F010B. ("OVLD BYP in OPEN (CLOSE)")
F024A B (RHR A&B Full Flow Test)	10B212 10B222	Auto close on LPCI init. signal (auto or man) A logic for 24A; B logic for 24B. Can be overridden when all of the following exist concurrently: (1) LPCI init. signal present (2) F017A(B) 100% CLOSED. (AUTO CLOSE OVRD p/b on 650 panel)	Initiation signal must be reset to disable manual OVRD condition. Placing Channel B RSP to EMERG transfer control of F024B to RSP (and B/P all automatic control features and OVLD protection)F024A(B) .
F027A B (Torus Spray)	10B212 10B222	Auto close on LPCI init. logic "A" for 27A B logic 27B (man OVRD when following conditions exist concurrently (1) LPCI init. present (2) DW pres. (3) respective injection valve F017A(B) 100% closed)	OVERRIDDEN and illuminates when auto-close signal B/P. B RSP/EMERG closes F027B bypassing all auto control and OVLD protection. Man OVRD in effect till init. signal reset F027A(B) must be 100% closed to open F006A(B)

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	209001 K4.09	
	Importance Rating	3.3	

Knowledge of LOW PRESSURE CORE SPRAY SYSTEM design feature(s) and/or interlocks which provide for the following: Load Sequencing

Proposed Question: Common 24

Given the following conditions:

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

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

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

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

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

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

Proposed Answer: A

Explanation (Optional): IAW HC.OP-SO.BE-0001

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

B. Incorrect - They all start 6 seconds after diesel generator output breaker closes.

C. Incorrect – They all start 6 seconds after diesel generator output breaker closes.

D. Incorrect - They all start 6 seconds after diesel generator output breaker closes.

Technical Reference(s) HC.OP-SO.BE-0001 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: CSSYS0E005 (As available)

Question Source: Bank # 80663
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 X

Comments:

H24

5.3 **Core Spray System Automatic Initiation/Observation**

5.3.1. WHEN A(B,C,D) INIT AND SEALED-IN is on, **OBSERVE** the following:

A. Core Spray Pumps start after a 10 second time delay (6 seconds after Diesel Generator breaker closes following LOP as indicated by:

- PUMP A(B,C,D) RUNNING is on. ____
- AI-6357A(B,C,D) CORE SPRAY PUMP A(B,C,D) MOTOR AMPS indicates ≤ 85.5 amps. ____
- PI-R600A(B)-E21 CORE SPRAY SYSTEM A(B) PRESS indicates < 475 psig. ____

B. **ENSURE** HV-F015A(B) TEST RETURN VALVE is CLOSED. ____

C. WHEN PI-R605-C32 REACTOR PRESSURE < 461 psig, HV-F005A(B) LOOP A(B) INBD INJECT VLV will open. ____

D. **OBSERVE** FI-R601A(B)-E21 CORE SPRAY SYSTEM A(B) FLOW for the following:

- WHEN flow is > 775 gpm, HVF031A(B) LOOP A(B) MINIMUM FLOW ISLN VLV closes. ____
- Flow increases to 6350 gpm. ____

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	259002 K5.03	
	Importance Rating	3.1	

Knowledge of the operational implications of the following concepts as they apply to REACTOR WATER LEVEL CONTROL SYSTEM : Water level measurement.

Proposed Question: Common 25

Given the following:

- The plant is at 85% power
- All three Reactor Feed Pumps are in Auto
- RPV Narrow Range Level instruments indicate:
    - N004A = 34 inches
    - N004B = 35 inches
    - N004C = 35.5 inches

Which of the following describes the plant response to actual Reactor water level if a slow leak developed through the N004B detector equalizing valve eventually causing a gross failure of N004B?

Actual Reactor water level would...

A. lower 1 inch, then rise 0.5 inches.

B. rise 1 inch, then lower 0.5 inches.

C. lower 0.5 inch, then rise 1.5 inches.

D. rise 0.5 inch, then lower 1.5 inches.

Proposed Answer: C

Explanation (Optional): C. Correct

C. CORRECT - lower 0.5 inch, then rise 1.5 inches. initially N004B is selected since DFCS selects the MEDIAN RPV level signal when three good signals are available. With a leak through the N004B equalizing valve, N004B INDICATED level would begin to rise, resulting in a lowering of ACTUAL RPV level. As soon as N004B exceeded 35.5 inches INDICATED, N004C would become the MEDIAN RPV level signal. ACTUAL RPV level would have lowered 1/2 inch during this transition. When N004B gross fails, N004A (the lowest of the two remaining signals) will become the controlling level signal. RPV water

level will then rise since INDICATED level on N004A is 34 inches. This is a 1.5 inch rise from the previous level.

IAW FW Control Lesson Plan, Page 18 - On a failure of a narrow range Rosemount Level Detector (PDT-N004A, B or C) with the feedwater system in automatic three-element control from the Master Level Controller, level stays near its setpoint since the level signal is now the lower of the two good remaining level inputs. If another level transmitter were to fail, the remaining signal is now the controlling signal.

A. INCORRECT - lower 1 inch, then rise 0.5 inches. Level initially lowers by 1/2 inch.
B. INCORRECT - rise 1 inch, then lower 0.5 inches. Level initially lowers.
D. INCORRECT - rise 0.5 inch, then lower 1.5 inches. Level initially lowers.

Technical Reference(s) Engineering Drawing H-1-AE-ECS-0128-0 (Attach if not previously provided)
FW Control LP - NOH04FWCONTC-04

Proposed references to be provided to applicants during examination: none

Learning Objective: FWCONTE001 (As available)

Question Source: Bank # 53240
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

Comments:

OBJ. 11

d. On a loss of any Feed Flow signal (high or Low) the control circuit will transfer to single element control. The operator can detect this failure by noting that the Total Feed Flow signal on the FR-R607 recorder on panel 10C650C and by the DFCS TROUBLE overhead alarm. Additionally, the Operator Display screen(s) will digitally display the failed detector(s).

Fig. 3

e. On a failure of a narrow range Rosemount Level Detector (PDT N004A, B or C) with the feedwater system in automatic three-element control from the Master Level Controller, level stays near its setpoint since the level signal is now the lower of the two good remaining level inputs. If another level transmitter were to fail, the remaining signal is now the controlling signal.

1) The operator can detect the level transmitter has failed high because the DFCS TROUBLE overhead alarm will annunciate, the appropriate screen will display the failed transmitter output on the Operator Display Station, and indicate upscale on the respective LI-R603A, B, or C on panel 10C650C.

2) The operator can detect the level transmitter has failed low because the DFCS TROUBLE overhead alarm will annunciate, the appropriate screen will display the failed transmitter output on the Operator Display Station, and indicate downscale on the appropriate LI-R603A, B, or C on panel 10C650C.

Fig. 7

f. If the RFP control is in DP Control, RFP control would shift to Manual on a failure of the Reactor Pressure or Reactor Feed Pump Discharge Header Pressure transmitter. The operator can detect this failure from the DFCS TROUBLE and RFP TURBINE AUTO XFR TO MANUAL alarms, and the CRT display screen.

IV. INSTRUMENTATION ALARMS AND CONTROL

OBJ. 5a-d, 6

A. Instrumentation

1. Panel 10C650C - Main Control Room

a. Indications

1) Individual RFPT LOW PRESS CONTROL VLVS position indication: PI-1794A1, B1, C1 (0-10 x 10% scale)

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	261000 A3.02	
	Importance Rating	3.2	

Ability to monitor automatic operations of the STANDBY GAS TREATMENT SYSTEM including: Fan start

Proposed Question: Common 26

Given the following:

- The plant is operating at full power
- A Loss of Offsite power occurs
- Drywell pressure is 3.3 psig and rising
- "A" Emergency Diesel Generator fails to start
- "A" FRVS Vent Fan is in Auto Lead
- "B" FRVS Vent Fan is in Auto

Which one of the following describes the status of FRVS 2 minutes after the LOCA sequencers actuate?

(Assume NO operator action)

A. Only 3 Recirc Fans and NO Vent Fans start

B. Only 3 Recirc Fans and ONE Vent Fan start

C. Only 4 Recirc Fans and NO Vent Fans start

D. Only 4 Recirc Fans and ONE Vent Fan start

Proposed Answer: D

Explanation (Optional): D. Correct – Loss of power to "A" Bus will prevent the "A" and "E" Recirc fan from starting. The "F" Recirc fan will not start until 30 seconds after the LOCA sequencer actuates. The "A" vent fan has no power and the "B" will start after 45 second time delay. The B fan has a flow sensor in the A ductwork.

IAWHC.OP-SO.GU-0001 - FRVS Recirculation Fans AV213 through FV213 in AUTO and FRVS Vent Fan in AUTO LEAD will automatically start under any of the following conditions:

• High Drywell Pressure (1.68 psig).
• Low RPV Water Level (Level 2, - 38").
• Refueling Floor Exhaust Duct High Radiation
• Reactor Building Exhaust Air High Radiation

D. Correct.

A. Incorrect. B vent fan will be running.
B. Incorrect. 4 recirc fans start.
C. Incorrect. Only 3 recirc fans running and B recirc fan

Technical Reference(s) HC.OP-SO.GU-0001 (Attach if not previously provided)
LPNOH01RBVENTC-00

Proposed references to be provided to applicants during examination: none

Learning Objective: RBVENTE006 (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

Comments:

# Indicates Bezel alarms, yellow flashing backlights.
*Indicates no audible alarm.

OBJ. 3c,7,24c,28

c. Auto Features

1) The FRVS vent fan selected for Auto Lead operation will auto start on the same signals that start the FRVS recirc fans.

2) The LOCA Sequencers will start the Auto Lead fan 19 seconds after initiation.

3) The vent fan selected for Auto operation will auto start under the following conditions:

a) Automatic start signal present as described in 1 and 2 above and

4EC-3665

b) A low flow condition, (6950 scfm), exists on the Auto Lead fan for more than 45 seconds.

4) A low flow condition, (6950 scfm), for 30 seconds will trip the associated fan.

5) Deluge flow, (9.0 gpm), will trip the associated fan.

Table 4

M-57-1

9. Purge and Vent Valves

a. All valves are pneumatically operated and fail closed on a loss of power or air. The 26" Outboard Primary Containment to CPCS valve and the 24" Outboard Torus to CPCS valve are each bypassed by 2" motor operated vent valves that can be used to control containment pressure or supply a backup flowpath to the H_2 recombiners.

b. The RBVS supplies purge air through HD-9372C to the torus and drywell.

c. Suppression Chamber and Drywell atmospheres are vented or purged to RBVS via HD-9372A.

d. The Containment Atmosphere Control Vent and Purge Valves will automatically close on the following:

1) 1.68 psig Drywell pressure.

Deleted: 71

OBJ. 6,27

c. Automatic features

1) The following signals will start all six recirc fans:

a) -38" Rx Lvl.

b) 1.68 psig Drywell Pressure.

c) Reactor Building Vent. Exh. hi-hi rad $1x10^{-3}$ µci/cc.

d) Refuel floor Exh. hi-hi rad $2x10^{-3}$ µci/cc.

Obj. 3c, 24c

2) The LOCA Sequencers start recirc fans A,B,C and D 19 seconds after initiation. Fans E and F start 30 seconds after initiation.

3) After all six fans are started in response to an auto (LOCA) signal, E and F may be stopped and placed back in Auto. Then, if one of the running recirc fans (A,B,C or D) has a low flow condition, both E and F will restart with no time delay. Fans E and F are the only fans with this automatic start feature.

4) A low flow condition, (18,000 scfm for A-E and 15,000 for "F") on any operating fan for 15 seconds will trip that fan and provide a low flow alarm.

5) A low flow condition of 17,200 scfm for the A-D fans, if started due to an automatic signal, will auto start the E & F fans if they are in auto. Signals are from two independent flow elements, transmitters and switches one is sent to echo fan, (FSL-9377AA thru DA), and one is sent to the foxtrot fan, (FSL-9377AB-DB).

6) Deluge flow, (9.0gpm), will trip the associated fan.

M-84-1

Figure 1

Obj. 3c,24c

7. FRVS Vent System

a. The system consists of two 100% capacity units whose purpose is to maintain negative reactor building pressure during accident conditions. Each unit is identical and consists of:

Deleted: 71

#26

ELECTRIC POWER SUPPLIES FOR RBVS AND FRVS FANS

TABLE 2

FAN		POWER SUPPLY	CHANNEL
RBVS	AVH300	10B440-10B448	NOTE 1 D
	BVH300	10B41010B415	NOTE 1 A
	CVH300	10B430-10B437	NOTE 1 C
RBVE	AVH301	10B440-10B448	NOTE 1 D
	BVH301	10B420-10B426	NOTE 1 B
	CVH301	10B410-10B415	NOTE 1 A
FRVS	Recirc Unit AVH213	10B410	A → no power
	Recirc Unit BVH213	10B420	B
	Recirc Unit CVH213	10B430	C
	Recirc Unit DVH213	10B440	D
	Recirc Unit EVH213	10B450	A → no power
	Recirc Unit FVH213	10B460	B
FRVS	Vent Fan A	10B212	A → no power
	Vent Fan B	10B222	B

NOTE 1: The RBVS and RBVE FANS are supplied Power thru 1E and Non-1E Breakers. The associated channel 1E breakers trip on -38 Rx LVL, 1.68# Drywell Pressure, Reactor Bldg. Vent Exh. Hi-Hi rad $1x10^{-3}$µci/cc, and Refuel Floor Vent Exh. Hi-Hi rad $2x10^{-3}$µci/cc. The Non-1E breakers trip on undervoltage.

Deleted: 71

3.2.6 With respect to SACS cooling and EDG power supply, the following is required for FRVS Recirculation Fan Operability: _____

In OPERATIONAL CONDITION 1, 2, and 3, SACS and EDG backing are required support systems for all fans.

In OPERATIONAL CONDITION "*" (OPDRVs), SACS and EDG backing are required support systems for only three recirc fans (additional fans may be considered operable without SACS cooling when powered from an offsite source).

OPERATIONAL CONDITION "*" (handling recently irradiated fuel in the secondary containment) is prohibited by administrative procedures.

During fuel handling of fuel that is not recently irradiated and CORE ALTERATIONS, either the RBVS or one FRVS vent fan is required to be operating and capable of drawing air into the building and exhausting through a monitored pathway. **REFER TO** SH.OP-AP.ZZ-0108(Q) for details. **[80005080, 70020722]** _____

3.3 **Interlocks**

3.3.1 FRVS Recirculation Fans AV213 through FV213 in AUTO and FRVS Vent Fan in AUTO LEAD will automatically start under any of the following conditions: _____

- High Drywell Pressure (1.68 psig).
- Low RPV Water Level (Level 2, - 38").
- Refueling Floor Exhaust Duct High Radiation ($\geq 2x10^{-3}$ micro Ci/cc).
- Reactor Building Exhaust Air High Radiation ($\geq 1x10^{-3}$ micro Ci/cc).

3.3.2 With the FRVS Recirculation Fans EV213 and FV213 in AUTO, and the A, B, C, and D fans running due to receiving a LOCA OR High Rad signal, a low flow signal from a running FRVS Recirculation Fan will automatically start the EV213 and FV213 fans. _____

3.3.3 The PCIS signal is locked into the FRVS logic and resets after the PCIS is reset and the A, B, C, and D FRVS Recirculation fans are manually secured and all four fans are below the low flow setpoint. Placing the E and F Fans in "Lockout" will prevent an Auto Start while securing the A, B, C, and D fans. This action will inop 2 FRVS Fans and place the plant in an Action Statement. Tech Spec 3.6.5.3.2 should be reviewed to ensure compliance. _____

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	201003 K1.01	
	Importance Rating	3.2	

Knowledge of the physical connections and/or cause-effect relationships between CONTROL ROD AND DRIVE MECHANISM and the following: CRD Hydraulic system

Proposed Question: Common 27

Which one of the following describes scram valve response and indication on the full core display following a reactor scram?

A. The scram inlet valve opens faster than the scram outlet valve.
The blue scram light will be illuminated as soon as the scram inlet valve is fully open.

B. The scram outlet valve opens faster than the scram inlet valve.
The blue scram light will be illuminated as soon as the scram outlet valve is fully open.

C. The scram inlet valve opens faster than the scram outlet valve.
The blue scram light will be illuminated when both scram inlet AND outlet valves are fully open.

D. The scram outlet valve opens faster than the scram inlet valve.
The blue scram light will be illuminated when both scram inlet AND outlet valves are fully open.

Proposed Answer: D

Explanation (Optional): IAW CRD Lesson Plan NOH04CRDHYD-04 Section 10.h. & 10.g.3) - Scram Outlet Valve (XV-127) - fast acting globe valve that is opened by an internal spring and which exhausts water from the top of the drive piston. The scram outlet valve opens faster than the scram inlet valve because of a stronger spring and more rapid venting. A position indicator switch on the valve energizes a blue light on the Control Room panel 10C650C when the scram inlet and outlet valves are open.

D. Correct.

A. Incorrect – The scram light requires both valves open. The outlet valve opens faster.
B. Incorrect – The scram light requires both valves open
C. Incorrect – The outlet valve opens faster. The light will not illuminate unless both valves are open

Technical Reference(s) Lesson plan NOH04CRDHYD-04 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: CRDHYDE025 (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 X

Comments:

b) In OPERATIONAL CONDITIONS 1 and 2, control rod speeds are determined by measuring the stroke time between positions 04 and 08 (in both the insert and withdraw directions), then ensuring that the stroke time is $\geq$ 3.7 seconds and $\leq$ 4.4 seconds.

6) Additional requirements for controlling the position of the integral flow control devices are also provided following maintenance activities on the respective CRDM.

g. Scram Inlet Valve (XV-126) - Opens to supply pressurized water to the bottom of the drive piston.

1) Fast acting globe valve operated by an internal spring and system pressure.

2) Held closed by air pressure applied to the top of a diaphragm operator.

3) A position indicator switch on the valve energizes a blue light on the Control Room panel 10C650C when the scram inlet and outlet valves are open.

h. Scram Outlet Valve (XV-127) - fast acting globe valve that is opened by an internal spring and which exhausts water from the top of the drive piston. The scram outlet valve opens faster than the scram inlet valve because of a stronger spring and more rapid venting.

i. Scram Accumulator - stores sufficient energy to fully insert a control rod at lower reactor vessel pressures (less than 600 psig). At higher reactor vessel pressure, scram accumulator pressure is supplemented by reactor pressure.

1) The scram accumulator is a hydraulic cylinder with a free floating piston. The piston separates water from nitrogen gas.

2) A check valve in the scram accumulator charging line prevents loss of water pressure if supply pressure in the charging header is lost.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	226001 A3.01	
	Importance Rating	3.0	

Ability to monitor automatic operations of the RHR/LPCI: CONTAINMENT SPRAY SYSTEM MODE including: Valve operation

Proposed Question: Common 28

A LOCA has occurred and the following conditions exist.

- Reactor is scrammed
- Drywell Pressure is 10.4 psig and rising
- RHR Loop "A" is injecting to the RPV
- RHR Loop "B" is in Torus Spray

Which one of the following describes how RHR Containment Spray Isolation Valves BC-HV-F016A(B) and BC-HV-F021A(B) logic would function under these conditions?

A. NO valves could be opened simultaneously

B. ONLY the F016A & F021A could be opened simultaneously

C. ONLY the F016B & F021B could be opened simultaneously

D. BOTH the F016A & F021A and the F016B & F021B could be opened simultaneously

Proposed Answer: C

Explanation (Optional): C. Correct

IAW RHR Lesson Plan NOHO1RHRSYSC-06, Section IV.A.14.b. Page 43 - F016A(B) and F021A(B) are interlocked such that both valves can only be opened simultaneously when: There is a LPCI initiation signal present AND High drywell pressure condition exists AND F017A(B) IS 100% CLOSED.

In this case, a LPCI initiation signal is present and high drywell pressure exists. With the F017A is open and therefore the F016A &F021A cannot be opened simultaneously. With RHR loop "B" in Torus Cooling, the F017B is closed, therefore, the F016B & F021B can be opened simultaneously.

A. Incorrect. - The B loop valves can be opened simultaneously
B. Incorrect. – ONLY the B loop valves can be opened simultaneously
D. Incorrect - ONLY the B loop valves can be opened simultaneously

Technical Reference(s) RHR Lesson Plan NOHO1RHRSYSC-06 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: RHRSYSE011 (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

Comments:

2) OVLD/PWR FAIL (AMBER) status indication for all/valves adjacent to control switch on 10C650. (See IV.A.2.a.2))

NOTE: Containment Spray flow indication on 10C650C, PAMS section RHR Loop A and B Containment Spray flow meter indication FI-4462A,B. (0 - 12,000 GPM)

Obj. E011

b. Control

1) OPEN/CLOSE push-button controls are provided on 10C650 for F016A(B) and F021A(B).

2) F016A(B) and F021A(B) are interlocked such that both valves can only be opened simultaneously when:

There is a LPCI initiation signal present

AND

High drywell pressure condition exists

AND

F017A(B) IS 100% CLOSED.

Obj. E015

3) F016B and F021A(B) automatically close when their associated RSP transfer switch is placed in the EMERGENCY position

a) Channel A - F021A

b) Channel B - F016B, F021B

15. RHR Shutdown Cooling Return Isolation Valves (HV-F015A,B)

Obj. E004d, Obj. E011

a. Instrumentation

1) OPEN/CLOSE valve position indication is available for each valve on 10C650, and for HV-F015B on 10C399.

2) OVLD/PWR FAIL status indication is available adjacent to each valve control switch on 10C650. (See IV.A.2.a.2).

b. Control

1) INCR/DECR push-button controls are provided on 10C650 for F015A(B).

2) Torque switch protection can be bypassed by the operator when positioning the valve if the "OVLD BYP IN OPEN/OVLD BYP IN CLOSE" push buttons are depressed.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	234000 K3.03	
	Importance Rating	3.1	

Knowledge of the effect that a loss or malfunction of the FUEL HANDLING EQUIPMENT will have on following: Fuel handling operations.

Proposed Question: Common 29

The plant is in a refueling outage with the following conditions.

- Mode Switch in REFUEL
- All Control Rods are full in
- Refuel Platform over the core
- Fuel grapple is being raised carrying a fuel bundle

Then, the Refuel Bridge System "Rods Out Relay" contact fails indicating all rods are NOT full in.

Which one of the following describes the effect on the Main Hoist, if any, of the failed contact on fuel handling operations?

A. NO effect.

B. ONLY Main Hoist raise motion is prevented.

C. ONLY Main Hoist lower motion is prevented.

D. Main Hoist raise and lower motion is prevented.

Proposed Answer: D

Explanation (Optional): IAW Refueling lesson plan NOHO1 REFUEL-03 Table 2, first page.

D. Correct – with the system "seeing" one rod out with the platform over the vessel, no motion is permitted

A. Incorrect – ALL motion is prevented due to the system "seeing" one rod out.
B. Incorrect - ALL motion is prevented due to the system "seeing" one rod out.
C. Incorrect - ALL motion is prevented due to the system "seeing" one rod out.

Technical Reference(s) NOHO1 REFUEL-03 Table 2 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: REFUELE005 (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 X

Comments:

#29-NRC

Table 2

Motion Control (Manual Mode)

Common Hoist Motion Requirements

Following are the requirements for either hoist raise or hoist lower motion.

1.	Hoist motion is not allowed if a fault condition exists unless the fault is overridden by activating the Fault Override Keyswitch.
2.	Hoist motion is not allowed manually when in the semi-automatic or automatic mode. (Note: Movement of the joystick will stop semi-auto or auto operation.)
3.	Hoist motion is not allowed if not in a permissive secure travel zone unless bypassed by activating the Travel Override Push-button. If fuel loaded, the hoist may only be raised to the Up Limit when in Travel Override.
4.	{Fuel Hoist Interlock} Hoist motion is not allowed: When the Rods Out Relay contact (ROR) is opened indicating at least one control rod is withdrawn. AND The refueling platform is over the reactor vessel (RS1 contact is open)
5.	When the Interlock Override Key Switch is turned to the override position the hoist motion outputs from the PLC are disabled. (Note: the hoist is externally limited to slow speed in this condition.)
6.	When hoist motion is commanded the hoist screen will be displayed on the touchscreen.
7.	When hoist motion is commanded and all the interlocks are satisfied the output for the hoist inhibit is turned on.
8.	When hoist motion is commanded and all the interlocks are satisfied the output for the motor brake and redundant brake is turned on to energize the hoist brakes.
9.	Whenever the Hoist is in motion, the encoder is monitored to insure that the hoist position is changing in the direction expected for the direction of travel being commanded. Failure of the encoder value to change in the predicted manner will result in a Hoist Motion Fault. This error condition can be cleared by cycling power (Off, On) on the Start/Stop Station.
10.	When the hoist motion command is turned off, the output for hoist inhibit is turned off after the speed command decays below a preset value or a two second timer elapses.
11.	When the hoist motion command is turned off and one second after the output for hoist inhibit is turned off and the output for the hoist brakes is then turned off.

4.	**Bridge Reverse Stop Interlock #1** Bridge Reverse Stop Interlock #1 prevents reverse bridge motion and illuminates the Bridge Reverse Stop Interlock #1 light when: a. The Rod Out Relay (ROR) contact is open indicating at least one control rod is withdrawn in the reactor. AND b. A contact of bridge refueling switch RS1 opens indicating that the bridge is over the reactor vessel. AND c. An internal setpoint of the main hoist load weighing system is operated indicating a hoist fuel loaded condition (550 pounds or more on grapple.) NO
5.	**Bridge Reverse Stop Interlock #2** Bridge Reverse Stop Interlock #2 prevents reverse bridge motion and illuminates the Bridge Reverse Stop Interlock #2 light when: a. A contact of RS2 opens indicating that the bridge is over the reactor. AND b. A signal from the Reactor Manual Control System indicating that the Reactor mode switch is in Start Up. YES
6.	**Rod Block Interlock #1** Rod Block Interlock #1 prevents withdrawal of control rods from the reactor by opening output contact Rod Block #1 Interlock when the bridge is over the reactor vessel. This illuminates the Rod Block Interlock #1 light and occurs when: A contact of RS1 opens indicating that the bridge is over the reactor vessel. AND a. An internal setpoint of the main hoist load weighing system is operated indicating a hoist fuel loaded condition (535 pounds or more on grapple). NO AND A signal from the reactor manual control system indicating the Reactor Mode Switch in Refuel.
7.	**Rod Block Interlock #2** Rod Block Interlock #2 prevents withdrawal of control rods from the reactor when Reactor Mode switch is in Startup and the bridge is over the vessel. This is indicated on the machine by opening contacts to the PLC. The Rod Block Interlock #2 light illuminates when: A contact of RS2 opens indicating that the bridge is over the reactor vessel. AND A signal from the Reactor Manual Control System indicating that the Reactor mode switch is in Start Up. YES

Table 5
Refueling Interlocks

1.	**Fuel Hoist Interlock** The main fuel hoist motion will be inhibited and the Fuel Hoist Interlock light will be illuminated when the following conditions exist. The Rod Out Relay (ROR) contact is opened indicating at least one control rod is withdrawn in the reactor. AND The refueling platform is over the reactor vessel. i.e. (RS1 contact is open)
2.	**Frame Hoist Interlock** The frame hoist motion will be inhibited by opening the output contact (FRI) and the Frame Hoist Interlock light will be illuminated when the following conditions exist. The Rod Out Relay (ROR) contact is opened indicating at least one control rod is withdrawn in the reactor. AND There is a 400 pound load or greater detected on the Frame hoist. (fuel loaded) AND The refueling platform is over the reactor vessel. i.e. (RS1 contact is open)
3.	**Monorail Hoist Interlock** The Monorail hoist motion will be inhibited by opening the output contact (MRI) and the Monorail Hoist Interlock light will be illuminated when the following conditions exist. The Rod Out Relay (ROR) contact is opened indicating at least one control rod is withdrawn in the reactor. AND NO There is a 400 pound load or greater detected on the Monorail hoist. (fuel loaded) AND The refueling platform is over the reactor vessel. i.e. (RS1 contact is open)

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	214000 K4.01	
	Importance Rating	3.0	

Knowledge of (ROD POSITION INFORMATION SYSTEM) design feature(s) and or interlocks which provide for the following: Reed switch locations

Proposed Question: Common 30

The control rod reed switches are designed to illuminate a Full Core Display amber DRIFT indicator under which one of the following conditions?

NO rod motion command present and...

A. ONLY an odd reed switch closed.

B. ONLY an even reed switch closed.

C. an odd AND even reed switch closed.

D. the ROD DRIFT TEST PB is depressed.

Proposed Answer: A

Explanation (Optional): IAW HC.OP-SO.SF-0001, Att.6 Section B.2. – ROD DRIFT a control rod is changing position with no command signal or the ROD DRIFT TEST PB pressed while a rod is being moved (amber).

A. Correct -see HC.OP-SO.SF-0001 attachment # 6

B. Incorrect - requires odd reed switch without rod motion command signal.
C. Incorrect - requires odd reed switch without rod motion command signal.
D. Incorrect - Rod motion must be present

Technical Reference(s) HC.OP-SO.SF-0001 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: MANCONE002 (As available)

Question Source: Bank # 54400

Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 X
_____

Comments:

#30

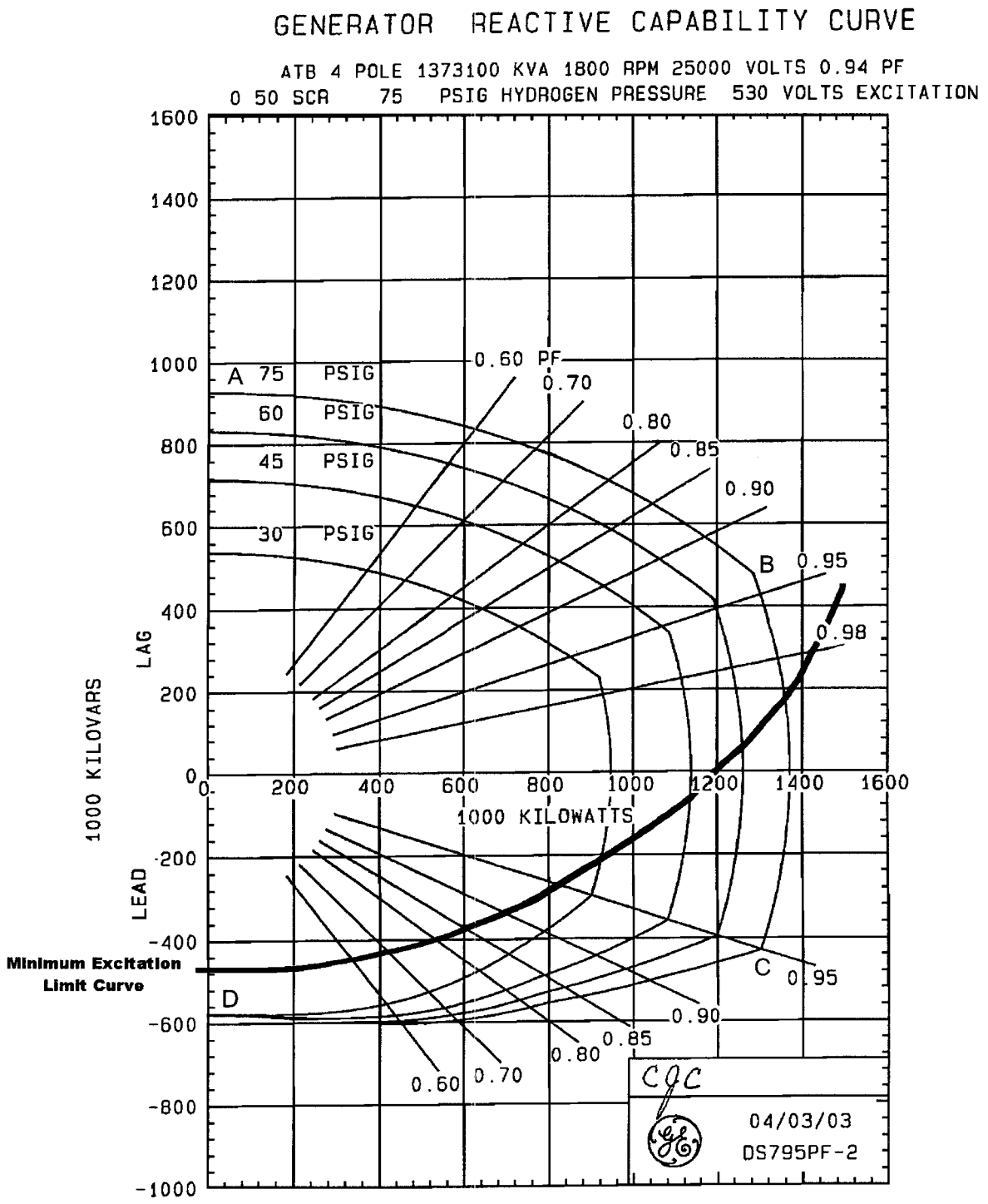
ATTACHMENT 6
(Page 3 of 3)
REACTOR MANUAL CONTROL SYSTEM OPERATION
REACTOR MANUAL CONTROL SYSTEM FUNCTIONAL DESCRIPTION

CONTROL/INDICATION	FUNCTION
B. **Full Core Display**	
1. Rod Select	Indicates the selected control rod (white light)
2. Rod DRIFT	Indicates a control rod is changing position with no command signal or the ROD DRIFT TEST PB pressed while a rod is being moved (amber).
3. Rod Position	Indicates FULL-OUT (red) and FULL-IN (green) position of control rods.
4. ACCUM Trouble	Indicates either low N_2 press (940 +85 -0 PSIG) or high H_2O level (>37cc) in that control rod's HCU accumulator (orange light).
5. SCRAM	Indicates each individual rod's inlet and outlet scram valves are both open (blue light).

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	245000 K5.07	
	Importance Rating	2.6	

Knowledge of the operational implications of the following concepts as they apply to MAIN TURBINE GENERATOR AND AUXILIARIES SYTEMS: Generator operations and limitations

Proposed Question: Common 31



CURVE AB LIMITED BY FIELD HEATING
CURVE BC LIMITED BY ARMATURE HEATING
CURVE CD LIMITED BY ARMATURE CORE END HEATING

GENERATOR REACTIVE CURRENT CAPABILITY CURVE

VDrive\Drawings\1000\Av1561l.vsd

Rev 9

Given the attached Generator Capability Curve and the following information:

- Hydrogen Pressure 65 psig
- Generator MVARS LAG = 200
- MWatts = 1290

The load dispatcher requests increasing MVARS from 200 to 400.

Determine what, if any, curve limitation will be exceeded if the MVARS are increased as requested.

A. NO curve limitation will be exceeded.

B. The curve limitation for Field heating will be exceeded.

C. The curve limitation for Armature heating will be exceeded.

D. The curve limitation for Armature Core End heating will be exceeded.

Proposed Answer: C.

Explanation (Optional): Per the curve (see markup), the region for Armature Heating would be exceeded (BC region)

Technical Reference(s) Gen Capability curve (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: MNGEN0E009 (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

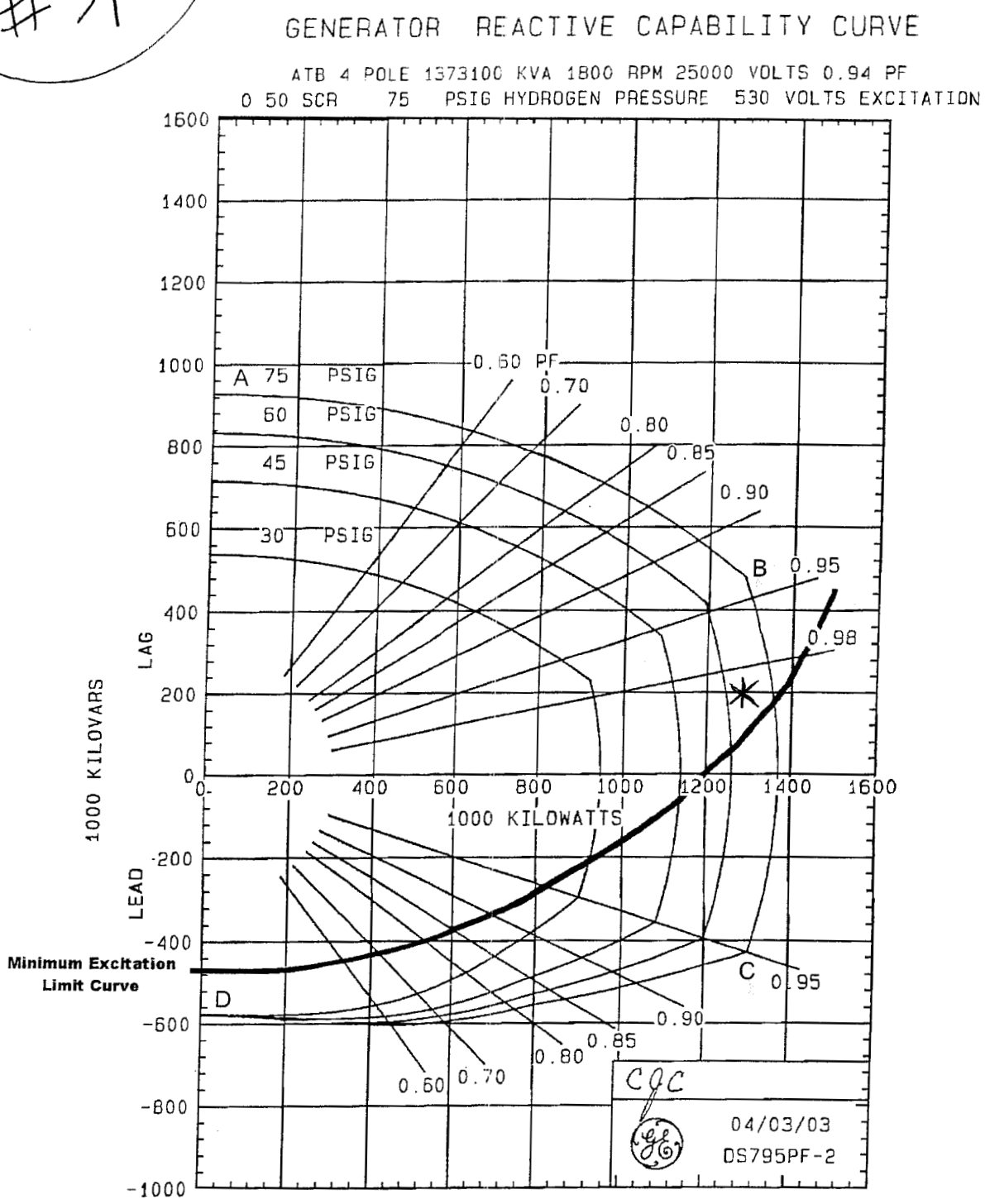
Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

Comments:

#31



CURVE AB LIMITED BY FIELD HEATING
CURVE BC LIMITED BY ARMATURE HEATING
CURVE CD LIMITED BY ARMATURE CORE END HEATING

GENERATOR REACTIVE CURRENT CAPABILITY CURVE

VDrive\Drawings\1000\Av1561l.vsd

Rev. 9

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	204000 K6.08	
	Importance Rating	3.5	

Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR WATER CLEANUP SYSTEM: PCIS/NSSSS.

Proposed Question: Common 32

The plant is operating at full power when an I&C surveillance inadvertently causes a loss of power to the 120VAC RPS Bus "A".

Which one of the following describes the effect of the loss, if any, on the Reactor Water Cleanup system?

A. NO effect.

B. ONLY RWCU PMP SUCT CONT INBD ISOLATION VALVE BG-HV-F001 will close.

C. ONLY RWCU PMP SUCT CONT OUTBD ISOLATION VALVE BG-HV-F004 will close.

D. RWCU PMP SUCT CONT INBD ISOLATION VALVE BG-HV-F001 will close.
RWCU PMP SUCT CONT OUTBD ISOLATION VALVE BG-HV-F004 will close.

Proposed Answer: B

Explanation (Optional): IAW RWCU Lesson Plan NOHO4RWCU00C-01 – Section III.B.3.b.5) - A loss of power to a channel of the (NS4) Leak Detection System will cause the respective valve to isolate. The power supplies are 120 VAC RPS Bus A for channel A (NS4) (shuts F001) and 120 VAC RPS Bus B for channel D (NS4) (shuts F004).

B. Correct

A. Incorrect. - The F001 will shut.
C. Incorrect. – The F004 will remain open
D. Incorrect – The F004 will remain open

Technical Reference(s) RWCU Lesson Plan NOHO4RWCU00C-01 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: RWCU00E013 (As available)

Question Source: Bank # _____
Modified Bank # _____ (Note changes or attach parent)
New X

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 X

Comments:

- RWCU Pipe Chase Areas

g) RWCU high differential flow 56.3 gpm after 45 second time delay (LDS via NS4)

h) Redundant Reactivity Control System actuation. [either "A" or "B" RRCS channel]

3) The HV-F001 will also auto isolate (in addition to the common) on:

a) "A" NS4 manual isolation pushbutton armed and depressed

b) "A" SLC pump start

4) The HV-F004 will also auto isolate (in addition to the common) on:

a) "D" NS4 manual isolation pushbutton armed and depressed

b) "B" SLC pump start

c) Non-regenerative heat exchanger high outlet (F/D inlet) temperature (140°F) (NS4)

5) A loss of power to a channel of the (NS4) Leak Detection System will cause the respective valve to isolate. The power supplies are 120 VAC RPS Bus A for channel A (NS4) (shuts F001) and 120 VAC RPS Bus B for channel D (NS4) (shuts F004).

6) The power monitor relay for each channel has a test pushbutton on Panels H11-P609 and H11-P611 for A and D respectively. Depressing a test pb will simulate a loss of power to that channel and cause the respective RWCU valve to isolate.

7) Both valves have status indicating lights on the NS4/PCIS status board at 10C650D (OPEN/CLOSED)

8) In order to ensure long term compliance with NRC GL 89-10, a DCP was completed to modify the F001 and F004 valves. The motor size for the valves was raised,. The breakers for the valves were replaced with larger amp breakers. The magnetic trip for the breakers was set at higher and the overload heaters were also raised..

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	201001 A1.10	
	Importance Rating	2.8	

Ability to predict and/or monitor changes in parameters associated with operating the CONTROL ROD DRIVE HYDRAULIC SYSTEM controls including: CRD cooling water flow.

Proposed Question: Common 33

Given the following conditions:

- Reactor operating at 100% rated power
- CRD flow control valve in automatic
- The RO throttles closed on the Pressure Control Valve (BF-HV-F003) for two seconds

Which one of the following describes how parameters will stabilize when this CRD system transient is over?

A. Differential pressure between the Drive Water Header and the RPV will lower. Cooling water flow will lower.

B. Differential pressure between the Drive Water Header and the RPV will rise. Cooling water flow will lower.

C. Differential pressure between the Drive Water Header and the RPV will lower. Cooling water flow will remain the same.

D. Differential pressure between the Drive Water Header and the RPV will rise. Cooling water flow will remain the same.

Proposed Answer: D

Explanation (Optional): IAW CRD Lesson Plan NOH04CRDHYD-04, Section II.B.6, page 21 - The motor-operated PCV is positioned to maintain approximately 265 psid between drive water header pressure and reactor pressure (senses above core plate pressure). To RAISE the differential pressure between the drive water header and the RPV, press the DECREASE PB to cause the motor-operated PCV to travel in the closed direction

Throttling the drive water pressure control valve closed will increase pressure in the line. Cooling water flow will be unaffected due to a different flowpath from the system pumps(see P&ID M-46-1)

D. Correct.

A. Incorrect. D/P will rise. Cooling water flow will be constant
B. Incorrect. Cooling water flow will be constant
C. Incorrect. – D/P will rise

Technical Reference(s) P&ID M-46-1 (Attach if not previously provided)
CRD Lesson Plan
NOH04CRDHYD-04

Proposed references to be provided to applicants during examination: None

Learning Objective: CRDHYD0E006 (As available)

Question Source: Bank # 56276
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X
_____

Comments:

#33

2) One manually operated PCV in parallel, which allows continued operation if the motor-operated PCV is out of service.

c. Instrumentation and Control

1) D/P between the drive water header and reactor pressure is continuously monitored by:

a) Local indication

b) Reactor control panel indication on CRIDS

OBJ. 7

2) The motor-operated PCV is positioned to maintain approximately 265 psid between drive water header pressure and reactor pressure (senses above core plate pressure).

3) Pressure control is accomplished by the use of four PBs;

FIG. 7

a) Two PBs (OPEN/CLOSE) will cause the motor-operated PCV to travel either full open or full closed, depending on which PB is depressed.

b) Two PBs (INCREASE/DECREASE) will cause the motor-operated PCV to throttle open or closed for as long as the PB is depressed.

- To RAISE the differential pressure between the drive water header and the RPV, press the DECREASE PB to cause the motor-operated PCV to travel in the closed direction.
- To LOWER the differential pressure between the drive water header and the RPV, press the INCREASE PB to cause the valve to travel in the open direction.

4) The OPEN PB is backlit to indicate the motor-operated PCV (HV-F003) is <u>not</u> full closed and the CLOSED PB is "backlit" to indicate the valve is <u>not</u> full open.

d. Automatic Actuations and Interlocks

- None -

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	239001 A2.12	
	Importance Rating	4.2	

Ability to (a) predict the impacts of the following on the MAIN AND REHEAT STEAM SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: PCIS/NSSSS actuation.

Proposed Question: Common 34

The plant is in OPCON 4 with the following conditions:

- Mode Switch is in SHUTDOWN
- All RPS Channels are reset
- Main Condenser Pressure is 30" HgA
- Main Turbine is tripped
- All MSIVs are open

Then, NSSSS channels A and B inadvertently actuate.

The cause of the isolation signal has been cleared.

Which one of the following describes the response of the MSIVs when NSSSS actuated and what actions must now be taken to reopen an MSIV that had closed IAW HC.OP-SO.SM-0001?

A. ONLY the Inboard MSIVs CLOSE.
The MSIV control switches must be placed in the CLOSED position then the NSSSS LOGIC RESET PBs must be depressed.

B. ONLY the Inboard MSIVs CLOSE.
The NSSSS LOGIC RESET PBs must be depressed then the MSIV control switches must be placed in the CLOSED position.

C. All MSIVs CLOSE.
The MSIV control switches must be placed in the CLOSED position then the NSSSS LOGIC RESET PBs must be depressed.

D. All MSIVs CLOSE.
The NSSSS LOGIC RESET PBs must be depressed then the MSIV control switches must be placed in the CLOSED position.

Proposed Answer: C

Explanation (Optional): C. Correct

IAW NSSSS Lesson Plan NOH04NSSSSOC-02 - All MSIV control switches must be in

"CLOSE" to enable reset of the isolation. This prevents automatic reopening of the MSIVs following isolation reset. The initiating condition must have cleared to reset the isolation logic.

NSSSS Channels A or C and B or D must trip to close the MSIV.

A. Incorrect. All MSIVs close
B. Incorrect. All MSIVs close. The MSIV control switches must be placed in CLOSE first
A. Incorrect. The MSIV control switches must be placed in CLOSE first

Technical Reference(s) NOH04NSSSSOC-02 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: MSSTEAME012 (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

Comments:

H34

3. Main steam isolation valve logic

Figure 6

a. All four NSSSS channels provide input to the MSIV trip logic. Each trip channel is single coincidence, requiring only one of the initiating conditions to be present to de-energize its respective initiation relay.

b. Each MSIV has two normally energized, solenoid-actuated pilot valves controlling air to the valve operator. Channels A and C control the position of two series arranged contacts in the power supply to one of the two pilot valves. Channels B and D control the position of similar contacts to the other.

c. The contacts are normally closed, maintaining the solenoids energized and allowing normal operation of the MSIVs.

d. An MSIV logic trip will de-energize the associated initiation relay by opening its associated contact and interrupting power to the solenoid. Tripping of either channel associated with that solenoid will interrupt power.

   1) A <u>or</u> C for one solenoid

   2) B <u>or</u> D for the other solenoid

e. When the solenoid is de-energized, the pilot valve will reposition, blocking the air supply to the valve operator. Due to the air supply piping arrangement, <u>*both*</u> solenoids must de-energize to block the air supply and vent the operator of a MSIV. Both inboard and outboard MSIVs operate in the same manner.

f. This logic arrangement is referred to as "one-out-of-two-twice".

   1) One of the two channels controlling the power to each solenoid must trip (one-out-of-two).

   2) Both solenoid operated pilot valves must reposition to effect MSIV closure (twice).

   3) Channels A or C and B or D must trip to close the MSIV.

g. Manual isolation
All MSIVs (both inboard and outboard) will close by depression of A or C and B or D manual initiation pushbuttons.

#34

Objective 5c

c. Reset

1) Accomplished via RESET pushbutton on Control Room Panel 10C651C.

2) Each "tripped" channel logic RESET must be pressed

3) Indication of the main steam line drain valve isolation logic status is available on Control Room Panel 10C609 (inboard) and 10C611 (outboard).

d. Interlocks

1) Isolation function, once initiated, is sealed-in ensuring isolation goes to completion, even if the initiating condition clears.

2) To effect a manual isolation, the stop collar on the associated pushbutton must be rotated clockwise and the pushbutton must be pressed. This interlock ensures no single operator action can effect an isolation.

Objective 5c
Figure 7b

3) All MSIV control switches must be in "CLOSE" to enable reset of the isolation. This prevents automatic reopening of the MSIVs following isolation reset. The initiating condition must have cleared to reset the isolation logic.

4) Main condenser low vacuum bypass

PN1-B21-1090-0062 SHT 11, 13; PN1-C71-1020-0006 SHT 8, 10
Figure 7a

a) Four keylock NORM-BYPASS maintained contact switches

- One for each channel, B21H-S25A – D

Objective 5c

- Key removable in NORM
- A and C channels 10C609
- B and D channels 10C611

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	215002 A3.01	
	Importance Rating	3.1	

Ability to monitor automatic operations of the ROD BLOCK MONITOR SYSTEM including: Four rod display: BWR-3,4,5.

Proposed Question: Common 35

Given the following conditions:

- The plant is operating at 100% power
- NO LPRMs are bypassed
- Control Rod 30-31 is selected and is at notch position 10.

Rod Block Monitor Channel B is then bypassed with the joystick.

Which of the following describes the effect on the FOUR ROD DISPLAY indication?

A. NO LPRM BYPASSED lights are illuminated.

B. All LPRM BYPASSED lights are illuminated.

C. ONLY the B and D level LPRM BYPASSED lights are illuminated.

D. ONLY the B level LPRM BYPASSED lights are illuminated.

Proposed Answer: C

Explanation (Optional):

C - Correct

IAW RBM Lesson plan NOH04RBMSYS-00, Page 29,30 – Section IV.A.2.b. - The amber LPRM level A/B/C/D BYPASSED lights (on the Four Rod Display) illuminate when <u>any</u> of the following conditions exist: The RBM BYPASS switch is in either the CH. A <u>or</u> CH. B position (<u>only</u> the BYPASSED lights for the LPRM detectors associated with the RBM channel will illuminate. RBM channel A: <u>only</u> the A <u>and</u> C level LPRM detector BYPASSED lights will illuminate. RBM channel B: <u>only</u> the B <u>and</u> D level LPRM detector BYPASSED lights will illuminate)

A. Incorrect. The B and D lights will illuminate

B. Incorrect. ONLY the B and D lights will illuminate

D. Incorrect. The D lights will also illuminate

Technical Reference(s) NOH04RBMSYS-00 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: LPRM00E002 (As available)

Question Source: Bank # _____
Modified Bank # _____ (Note changes or attach parent)
New X

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

Comments:

#35

RBM channel output trips should be bypassed if average core power (as sensed by the reference APRM channel) is less than 30% of rated.

d. BYPASS light

The white BYPASS light illuminates when any of the three RBM channel BYPASS conditions exist.

e. HIGH light

The white HIGH light illuminates when the "HIGH" UPSC Trip Unit trip setpoint (0.66(W-ΔW) + 65%) is being enforced.

f. INTMD light

The white INTMD light illuminates when the INTERMEDIATE UPSC Trip Unit trip setpoint (0.66(W-ΔW) + 57%) is being enforced.

g. LOW light

The white LOW light illuminates when the LOW UPSC Trip Unit trip setpoint (0.66(W-ΔW) + 49%) is being enforced.

h. SETUP ALARM SETPOINT light

The white SETUP ALARM SETPOINT light illuminates when conditions permit the UPSC Trip Unit trip setpoint to be "SETUP" to the next higher value (INTERMEDIATE or HIGH).

2. Control Room Panel 10C650C

OBJ. 2
OBJ. 3a, b
Fig. 8 and 9

a. IRM/APRM/RBM recorders (2)

RBM channel Averaging Circuit output (0 to 125%) will be recorded on the IRM/APRM/RBM recorders (RED pen) when RBM A and/or RBM B is chosen on the RECORDER SELECT bezel.

b. LPRM Level A/B/C/D BYPASSED lights (16)

The amber LPRM level A/B/C/D BYPASSED lights (on the Four Rod Display) illuminate when any of the following conditions exist:

1) A peripheral control rod has been selected for movement (all BYPASSED lights will illuminate), or

2) An LPRM detector(s) has been bypassed and a control rod adjacent to that LPRM detector has been selected for

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movement (only the BYPASSED light for the associated LPRM detector(s) will illuminate), or

3) Average core power is less than 30% of rated, as sensed by the reference APRM channel (all BYPASSED lights will illuminate), or

4) The RBM BYPASS switch is in either the CH. A or CH. B position (only the BYPASSED lights for the LPRM detectors associated with the RBM channel will illuminate. RBM channel A: only the A and C level LPRM detector BYPASSED lights will illuminate. RBM channel B: only the B and D level LPRM detector BYPASSED lights will illuminate), or

5) An LPRM detector output signal is downscale (only the BYPASSED light for the associated LPRM detector will illuminate), or

6) A 3-string or 2-string control rod has been selected for movement (depending on the selected control rods location in the core, the BYPASSED lights for all absent LPRM detectors will illuminate).

c. LPRM meters (16)

1) The LPRM meters provide indication (0 to 125%) of the local power levels surrounding the control rod selected for movement. These indications are the signals supplied to the RBM channel Averaging Circuit.

2) When a 3-string or 2-string control rod is selected for movement, the LPRM meters for all absent LPRM detectors will indicate "zero" (fully downscale).

3. Control Room Panel 10C608

Fig. 7
OBJ. 2
OBJ. 3a, b

NOTE: The following indications on Panel 10C608 are identical for both RBM channels.

a. BYPASS light

The white BYPASS light illuminates when any of the three RBM channel BYPASS conditions exist. All output trip signals (control rod withdrawal block signals) from the respective RBM channel should be bypassed when this light is illuminated.

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Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	241000 A4.07	
	Importance Rating	3.5	

Ability to manually operate and/or monitor in the control room: Main stop/throttle valves (operation).

Proposed Question: Common 36

Given the following conditions:

- The Main Turbine is reset.
- As part of the Main Turbine Startup, the RO has depressed the CONTROL, PRE-WARMING, SHELL WARMING ON, on the DEHC HMI.

Which of the following directly occurs as a result of these actions?
(Assume VPL Limiter is at its normal setting of 100%)

A. Intercept Valves OPEN

B. Turbine Stop Valves OPEN

C. Turbine Control Valves OPEN

D. Intermediate Stop Valves OPEN

Proposed Answer: C

Explanation (Optional): C. Correct

IAW HC.OP-AC-0001 step 5.2.7, The Control Valves OPEN, all others close or go closed.

Shell warming is initiated by depressing the CHEST-SHELL WARMING SHELL pushbutton. This opens the Turbine Control Valves and the pilot valve to the #2 Turbine Stop Valve. The stop valves remain closed. Intermediate Stop Valves are open on reset and go closed. The Intercept Valves remain closed.

A. Incorrect. IVs remain closed
B. Incorrect. Turbine Stop Valves remain closed
D. Incorrect. ISVs go closed

Technical Reference(s) HC.OP-AC-0001 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: MNTURBE024 (As available)

Question Source: Bank # 62008
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

Comments:

H36

5.2.7 (continued)

NOTE
While chest OR shell warming, the temperature limits of Att. 2 should be observed.

K. **SELECT** [Control], [Pre-Warming] ____

L. **SELECT** Shell Warming [ON] ____

M. **SELECT** [Control], [Valve Limiters] ____

N. **ADJUST** Valve Position Limiter, VPL Setpoint to 100.0%. ____

O. **OBSERVE** the following:

- All Control Valves open fully. ____
- All Intermediate Stop Valves (ISV) go closed. ____
- All Intercept Valves (IV) remain closed. ____
- All Main Stop Valves (MSV) remain closed. ____

P. **VERIFY** proper valve positions utilizing Attachment 7. ____

Q. **DETERMINE** required time period for pre-warming using Attachment 5. ____

(continued on next page)

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	223001 G2.2.39	
	Importance Rating	3.4	

Equipment Control: Knowledge of less than or equal to one hour Technical Specifications for a system (Primary Cont)

Proposed Question: Common 37

The plant is operating at 75% power and you note the following readings while taking logs at the start of your shift:

- Drywell average temperature is 137 degrees F.
- Drywell Pressure is 1.2 psig.
- Suppression Pool water level is 74 inches.

Which of the following must be restored to within Technical Specifications limits within ONE hour to preclude further actions?

A. Suppression Pool Level ONLY.

B. Suppression Pool Level AND
Drywell Pressure ONLY.

C. Drywell Average Temperature AND
Suppression Pool Level ONLY.

D. Drywell Average Temperature AND
Suppression Pool Level AND
Drywell Pressure.

Proposed Answer: A

Explanation (Optional):

A. Correct

IAW TS 3.6.2 - The suppression chamber shall be OPERABLE with: With an indicated water level between 74.5" and 78.5". Per Action a) With the suppression chamber water level outside the above limits, restore the water level to within the limits within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

B. Incorrect. - Action is required within 1 hour if DW pressure exceeds 1.5 psig

C. Incorrect. - Action is required in 8 hours for exceeding Drywell average temperature limits.
D. Incorrect. – Action is required in 8 hours for exceeding Drywell average temperature limits.

Technical Reference(s) TS 3.6.2.1, 3.6.1.7, 3.6.1.6 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: PRICONE009 (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 X

Comments:

#37

CONTAINMENT SYSTEMS

DRYWELL AVERAGE AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.7 Drywell average air temperature shall not exceed 135°F.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With the drywell average air temperature greater than 135°F, reduce the average air temperature to within the limit within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.7 The drywell average air temperature shall be the volumetric average of the temperatures at the following locations and shall be determined to be within the limit at least once per 24 hours:

	Elevation Zone	Approximate Azimuth*
a.	86'11"-112'8" (under vessel)	90°, 225°, 90°, 270°
b.	86'11"-111'10" (outside of pedestal)	135°, 300°, 100°, 190°
c.	111'10"-139'2"	55°, 240°, 155°, 315°
d.	139'2"-168'0"	45°, 215°, 0°, 90°, 180°, 270°
e.	168'0"-192'7"	95°, 130°, 300°, 355°, 45°, 225°

* At least one reading from each elevation zone is required for a volumetric average calculation.

#37

3.6.2.1 The suppression chamber shall be OPERABLE with:

a. The pool water:

1. With an indicated water level between 74.5" and 78.5" and a

2. Maximum average temperature of 95°F during OPERATIONAL CONDITION 1 or 2, except that the maximum average temperature may be permitted to increase to:

a) 105°F during testing which adds heat to the suppression chamber.

b) 110°F with THERMAL POWER less than or equal to 1% of RATED THERMAL POWER.

3. Maximum average temperature of 95°F during OPERATIONAL CONDITION 3, except that the maximum average temperature may be permitted to increase to 120°F with the main steam line isolation valves closed following a scram.

b. A total leakage between the suppression chamber and drywell of less than the equivalent leakage through a 1-inch diameter orifice at a differential pressure of 0.80 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With the suppression chamber water level outside the above limits, restore the water level to within the limits within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With the suppression chamber average water temperature greater than 95°F and THERMAL POWER greater than 1% of RATED THERMAL POWER, restore the average temperature to less than or equal to 95°F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours, except, as permitted above:

1. With the suppression chamber average water temperature greater than 105°F during testing which adds heat to the suppression chamber, stop all testing which adds heat to the suppression chamber and restore the average temperature to less than 95° within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With the suppression chamber average water temperature greater than 110°F, place the reactor mode switch in the Shutdown position and operate at least one residual heat removal loop in the suppression pool cooling mode.

3/4.6.5 SECONDARY CONTAINMENT
SECONDARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.5.1 SECONDARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3 and *.

ACTION:

Without SECONDARY CONTAINMENT INTEGRITY:

a. In OPERATIONAL CONDITION 1, 2 or 3, restore SECONDARY CONTAINMENT INTEGRITY within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In Operational Condition *, suspend handling of recently irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.1 SECONDARY CONTAINMENT INTEGRITY shall be demonstrated by:

a. Verifying at least once per 24 hours that the reactor building is at a negative pressure.

b. Verifying at least once per 31 days that:

1. All secondary containment equipment hatches and blowout panels are closed and sealed.

2. a. For double door arrangements, at least one door in each access to the secondary containment is closed.

   b. For single door arrangements, the door in each access to the secondary containment is closed except for routine entry and exit.

3. All secondary containment penetrations not capable of being closed by OPERABLE secondary containment automatic isolation dampers/valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic dampers/valves secured in position.

c. At least once per 18 months:

1. Verifying that four filtration recirculation and ventilation system (FRVS) recirculation units and one ventilation unit of the filtration recirculation and ventilation system will draw down the secondary containment to greater than or equal to 0.25 inches of vacuum water gauge in less than or equal to 375 seconds, and

*When recently irradiated fuel is being handled in the secondary containment and during operations with a potential for draining the reactor vessel.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	259001 A1.06	
	Importance Rating	2.7	

Ability to predict and/or monitor changes in parameters associated with operating the REACTOR FEEDWATER SYSTEM controls including: Feedwater heater level.

Proposed Question: Common 38

The plant is at full power when the operators notice an increasing shell level in the 2A Feedwater Heater (FWH).

An operator is sent out to the local panel to operate FWH 2A drain valve manually when the Hi-Hi shell level setpoint is reached for the 2A FWH.

Which one of the following describes how the 2A FWH is affected?

A. Extraction steam to the heater is isolated AND Condensate flow through the heater tube side is isolated.

B. Condensate flow through the heater tube side is isolated. NO other automatic actions occur.

C. Extraction steam to the heater is isolated AND the cascading drain flow from the 3A FWH is isolated.

D. Condensate flow through the heater tube side is isolated AND the cascading drain flow from the 3A FWH is isolated.

Proposed Answer: D

Explanation (Optional): D. Correct

IAW FWH Lesson Plan NOH04FWHEATC-01 Section III.C.2.b)- Page 32, If the "Hi-Hi" setpoint is reached, the following actions will occur: [For the 1,2 (A, B, C) only, the isolation will occur after a ten second time delay once the Hi-Hi level is reached]

- FWHTR's 1,2 (A, B, C) - Condensate flow through the heater tube side is isolated (this reduces the extraction flow to that heater) AND Cascading drains from heater 3 (A, B, C) are isolated

A. Incorrect – no extraction steam isolation
B. Incorrect – Also, cascading drain flow from the 3A FWH is isolated.
D. Incorrect – no extraction steam isolation

Technical Reference(s) NOH04-FWHEATC-01 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: FWHEATE008 (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 X

Comments:

The level control valve is positioned as a function of heater 3 (A, B, C) level.

(5) Extraction steam to FWHTR 2 (A, B, C) is also condensed by secondary condensate pump flow before being mixed with the incoming drain flow. The condensate flows directly to the external drain cooler where it is subcooled by secondary condensate pump flow through the drain cooler. The drain flow is then directed to the main condenser via a motor operated isolation valve (HV-1469 A, B, C) and a level control valve (LV-1464 A, B, C).

(a) The level control valve is positioned as a function of FWHTR 2 level.

(b) The drain cooler is maintained full of water during normal operation.

(6) Secondary condensate pump flow through FWHTR 1 (A, B, C) condenses its extraction steam which is then drained to the main condenser via a loop seal.

Obj. 4d

b) Abnormal

(1) An increasing heater shell level (in FWHTR's 2-6 A, B, C) will open the associated alternate drain (dump) valve to restore water level.

(a) The dump valve is modulated as a function of the high level in its respective heater.

(b) The drains are directed to the main condenser shell.

(2) If heater level reaches the "Hi" setpoint, the alternate drain valve will fail open.

(3) If the "Hi-Hi" setpoint is reached, the following actions will occur: [For the 1,2 (A, B, C) only, the isolation will occur after a ten second time delay once the Hi-Hi level is reached]

(a) FWHTR's 3-6 (A, B, C)

(i) Extraction steam to that heater is isolated.

(ii) Cascading drain flow from the upstream heater is isolated (level control valve fails closed).

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Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295018 AK1.01	
	Importance Rating	3.5	

Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: Effects on component/system operations.

Proposed Question: Common 39

Given the following:

- The plant is operating at 100% reactor power.
- "A" SACS Loop is supplying TACS with "A" & "C" SACS Pumps running.
- "D" SACS Pump is running supplying the "B" SACS Loop loads.
- A trip of the "A" SACS Pump has occurred.
- The "C" SACS Pump remains running.
- The idle "B" SACS Pump has auto started.

Which one of the following describes the affects, if any, on the SACS/TACS system? (assume NO operator action)

SACS/TACS Isolation valves, EG-HV-2522A __(1)__ and EG-HV-2522C __(2)__

A. (1) CLOSES
(2) CLOSES

B. (1) remains OPEN
(2) remains OPEN

C. (1) remains OPEN
(2) CLOSES

D. (1) CLOSES
(2) remains OPEN

Proposed Answer: D

Explanation (Optional): D. Correct IAW HC.OP-SO.EG-0001interlocks section

D. Correct - "A" valve receives a close signal from the "A" pump stop input. The "C" valve does not close because the "C" pump is still running.

A. Incorrect - The "C" valve does not close because the "C" pump is still running.

B. Incorrect – "A" valve receives a close signal from the "A" pump stop input
C. Incorrect - "A" valve receives a close signal from the "A" pump stop input. The "C" valve does not close because the "C" pump is still running.

Technical Reference(s) HC.OP-SO.EG-0001interlocks section (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: STACS0E018 (As available)

Question Source: Bank # 55907
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

Comments:

#39

IMMEDIATE OPERATOR ACTIONS

NONE

AUTOMATIC ACTIONS

IF	THEN
Expansion tank Low-Low-Low level (33%) (See NOTE 1)	1. TACS supply and return valves from affected SACS loop close. (HV-2522/2496) 2. The affected loop SACS cross-connecting valves for the Fuel Pool Heat Exchanger AND Containment Instrument Gas Compressors close.
Low SACS to TACS flow from In-Service loop (9,900 gpm)	1. In-service loop SACS pumps transfer to MAN. 2. Standby Loop SACS Pumps Start IF in AUTO. 3. TACS supply and return valves on standby SACS loop open IF respective pump AND valve control is in AUTO.
Low Pump d/p (34.3 psid)	Respective pump trips (Equates to 17,000 gpm pump flow)
Pump Trip (Automatic or Manual) AND TACS Override NOT enabled.	Associated TACS supply and return valve shut. (HV-2522/2496)
SACS Supply Temperature High (88°F)	Associated Heat Exchanger Bypass Valves trip closed. (HV-2457/2517)
Low TACS pressure at supply OR return accumulators (22 psig).	TACS HV-2522E/F close

NOTES:

1. SACs Expansion Tank Values: Low Level- 65.0%
   Low Low Level- 44.0%
   Low Low Low Level- 33.0%

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295006 AK1.03	
	Importance Rating	3.7	

Knowledge of the operational implications of the following concepts as they apply to SCRAM : Reactivity control.

Proposed Question: Common 40

What are the operational implications of RPS having a built in time delay of 10 seconds before allowing a SCRAM to be manually reset?

A. To ensure all the control rods fully insert.

B. To allow the Scram Air header to repressurize.

C. To allow reactor water level to recover above the scram setpoint.

D. To ensure the Scram Discharge Volume vent and drain valves are fully closed.

Proposed Answer: A

Explanation (Optional): IAW RPS Lesson Plan NOH01RPS00C-05, Page 37, Section E.6. - Manual reset of a full scram is inhibited for 10 seconds after initiation to ensure control rods drive to FULL IN position.

A. Correct.

B. Incorrect. Not the reason cited in Lesson plan

C. Incorrect. Not the reason cited in Lesson plan

D. Incorrect. Not the reason cited in Lesson plan

Technical Reference(s) NOH01RPS00C-05 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: RPS000E007 (As available)

Question Source:			
	Bank #	68852	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History: Last NRC Exam 2002

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X

Comments:

E. Operating Procedures

1. Review the following procedures

   a. HC.OP-SO.SB-0001, RPS Operation

   b. HC.OP-SO.SM-0001, Isolation System Operation

2. For RPS operation, seven separate subsections are used as appropriate.

   a. 5.1 System Operability Observation

   b. 5.2 RPS Scram

   c. 5.3 Resetting RPS Trips

   d. 5.4 Transfer of RPS Bus A (B) Power from RPS MG Set A (B) to RPS Alternate Transformer A (B)

   e. 5.5 Transfer of RPS Bus A (B) Power from RPS Alternate Transformer A (B) to RPS MG set A (B)

   f. 5.6 Bypassing Reactor Mode Switch Scram

   g. 5.7 Resetting the Reactor Protection EPA Breaker Trips

   h. 5.8 Moving Reactor Mode Switch out of Shutdown in OPCON 3 or 4

3. In sections 5.1, 5.4, and 5.5, local operations including stopping or starting the RPS MG sets.

4. For scram, confirming "All Rods In" is the most important action using various indicators. Placing the Reactor Mode Switch in the SHUTDOWN position is a backup trip action.

5. Resetting a Full Scram as soon as reasonable is necessary to reduce CRD internal seal damage from excessive flow, and to minimize vessel thermal stratification if the Recirculation Pumps are out of service.

   [Resetting the scram will also depressurize the scram discharge volume. The RCPB is extended out to the scram discharge volume after a scram and any leakage past the scram discharge volume vent and drain valves could result in a LOCA outside of containment.]

6. Manual reset of a full scram is inhibited for 10 seconds after initiation to ensure control rods drive to FULL IN position.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295001 AK1.02	
	Importance Rating	3.3	

Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Power/flow distribution.

Proposed Question: Common 41

Plant conditions are as follows:

- Initially the reactor is at 96.9% power with both Recirculation loops in operation.
- OPRMs are inoperable.

Following a lightning strike event and various power losses, core flow and power are reduced.

- Core flow is at 40% of rated and steady.
- Reactor power is 40% and slowly rising.
- APRM Recorders are currently reading <1 percent peak-to-peak oscillations.

If flow remains at 40%, which of the following is the lowest power level that the mode switch shall be locked in Shutdown?

A. 44%

B. 47%

C. 50%

D. 56%

Proposed Answer: C.

Explanation (Optional):

C. Correct - Region 1 of the current MELLA OPRM INOP power to flow map has 40% core flow cross the curve at 50.4% power

A. Incorrect - 44% if choosing region 2 looks correct - does not require the action

B. Incorrect - Also in region 2 - does not require the action

D. Incorrect - Region 1 but not the lowest value in region 1

Technical Reference(s) IOZZ-0006, Att 1, (Attach if not previously provided)

OP-AB.RPV-0003, pgs 2 and 3 of 41

Proposed references to be provided to applicants during examination: Both Power-to-Flow maps HC-OP. IO-ZZ-0006 Att. 1

Learning Objective: IOP006E004 (As available)

Question Source: Bank # ID: Q81326
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam _____

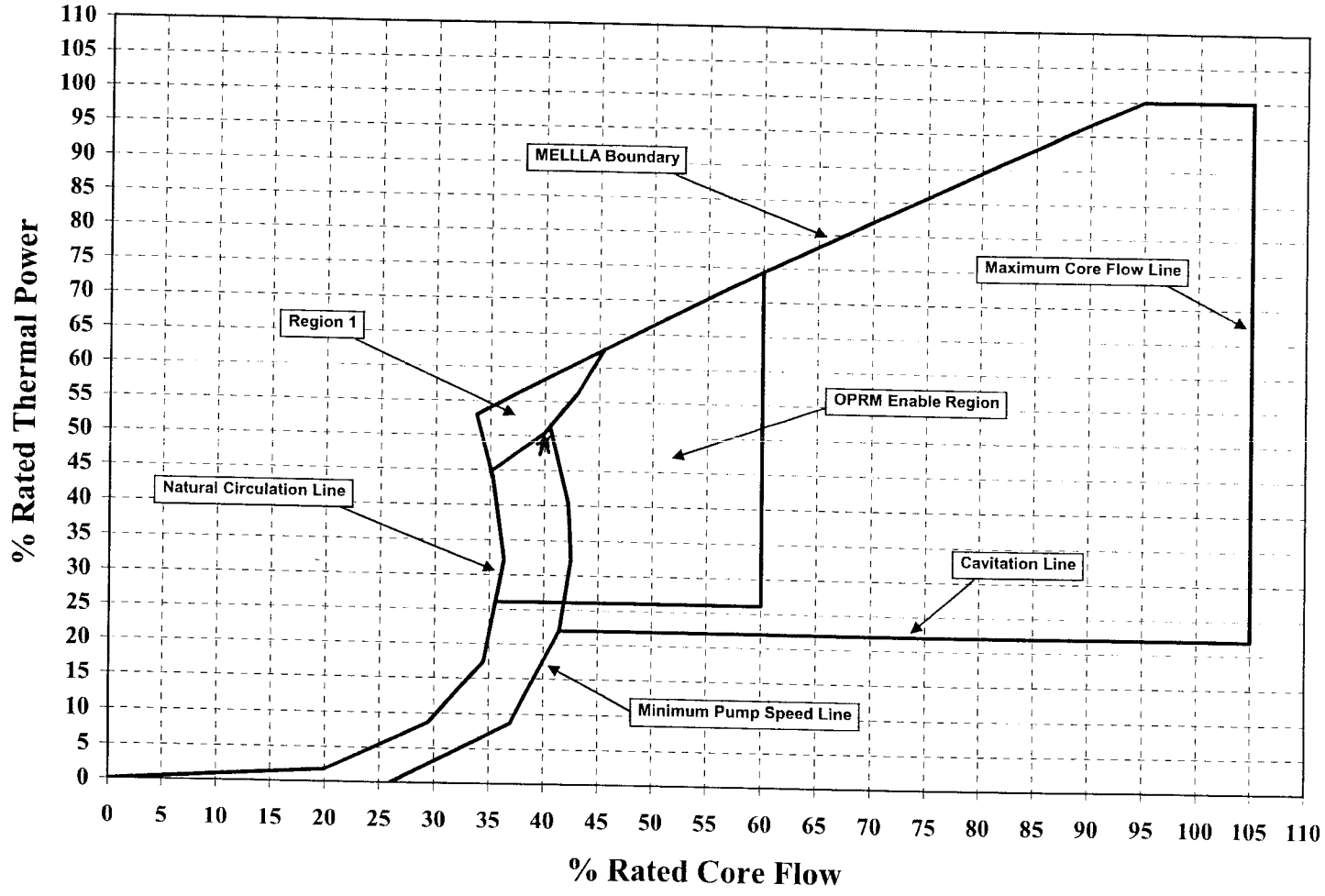
Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

Comments:

A41
NRC

ATTACHMENT 1
POWER CHANGES DURING OPERATION
POWER TO FLOW MAPS
OPRMs OPERABLE
(Page 1 of 2)



#41 - NRC

IMMEDIATE OPERATOR ACTIONS

CONDITION	ACTION
No Recirc. Pumps running AND Reactor is Critical **[CD-354F]** Date/Time: _____	☐ **LOCK** the Mode Switch in SHUTDOWN.
Indications of Power Oscillations (Thermal Hydraulic Instability) Date/Time: _____	☐ **LOCK** the Mode Switch in SHUTDOWN. **[CD354F]**
NOTE 1 OPRM's INOPERABLE AND in REGION 1 of Figure 1 Date/Time: _____	☐ **LOCK** the Mode Switch in SHUTDOWN.

AUTOMATIC ACTIONS

IF	THEN
OPRM TRIP	Half or Full Scram

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	700000 AK2.02	
	Importance Rating	3.1	

Knowledge of the interrelations between GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES and the following: Breakers, relays.

Proposed Question: Common 42

The plant was operating at full power when a grid disturbance results in actuation of the Main Generator Lockout Relay (86GR)?

Which one of following describes the response of the plant?

A. Generator output breakers BS6-5 & BS2-6 TRIP.
Isophase Bus Duct Cooling Fans TRIP.
The Main Turbine AND Alterex Exciter Field Breaker are unaffected.

B. The Main Turbine AND Alterex Exciter Field Breaker TRIP.
Generator output breakers BS6-5 & BS2-6 AND
Isophase Bus Duct Cooling Fans are unaffected.

C. Main Turbine, Alterex Exciter Field Breaker AND
Generator output breakers BS6-5 & BS2-6 TRIP.
Isophase Bus Duct Cooling Fans are unaffected.

D. The Main Turbine, Alterex Exciter Field Breaker, Generator output breakers BS6-5 & BS2-6 AND Isophase Bus Duct Cooling Fans ALL TRIP.

Proposed Answer: C

Explanation (Optional):
The following automatic actions are initiated when the Main Generator regular lockout relay (86GR) is actuated.

Trips and blocks closing of gen output breakers BS6-5 & BS2-6
Trips the Main Turbine
Trips and blocks the closing of the Alterex Exciter Field Breaker

C. Correct – see above

A. Incorrect – The isophase fans are not interlocked to trip on a generator lockout and the turbine will trip and Alterex Exciter Field Breaker will open.

B. Incorrect - The isophase fans are not interlocked to trip on a generator lockout and the generator output breakers BS6-5 & BS2-6 will trip open.

D. Incorrect - The isophase fans are not interlocked to trip on a generator lockout.

Technical Reference(s) OP-SO.MA-0001 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: MNGEN0E011 (As available)

Question Source: Bank # _____
Modified Bank # _____ (Note changes or attach parent)
New X

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 X

Comments:

#42-NRC

3) Turbine (EHC) controls vary the amount of steam delivered to the Main Turbine to control speed when not paralleled or MW load when paralleled to the grid.

b. The Iso-phase Bus Duct Cooling controls (discussed in the component section) are located in the Turbine Building (elevation 102') on Panels 10C116 and 10G107.

c. The Main Transformer controls are in cabinets located next to each Main Transformer.

2. Automatic Actuations

E-0003-1 E-3030-0 Sht 2 E-0121-0 Shts 1,2 (Ref. only)
TABLE 2
Obj. 11a

a. The Main Generator is protected from unsafe conditions by a REGULAR Lockout Relay [(4)86GR] and a BACKUP Lockout Relay [(4)86GB]. Both relays are normally de-energized. The conditions that will cause the lockout relays to energize are listed on Table 2. When either of the Main Generator lockout relays energizes, the following actions occur:

1) The Main Turbine trips, and

2) Main Generator Output Breakers BS2-6 <u>and</u> BS6-5 trip (the trip signal also blocks re-closure of the breakers), and

3) The Exciter Field Breaker trips (the trip signal also blocks re-closure of the breaker), and

4) Breaker Failure Protection for Main Generator Output Breakers BS2-6 <u>and</u> BS6-5 is initiated, and

5) An overhead annunciator on Control Room Panel 10C800 alarms, and

6) A process computer alarm is generated.

b. If the Main Generator REGULAR Lockout Relay is energized as a result of a Generator Differential Overcurrent condition, this also results in a trip of both Stator Coolant pumps (1AP119 and 1BP119).

c. The Main Generator REGULAR Lockout Relay is located in the Upper Relay Room (elevation 163') on panel 1AC654D. It can only be reset (de-energized) from this panel by placing its respective HAND/RESET switch in the RESET position. The same holds true for the Main Generator BACKUP Lockout Relay, which is located in the Upper Relay Room on panel 1AC654E.

TABLE 2

MAIN GENERATOR LOCKOUT SIGNALS

1. Generator Differential Overcurrent
2. Alternator Exciter Differential Overcurrent
3. Generator Negative Phase Sequence
4. Generator Anti-Motoring (Reverse Power) - time delayed 2.5 seconds with a concurrent Main Turbine trip signal

   THE FOLLOWING SIGNALS WILL ENERGIZE THE REGULAR LOCKOUT RELAY ONLY IF ONE OR BOTH MAIN GENERATOR OUTPUT BREAKERS ARE CLOSED

5. Generator Loss of Field - but only if Static Voltage Balance relay (4)68A is NOT energized.
6. Generator Out of Step - but only if Static Voltage Balance relay (4)68A is NOT energized.
7. Maximum Excitation (Field Overvoltage) - time delayed 10 seconds.
8. Generator Field Overexcitation (Volts/Hertz)

BACKUP Lockout Relay [(4)86GB] will energize if ANY of the following conditions exist:

1. Generator Overcurrent
2. Generator Anti-Motoring (Reverse Power) - time delayed 30 seconds

   THE FOLLOWING SIGNALS WILL ENERGIZE THE BACKUP LOCKOUT RELAY ONLY IF ONE OR BOTH MAIN GENERATOR OUTPUT BREAKERS ARE CLOSED.

3. Generator Loss of Field - but only if Voltage Balance relay (4)60 is energized (indicative of an Unbalanced condition) - time delayed 1.0 second.
4. Generator Out of Step - but only if Voltage Balance relay (4) 60 is energized (indicative of an Unbalanced condition)

* The Main Generator Field Ground lockout signal was removed as part of DCP 4 HM-0322

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295030 EK2.02	
	Importance Rating	3.7	

Knowledge of the interrelations between LOW SUPPRESSION POOL WATER LEVEL and the following: RCIC: Plant-Specific.

Proposed Question: Common 43

IAW HC.OP-EO.ZZ-0313, Emergency Makeup to the Suppression Pool via RCIC is accomplished by....

A. running RCIC with the full flow test flowpath open.

B. running RCIC with the minimum flow discharge flowpath open.

C. overriding and opening both RCIC suction MOV's simultaneously.

D. overriding and opening the Test Return and the Suppression Pool suction MOV's.

Proposed Answer: B

Explanation (Optional):

B. CORRECT - Running RCIC with the min flow discharge flowpath open, IAW EO-ZZ-313.

A. Incorrect. This would only recirculation CST water. No procedural guidance
C. Incorrect - There is a check valve in the Suppression pool suction path to prevent gravity draining the CST into the Suppression pool. No procedural guidance
D. Incorrect - There is a check valve in the Suppression pool suction path to prevent gravity draining the CST into the Suppression pool. No procedural guidance

.

Technical Reference(s) HC.OP-EO.ZZ-0313 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: EOP300E004 (As available)

Question Source:	Bank #	56168	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	X	

Comments:

H43
NRC

SUPPRESSION CHAMBER MAKE-UP USING RCIC

1.0 PURPOSE

This procedure provides the guidance to raise Suppression Chamber level by operating the RCIC system with suction from the CST and discharge to the Suppression Chamber through the minimum-flow valve. This action would be required if Suppression Chamber level was below 74.5" and normal means of make-up were unavailable.

2.0 PREREQUISITES

2.1 Water available in the Condensate Storage Tank

2.2 RCIC System is available for operation.

2.3 RCIC system is not being used to assure adequate core cooling.

3.0 PRECAUTIONS AND LIMITATIONS

3.1 This procedure constitutes actions that are only to be taken in response to severely degraded plant conditions. These actions warrant evaluation of current and future plant conditions, prioritization of available plant systems and components, and consideration of the expected consequences to the plant.

3.2 This procedure shall be executed only as directed by HC.OP-EO.ZZ-0102(Q), Primary Containment Control.

3.3 The Nuclear Shift Supervisor shall log the starting and completion times of this procedure.

3.4 All steps in this procedure are to be completed in sequence unless otherwise specified.

3.5 Comply with the Suppression Pool water temperature requirements of Technical Specification 3.6.2 during RCIC operations.

4.0 EQUIPMENT REQUIRED

None

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295004 AK2.01	
	Importance Rating	3.1	

Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF D.C. POWER and the following: Battery charger.

Proposed Question: Common 44

Regarding the battery chargers for the following DC Buses:

- HPCI 250 VDC bus
- 125 VDC bus 10D410

Which one of the following describes how their respective bus power is affected following the loss of their respective charger(s)?

The batteries for the __(1)__ are designed to supply their loads for __(2)__.

	(1)	(2)
A.	250 VDC 125 VDC	2 (two) hours 2 (two) hours
B.	250 VDC 125 VDC	4 (four) hours 4 (four) hours
C.	250 VDC 125 VDC	2 (two) hours 4 (four) hours
D.	250 VDC 125 VDC	4 (four) hours 2 (two) hours

Proposed Answer: B

Explanation (Optional): B. Correct

IAW DC Lesson Plan NOH01DCELEC-01, Page 33. The 250 VDC Class 1E batteries can supply system loads for four (4) hours without the battery chargers in operation. The 125 VDC Class 1E batteries can supply system loads for four (4) hours without the battery chargers in operation. The TS LCO time is 2 hours. TS 3.8.2.1.

A, C, D. Incorrect – All batteries are designed for 4 hours

Technical Reference(s) NOH01DCELEC-01 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: DCELECE015 (As available)

Question Source: Bank # _____
Modified Bank # _____ (Note changes or attach parent)
New X

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 X

Comments:

c. Instrumentation

1) DC ground detection ammeters

a) Distribution panels 1AD318, 1BD318,

b) Panel 10C679 in the Control and Equipment Room.

Obj. 22a

C. Instrumentation and Alarms

1. Computer points

a. 125 VDC Non-1E switchgear panel trouble alarm for each switchgear.

b. 125 VDC Non-1E battery charger trouble alarm

c. 125 VDC Non-1E battery monitor trouble alarm

OP-AR.ZZ-014 (ref only)

2. 125 VDC SYSTEM TROUBLE annunciator panel 10C800, window D3/F2

V. 250 VDC 1E DISTRIBUTION

A. General

Fig. 8,26

Obj. 1d

1. Purpose

The 250 VDC Class 1E Power System supplies 250 VDC power as required to pumps and valves for:

Obj. 6d&e

a. High Pressure Coolant Injection System (HPCI)

b. Reactor Core Isolation Cooling System (RCIC)

2. Design Bases

a. The 250 VDC Class 1E Power System is designed with sufficient capacity, redundancy, independence and testability to ensure the performance of safety functions assuming a single failure.

b. The 250 VDC Class 1E batteries can supply system loads for four (4) hours without the battery chargers in operation.

INSTRUCTIONAL CONTENT:

Obj. 1
Fig 18-AV6208

I. INTRODUCTION/PURPOSE

A. The High Pressure Coolant Injection (HPCI) System provides adequate make-up water during a small break loss of coolant accident (LOCA) to ensure adequate core cooling.

B. The HPCI System may be used for reactor vessel inventory or pressure control whenever the vessel is pressurized and isolated from the main condenser.

II. DESIGN BASES

The HPCI System shall be able to maintain reactor vessel inventory during a loss of coolant accident originating from a 1" or smaller break under the following conditions:

Formatted: Bullets and Numbering

A. Reactor pressure – 165 psia to 1156 psia

B. Developed head - 710 feet to 2950 feet

C. Flow rate - 5600 gpm within 35 seconds of initiation

D. Independent of AC power

III. DESCRIPTION

A. General

M-55-1, M-56-1

1. The HPCI System consists of the following major components:

a. HPCI turbine

b. HPCI pump

c. HPCI booster pump

d. HPCI lubricating/control oil system

e. HPCI turbine governor control system

f. Barometric condenser

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Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295037 AK3.01	
	Importance Rating	3.4	

Knowledge of the reasons for the following responses as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN : Recirc Pump Trip/Runback

Proposed Question: Common 45

A failure to scram has occurred and reactor power is 65%. The main turbine is on line.

The recirculation pumps are required to be runback to minimum speed before tripping the pumps to...

A. maintain the largest margin to the MCPR limit.

B. prevent an RPV high level trip to ensure HPCI/Core Spray injection flowpath.

C. prevent power instabilities due to operating at high power without adequate core flow.

D. prevent additional heat loading of the torus if power remains above the bypass valve capacity.

Proposed Answer: D

Explanation (Optional): EOP-101A RC/Q-8 bases

The most rapid flow rate reduction and, consequently, the most rapid power reduction, is achieved by tripping the recirculation pumps. However, if the recirculation pump trip is initiated from a high power level, the resulting rapid changes in steam flow, RPV pressure, and RPV water level may cause a trip of the main turbine-generator and a trip of RPV injection systems. If the main turbine-generator trips and reactor power exceeds the turbine bypass valve capacity, RPV pressure will increase until one or more SRVs open. Heatup of the suppression pool then begins and RPV level lowering may be required. If RPV injection systems trip, the resultant RPV water level transient may require emergency depressurization of the RPV and operation of less desirable RPV injection sources.
To effect a more controlled reduction in reactor power and thereby avoid main turbine-generator and RPV injection system trips and their associated complications, a recirculation flow runback is performed prior to tripping the recirculation pumps. If an automatic runback has occurred, the operator need only confirm the action.

D. Correct - prevent additional heat loading of the torus if power remains above the bypass valve capacity

A. Incorrect - maintain the largest margin to the MCPR limit. Removing RPV flow will rely

on natural circulation to prevent approaching the MCPR limit during an ATWS, it will certainly not lessen it

B. Incorrect - In an ATWS condition, HPCI injection through Core Spray flowpath is not desired.

C. Incorrect - prevent power instabilities due to operating at high power without adequate core flow. Actions taken will remove all forced circulation, and lower RPV level to lower power, power takes precedent over instabilities.

Technical Reference(s) EOP-101A bases (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: EO101AE006 (As available)

Question Source: Bank # 56604
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam none

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 X

Comments:

considered. However, if power is less than 4%, the basis for BIIT will be met as long as boron injection commences prior to the BIIT temperature associated with 4% power. A conservative value of 140°F has been selected based on supp pool temperature instrument readability.

Since failure-to-scram conditions may present severe plant safety consequences, the requirement to initiate boron injection is independent of any anticipated success of control rod insertion. When attempts to insert control rods satisfactorily achieve reactor shutdown, the requirement for boron injection no longer exists. (Control rod insertion is directed by Step RC/Q-5)

SLC initiation generates a RWCU isolation signal. This isolation prevents RWCU FDs from removing injected boron.

RC/Q-8

VERIFY recirc runback to minimum.

Discussion

An immediate and rapid reactor power reduction may be effected by reducing reactor coolant recirculation flow rate. This action may place plant operations in a high power-to-flow condition where, under certain circumstances, neutronic/thermal-hydraulic instabilities are possible. However, the action to reduce recirculation flow rate remains appropriate because:

- Severe suppression pool heating and potential containment failure result if recirculation flow is not reduced.
- Neutronic/thermal-hydraulic instabilities can be accommodated without fuel failure in most cases.
- RPV water level control actions which are being performed in Contingency #5 concurrently with the actions of this subsection will prevent or mitigate the occurrence of extremely large neutronic/thermal-hydraulic instabilities which may damage the fuel.
- Reactor power control actions to inject soluble boron and insert control rods, if successful, will terminate the failure-to-scram condition, thus preventing prolonged exposure to instabilities.

The most rapid flow rate reduction and, consequently, the most rapid power reduction, is achieved by tripping the recirculation pumps. However, if the recirculation pump trip is initiated from a high power level, the resulting rapid changes in steam flow, RPV pressure, and RPV water level may cause a trip of the main turbine-generator and a trip of RPV injection systems. If the main turbine-generator trips and reactor power exceeds the turbine bypass valve capacity, RPV pressure will increase until one or more SRVs open. Heatup of the suppression pool then begins and RPV level lowering may be required. If RPV injection systems trip, the resultant RPV water level transient may require emergency depressurization of the RPV and operation of less desirable RPV injection sources.

To effect a more controlled reduction in reactor power and thereby avoid main turbine-generator and RPV injection system trips and their associated complications, a recirculation flow runback is performed prior to tripping the recirculation pumps. If an automatic runback has occurred, the operator need only confirm the action.

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Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295026 EK3.04	
	Importance Rating	3.7	

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

Proposed Question: Common 46

EOP 101A steps RC/Q-10 and RC/Q -11 state: Before Suppression Pool temperature reaches 140 degrees F, initiate SLC and verify RWCU isolates.

What is the reason for these EOP steps?

A. It ensures the reactor will be in Cold Shutdown before the Suppression Pool reaches the Heat Capacity Temperature Limit.

B. It ensures the reactor will be in Hot Shutdown before the Suppression Pool reaches the Heat Capacity Temperature Limit.

C. It ensures the reactor will be in Cold Shutdown before the Suppression Pool reaches the Boron Injection Initiation Temperature.

D. It ensures the reactor will be in Hot Shutdown before the Suppression Pool reaches the Boron Injection Initiation Temperature.

Proposed Answer: B.

Explanation (Optional): B. Correct

IAW EOP-101A, step RC/Q-10 bases

A. Incorrect – Hot S/D is the bases
C. Incorrect – The (BIIT) is not the bases. HCTL in Hot S/D is the reason.
D. Incorrect - The (BIIT) is not the bases.

Technical Reference(s) EOP-101A bases page 11 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: EO101AE004 (As available)

Question Source: Bank # 53439
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 X
_____

Comments:

RC/Q-9

Trip the recirc pumps.

Discussion

If reactor power remains above the APRM downscale setpoint, the recirculation pumps are tripped to effect a prompt reduction in power. While tripping the pumps may place the plant in a high power-to-flow condition and thereby contribute to neutronic and thermal-hydraulic instabilities, continued recirculation pump operation may not be desirable or even possible:

- If RPV water level is lowered in accordance with the LP leg, the pumps will trip automatically when the low RPV water level trip setpoint is reached.
- Allowing reactor power to remain high would increase the flow demand on RPV injection systems and the heat load on the primary containment.

Tripping the recirculation pumps may also reduce the boron mixing efficiency if boron injection is required. However, three-dimensional scale model tests have demonstrated that natural circulation provides adequate mixing as long as RPV water level is above the elevation at which a natural circulation flowpath can be established.

If reactor power is below the APRM downscale trip setpoint, tripping the recirculation pumps results in little, if any, reduction in reactor power since power is already near the decay heat level. In this case, forced recirculation flow is continued, if possible, to enhance boron mixing if boron injection is later required.

RC/Q-10 and 11

Before suppression pool temperature reaches 140°F,
INITIATE SLC AND **VERIFY** RWCU isolates

Deleted: 4

Discussion

The Boron Injection Initiation Temperature (BIIT) is the greater of:

- The highest suppression pool temperature at which initiation of boron injection will permit injection of the Hot Shutdown Boron Weight of boron before suppression pool temperature exceeds the Heat Capacity Temperature Limit.
- The suppression pool temperature at which a reactor scram is required by plant Technical Specifications.

The BIIT is a function of reactor power. If boron injection is initiated before suppression pool temperature reaches the BIIT, emergency RPV depressurization may be precluded at lower reactor power levels. At higher reactor power levels, however, the suppression pool heatup rate may become so high that the Hot Shutdown Boron Weight of boron cannot be injected before suppression pool temperature reaches the Heat Capacity Temperature Limit even if boron injection is initiated early in the event. Refer to HC.OP-EO.ZZ-LIMITS-CONV, EOP Limit Curves and Cautions, for a detailed discussion of the BIIT. Reactor power above 4% results in immediate SLC injection. At 4% or less, there is no major benefit to injecting SLC if control rods can be inserted as long as supp pool temperature remains below the Boron Injection Initiation Temperature (BIIT). The BIIT for power levels at or below 4% is approximately 144°F (See Appendix C, WS-1). The value has been conservatively rounded to 140°F. SLC initiation generates a RWCU isolation signal. This isolation prevents boron from plating out on the RWCU piping, and eliminates the RWCU water volume not considered in the SLC injection total RPV volume.

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Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295025 EK3.05	
	Importance Rating	3.6	

Knowledge of the reasons for the following responses as they apply to HIGH REACTOR PRESSURE: RCIC operation: Plant-Specific.

Proposed Question: Common 47

Ten minutes after a scram, an MSIV Isolation occurs and RCIC is in pressure control augmented by SRVs. RCIC speed is then observed to oscillate with the flow controller in AUTO.

Which of the following explains the RCIC speed oscillations?
(assume NO other operator actions)

A. Swings in RPV pressure are occurring due to the methods being used for pressure control. This requires the RCIC speed to change as the controller maintains a constant flow.

B. In pressure control, the RCIC controller attempts to maintain a constant speed, but CANNOT respond fast enough to maintain speed as the RPV pressure changes.

C. Using RCIC for pressure control is inherently less stable than using it for level control due to the lower pressure in the CST compared to the RPV. The greater instability is seen as an increase in oscillations.

D. The comparatively small CST volume results in the RCIC suction and discharge points in the CST being close together, and at high flow the turbulence causes oscillations.

Proposed Answer: A.

Explanation (Optional):

A. Correct – As the SRVs cycle the reactor pressure will change, this changes the head the RCIC pump must operate against to maintain the constant flow the speed must change.

B. Incorrect - In auto the controller maintains flow and in manual it maintains speed. The candidate may reverse the methods of RCIC control.

C. Incorrect - The controller is equally stable in the pressure and level control modes. The candidate may believe that the operation of RCIC in other than its design function of injecting to the core is less stable

D. Incorrect - The CST has a relatively small volume compared to the SP but it does not result in oscillations. The operator may accept that the smaller flow volume results in

suction/discharge interaction

Technical Reference(s) LP NOH04RCIC00-05 RCIC (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: RCIC00E012 (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

Comments:

#47 - NRC

Obj. 2, 23a, 23d

Fig. 1 & 6

k. Turbine Governor Valve (FV-4283)

1) The purpose of this valve is to control steam flow to the RCIC turbine.

2) It is a 3" poppet valve which is hydraulically throttled closed using RCIC turbine lube oil as the hydraulic fluid.

3) With no hydraulic pressure, the valve is maintained open by spring pressure.

4) Bailey Flow Indicating Controller (FIC-R600).

a) MAN(UAL) - when depressed, the operator controls turbine speed via the (up arrow) (down arrow) pushbuttons.

b) AUTO - when depressed, the operator controls RCIC pump flow via the RAISE SETPOINT, LOWER SETPOINT pushbuttons. Actual flow, which is determined by RCIC turbine speed, is adjusted to match demanded flow. See Turbine Control System.

Obj. 2, 3a

Fig. 1, 9, 10

2. RCIC Turbine and Governor Control System

a. RCIC Turbine

1) Purpose - Converts heat energy of reactor vessel steam into mechanical rotational energy of the RCIC turbine shaft in order to provide the motive force to drive the RCIC pump.

2) Characteristics

a) Manufacturer - Terry Steam Turbine Company

b) Variable speed, single wheel, impulse turbine - Selected because of its suitability for rapid start service and the ruggedness of the solid wheel construction.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295031 EK3.05	
	Importance Rating	4.2	

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

Proposed Question: Common 48

Actions for Steam Cooling are being performed in accordance with HC.OP-EO.ZZ-0101, RPV Control, and RPV level has dropped to -200" on Fuel Zone indication. The procedure requires emergency depressurization.

Which of the following is the reason for emergency depressurizing?

A. Maintain peak cladding temperature below 1500 degrees F.

B. Maintain peak cladding temperature below 1800 degrees F.

C. Maintain total oxidation of the cladding less than 0.17 of the total cladding thickness.

D. Maintain the maximum H2 generation less than 0.01 times the hypothetical maximum.

Proposed Answer: B

Explanation (Optional): IAW EOP-101 Bases Discussion

Steam cooling is effected by allowing RPV water level to decrease through boil-off until it drops to the Minimum Zero-Injection RPV Water Level (MZIRWL). During this period the fuel temperatures in the uncovered portion of the core increase, and heat is transferred from the fuel rods to the steam. The MZIRWL is the lowest RPV water level at which the covered portion of the reactor core will generate sufficient steam to preclude any clad temperature in the uncovered portion of the core from exceeding 1800°F.

When RPV water level drops below the MZIRWL, steam cooling may no longer be sufficient to preclude the peak clad temperature from exceeding 1800°F. **Emergency RPV depressurization is then performed in accordance with EOP-202. Unless the RPV is already depressurized, it is expected that the resulting swell will be sufficient to quench the uncovered portion of the fuel and reduce PCT almost to the value that would exist if the core were submerged.** As the swell subsides and steam flow through the open SRVs decreases, however, PCT turns and again rises.

Opening the SRVs before RPV water level reaches the MZIRWL would reduce the time over which the core remains adequately cooled with no injection. Waiting much after RPV water level reaches the MZIRWL could result in significant core damage due to excessive fuel temperatures.

B. Correct - Maintain peak cladding temperature below 1800 degrees F.

A. Incorrect - Maintain peak cladding temperature below 1500 degrees F. - this is the MSCRWL the lowest RPV water level at which the covered portion of the reactor core will generate sufficient steam to preclude any clad temperature in the uncovered portion of the core from exceeding 1500°F

C. Incorrect - Maintain total oxidation of the cladding less than 0.17 of the total cladding Thickness. - This is an ECCS criteria based on < 2200 degrees F PCT

D. Incorrect - Maintain the maximum H2 generation less than 0.01 times the hypothetical maximum. – This is ECCS criteria based on < 2200 degrees F PCT

Technical Reference(s) EOP 101 Bases (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: EO101LE008 (As available)

Question Source: Bank # 56126
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 X

Comments:

EOP STEP(S): SC-2

JUSTIFICATION

The part of the step referring to pressure not stabilized and increasing has been implemented as part of the SC-1 override.

EPG STEP: C3-3

When RPV water level drops to [-208 in. (Minimum Zero-Injection RPV Water Level)], EMERGENCY RPV DEPRESSURIZATION IS REQUIRED; enter [procedure developed from Emergency Procedure Guideline Contingency #2].

PSTG STEP: SC-3

When RPV water level drops to -205 in. (Minimum Zero-Injection RPV Water Level), EMERGENCY RPV DEPRESSURIZATION IS REQUIRED; enter HC.OP-EO.ZZ-0202.

JUSTIFICATION

- 322.5 IN is the Minimum Zero-Injection RPV Water Level as referenced to the bottom of the RPV (Hope Creek Appendix C Calculation, Work Sheet 10).
- RPV water level instrumentation is referenced to 527.5 IN (instrument zero, Technical Specifications, Bases Figure B 3/4 3-8). Therefore, the 322.5 IN value has been converted to an indicated value of -204.8 IN which can be read on the RPV Fuel Zone level instrumentation.
- HC.OP-EO.ZZ-0202, Emergency Depressurization, is the plant specific procedure developed from Contingency #2.

EOP STEP(S): SC-3 & 4

JUSTIFICATION

The value of -204.8 IN has been conservatively rounded to -200 IN in order to provide a value that can be readily determined using Control Room instrumentation.

#48
NRC

PLANT SPECIFIC TECHNICAL GUIDELINE

HC.OP-EO.ZZ-0203(Q)

STEAM COOLING

SC-1

If while executing the following steps:

- Emergency RPV Depressurization is required, enter **OP-EO.ZZ-202**.
- RPV water level cannot be determined, enter **OP-EO.ZZ-206**.
- Any system, injection subsystem, or alternate injection subsystem is lined up for injection with at least one pump running and RPV water level cannot be restored and maintained above **-186.0 in.** (MSCRWL), EMERGENCY RPV DEPRESSURIZATION IS REQUIRED; enter OP-EO.ZZ-202.
- RPV water level is increasing, enter **OP-EO.ZZ-101** at **Step RC/P-2**.
- One or more SRVs are being used to stabilize RPV pressure and the continuous SRV pneumatic supply is or becomes unavailable, EMERGENCY RPV DEPRESSURIZATION IS REQUIRED; enter **OP-EO.ZZ-202**.

SC-2 Stabilize RPV pressure using one or more of the following systems: ❷✚×❼

- SRVs, only when suppression pool water level is above **0 in.**, defeating pneumatic supply isolation interlocks and restoring the pneumatic supply if necessary; open SRVs in a sequence that will uniformly distribute heat in the Suppression Pool.
- HPCI, defeating interlocks if necessary.
- RCIC, defeating interlocks if necessary.

SC-3 When RPV water level drops to **-205 in.** (Minimum Zero-Injection RPV Water Level), EMERGENCY RPV DEPRESSURIZATION IS REQUIRED; enter **OP-EO.ZZ-202**.

MINIMUM ZERO-INJECTION RPV WATER LEVEL

[Reference EPG/SAG Appendix B section 17.25]

The Minimum Zero-Injection RPV Water Level (MZIRWL) is the lowest RPV water level at which the covered portion of the reactor core will generate sufficient steam to prevent any clad temperature in the uncovered part of the core from exceeding 1800°F. This water level is utilized to preclude significant fuel damage and hydrogen generation for as long as possible.

The derivation of the MZIRWL is shown graphically in Figure B-17-19. Curve ① graphs the steam flow required to maintain clad temperature less than 1800°F. Curve ② graphs the actual steam flow generated by the reactor core. At the intersection of Curves ① and ②, the steam flow generated by the partially submerged core equals the steam flow needed for cooling the uncovered part of the core. The corresponding level is the MZIRWL.

The MZIRWL is determined assuming:

1. The reactor has been shutdown from rated power for ten minutes.
2. The reactor axial power shape was the most limiting top-peaked power shape prior to reactor shutdown.
3. No water is injected into the RPV.

Plant-specific data required to calculate the MZIRWL are:

1. Minimum active fuel length fraction which must be covered to maintain peak clad temperature below 1800°F without injection. [70.83 %.]
2. Active fuel length. [150 in.]
3. Water level at the bottom of active fuel. [-311 in.]

The MZIRWL differs from the Minimum Steam Cooling RPV Water Level (MSCRWL) in two respects:

1. The MZIRWL is based upon maintaining peak clad temperature below 1800°F. The MSCRWL is based upon maintaining peak clad temperature below 1500°F.
2. Water in the lower plenum is assumed to be saturated in the MZIRWL calculation. It is assumed to be subcooled in the MSCRWL calculation.

[The Hope Creek MZIRWL is -205 in. rounded to -200 in.]

The MZIRWL is referenced in EPG Step C3-3.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295019 AA1.03	
	Importance Rating	3.0	

Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: Instrument air compressor power supplies.

Proposed Question: Common 49

Given the following conditions:

- A loss of coolant accident has previously occurred
- The LOCA signal has cleared
- Instrument air header pressure is lowering

Which of the following describes requirements to manually start the Emergency Instrument Air Compressor (EIAC) before depressing the START pushbutton?

A. The LOCA signal must be reset then the 1E breaker closed.

B. The LOCA signal must be reset then the Non-1E breaker must be closed.

C. The LOCA signal must be reset then the 1E breaker closed AND Instrument air header pressure must then drop below 85 psig.

D. The LOCA signal must be reset then the Non-1E breaker must be closed AND Instrument air header pressure must then drop below 85 psig.

Proposed Answer: A

Explanation (Optional): A. Correct – IAW SO.KB-0001 Steps 3.3.4 & 3.3.5 - To restart the Emergency Instrument Air Compressor following a LOCA, the feeder breaker on Class 1E Unit Substation 10B450 must be reclosed. The Compressor can then be started from either the Control Room OR Local Panel 10C189. The Emergency Instrument Air Compressor will start anytime the MANUAL pushbutton is pressed.

B. Incorrect - EIAC is powered by a 1E supply.
C. Incorrect - IA header pressure is not a restraint.
D. Incorrect – EIAC is powered by a 1E supply. IA header pressure is not a restraint.

Technical Reference(s) HC.OP-SO.KB-0001 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: INSAIRE015 (As available)

Question Source: Bank # 53430
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 _____

Comments:

#49

3.3.4. The Emergency Instrument Air Compressor will start anytime the MANUAL pushbutton is pressed. ____

- To stop the Emergency Instrument Air Compressor after swapping to manual the STOP push-button must be pressed AND held for 3 seconds. ____

3.3.5. To restart the Emergency Instrument Air Compressor following a LOCA, the feeder breaker on Class 1E Unit Substation 10B450 must be reclosed. The Compressor can then be started from either the Control Room OR Local Panel 10C189. ____

3.3.6. IF RACS cooling water is less than 45 psig,
THEN the Emergency Instrument Air Compressor will NOT start.
IF the compressor is in AUTO
AND has been inhibited from start due to low RACS pressure,
THEN the compressor will AUTO start
AND run normally once RACS pressure has been restored. ____

3.3.7. WHEN the Emerg Inst Air Compressor is placed in-service
THEN the RACS System Demineralizers will isolate. ____

3.3.8. IF the Emergency Air Compressor breaker is racked out,
THEN you will get a one time compressor AUTO-START
WHEN you go from MANUAL to AUTO position (once the breaker is racked in). ____

3.3.9. The Emergency Instrument Air Compressor will run for a minimum of 45 minutes following an AUTO-START signal. (This assumes the compressor never loaded - Step 3.3.2) ____

3.3.10. 1AF104, Standby Instrument Air Dryer, will automatically be in-service with power failure or air pressure dropping below 85 psig. ____

4.0 EQUIPMENT REQUIRED

None

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295038 EA1.03	
	Importance Rating	3.7	

Ability to operate and/or monitor the following as they apply to HIGH OFF-SITE RELEASE RATE: Process liquid radiation monitoring system.

Proposed Question: Common 50

Given the following:

- A discharge of the Equipment Drain Sample Tank is in progress to the River
- The Liquid Radwaste Discharge Isolation Valve (HV-5377A) to the Cooling Tower Blowdown automatically closes

Which one of the following condition(s) would cause this termination?
(Assume NO operator action)

(1) Liquid Radwaste Effluent High radiation setpoint is reached
(2) Cooling Tower Blowdown dilution flow low flow setpoint is reached
(3) Liquid Radwaste Effluent sample flow rate HI setpoint is reached
(4) Cooling Tower Blowdown RMS High radiation setpoint is reached
(5) Liquid Radwaste Effluent High discharge flow setpoint is reached

A. (1) and (3) ONLY

B. (2), (4) and (5) ONLY

C. (2), (3) and (4) ONLY

D. (1), (2) and (5) ONLY

Proposed Answer: D

Explanation (Optional): D. Correct IAW HC.OP-AR.SP-0001 Rev.19 Alarm Point 9RX508 (page 23)

AUTOMATIC ACTION
Isolation of HV-5377A&B due to any one of the following:
• High radiation (HIGH LED on 0SP-RI-4861)
• High Disch Flow (setpoint determined by Liquid Effluent Permit)

- Low Dilution Flow (setpoint determined by Liquid Effluent Permit)
- Low Sample Flow (0HBFIS-4861)
- Monitor Failure

D. Correct.

A. Incorrect. (3) is incorrect. (5) is also correct
B. Incorrect. (4) is incorrect. (1) is also correct
C. Incorrect. (3) is incorrect. (1) is also correct.

Technical Reference(s) HC.OP-AR.SP-0001 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: RWOVERE005 (As available)

Question Source: Bank # 68906
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam 2002

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 X

Comments:

10. Dilution of liquid radwaste discharges shall be accomplished by introducing the effluent into the cooling tower blowdown line. The discharge connection with the circulating water system shall be outside the power block.

OBJ.4

11. Waste discharge from the liquid radwaste system shall be sampled before discharge, shall be monitored during discharge, and shall be automatically terminated when the instantaneous radioactivity concentration would reach 10CFR20 limits for an unrestricted area after dilution. The liquid radwaste release system automatically isolates on the following:

OBJ.5

a. Radwaste effluent high radiation

b. Radwaste effluent high flow

c. Cooling Tower blowdown low flow

d. Loss of operate (Liquid Radwaste Effluent Radiation Monitor Downscale)

e. Loss of sample flow

12. High purity wastes should have a suspended solids concentration <20 ppm.

13. Lower purity wastes should have a suspended solids concentration <500 ppm.

14. Liquid radwaste discharges should not exceed 1% of the 10CFR20 limits on an annual average basis at the point discharge.

15. Class 1E power shall be provided for primary and secondary containment isolation only.

16. Environment qualification shall be provided for primary and secondary containment isolation valves only.

17. Equipment and components of the liquid radwaste system shall be non-seismic Category I with the exception of the containment isolation valves and the associated piping.

18. Foundations and walls of the structure that house the liquid radwaste system shall be designed to the seismic criteria stipulated in Regulatory Guide 1.143. The equipment and associated piping of liquid radwaste system is non-seismic Category I.

19. The liquid radwaste tanks rooms shall be designed to contain the maximum liquid inventory in case the tank ruptures.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295003 AA2.04	
	Importance Rating	3.5	

Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER : System Lineups

Proposed Question: Common 51

The plant was at full power with all systems operable and in their normal alignment.

- 'A' and 'C' RACS pumps were in service

Then a Loss of Offsite Power occurred.

- All four EDG's started and their loads were sequenced on as designed.

Which one of the following describes the response of RACS and Chilled Water system to this transient?

The 'A' and 'C' RACS pumps trip,..

A. the 'A' and 'B' RACS pumps ONLY are automatically started by the LOP sequencers and Chilled Water is aligned to the Drywell coolers.

B. the 'A', 'B' and 'C' RACS pumps are automatically started by the LOP sequencers and Chilled Water is aligned to the Drywell coolers.

C. the 'A' and 'B' RACS pumps ONLY are automatically started by the LOP sequencers and RACS is aligned to the Drywell coolers.

D. the 'A', 'B' and 'C' RACS pumps are automatically started by the LOP sequencers and RACS is aligned to the Drywell coolers.

Proposed Answer: C

Explanation (Optional): C. Correct - IAW HC.OP-SO.ED-0001, Section 3.2.10 - A and B RACS Pump Motors are connected to Class 1E buses AND upon Loss of Power (LOP) without occurrence of a Loss of Coolant Accident (LOCA), A and B RACS pumps restart automatically (in 85 seconds) after the sequencer permissive is received. CHILLED WATER CONTAINMENT CLG SPLY SELECT GB-HV-9530 A1/A3 AND B1/B3 LOOP A and B SPLY/RTN CHW will close AND GB-HV-9530 A2/A4 and B2/B4 LOOP A and B SPLY/RTN RACS will open and if in AUTO,

and not in REMOTE, HV-2537 A and B HX INLET VLVS 1AE217 and 1BE217 INLET will open.
C RACS
Pump is connected to a non-1E bus (10B250) and upon loss of power, is de-energized with no restoration capabilities.

A. Incorrect. RACs is aligned to the DW Coolers
B. Incorrect. C RACs pump will have no power. RACs is aligned to the DW Coolers.
D. Incorrect. C RACs pump will have no power.

Technical Reference(s) HC.OP-SO.ED-0001 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: RACS00E009 (As available)

Question Source: Bank # ____
Modified Bank # 64579 (Note changes or attach parent)
New ____

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

Comments:

H51

3.2.10. A and B RACS Pump Motors are connected to Class 1E buses AND upon Loss of Power (LOP) without occurrence of a Loss of Coolant Accident (LOCA), A and B RACS pumps restart automatically (in 85 seconds) after the sequencer permissive is received. CHILLED WATER CONTAINMENT CLG SPLY SELECT GB-HV-9530 A1/A3 AND B1/B3 LOOP A and B SPLY/RTN CHW will close AND GB-HV-9530 A2/A4 and B2/B4 LOOP A and B SPLY/RTN RACS will open and if in AUTO, and not in REMOTE, HV-2537 A and B HX INLET VLVS 1AE217 and 1BE217 INLET will open. C RACS Pump is connected to a non-1E bus (10B250) and upon loss of power, is de-energized with no restoration capabilities. ____

3.2.11. All pump operations shall be performed IAW OP-HC-108-106-1001, Equipment Operational Control. ____

3.2.12. The below listed valves will isolate the Service Water supply to RACS Heat Exchangers on either a LOCA signal OR RACS room flooded signal:

- EA-HV-2203 SERVICE WATER LOOP A RACS HX HDR SPLY ____
- EA-HV-2204 SERVICE WATER LOOP B RACS HX HDR SPLY ____
- EA-HV-2207 RACS HX COOLING INLET VALVE ____
- EA-HV-2346 RACS HX COOLING OUTLET VALVE ____

3.2.13. Operations of RACS in restricted operation, without Service Water System, shall be limited to a maximum RACS System Temperature of 95 °F (as indicated by TE-2563). ____

3.2.14. IF RACS must be removed from service OR operated in the restricted mode, **INFORM** Chemistry to isolate the Crack Monitoring System. ____

3.2.15. **MAINTAIN** RACS Pump(s) suction temperature between 45 °F and 95 °F for normal operation. It may be necessary to provide artificial heat load and/or isolate SSWS flow to one RACS heat exchanger and throttle SSWS flow to < 4000 gpm in the remaining RACS Heat Exchanger to maintain temperature > 45°F. Limit flow of RACS through any one (1A-E-217 or 1B-E-217) Heat Exchanger to 3384 gpm, maximum. For normal 2 pump operation this means 2 Heat Exchangers are required. **[CD-929G]** ____

3.2.16. RACS allowable operating range is 45°F to 95°F, RACS UFSAR temperature range is 40°F to 95°F. ____

#51

The plant was at 100% power with the 'B' and 'C' RACS pumps in service when a Loss of Offsite Power occurred.

All four EDG's started and loaded properly.

What is the expected response of the RACS pumps to this transient?

A. The 'B' and 'C' RACS pumps trip. The 'A' and 'B' RACS pumps are automatically started by the LOP sequencers.

B. The 'B' and 'C' RACS pumps trip. The 'A' & 'B' RACS pumps must be manually restarted.

C. The 'B' RACS pumps trips and is automatically restarted by the LOP sequencer. The 'C' RACS pump remains running.

D. The 'C' RACS pump trips. The 'B' RACS pump remains running. The 'A' RACS pump is automatically started by the LOP sequencer.

Answer: A

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295005 AA2.03	
	Importance Rating	3.1	

Ability to determine and/or interpret the following as they apply to MAIN TURBINE GENERATOR TRIP : Turbine valve position.

Proposed Question: Common 52

Given the following conditions:

- Reactor power is 50%
- The plant is operating normally with Main Turbine First Stage Pressure at 243 psig.

Then,

- A main generator load reject has just occurred resulting in a power/load unbalance trip.

Which one of the following is the immediate response of the Turbine Control Valves (TCVs), Intercept Valves (IVs) and the Reactor Protection System (RPS)?

A. The TCVs and IVs Fast Close.
RPS will trip.

B. The TCVs and IVs Fast Close.
RPS will NOT trip.

C. The TCVs and IVs Throttle Close.
RPS will trip.

D. The TCVs and IVs Throttle Close.
RPS will NOT trip.

Proposed Answer: A

Explanation (Optional): A. Correct

IAW EHC Lesson plan NOH01EHC LOG-04, Page 18, If a power to load unbalance occurs, the control valve and intercept valve fast acting solenoids are actuated.
IAW Turbine Lesson plan NOH01NMTURB-04, page 66, RPS is automatically bypassed at <24% power which is equal to approximately 104.2 psig first stage turbine pressure.

B. Incorrect – RPS will trip
C. Incorrect – Valves will fast close

D. Incorrect – valves will fast close. RPS will trip

Technical Reference(s) NOH01EHC LOG-04 (Attach if not previously provided)
NOH01NMTURB-04

Proposed references to be provided to applicants during examination: none

Learning Objective: EHCLOGE009 (As available)

Question Source: Bank # ____
Modified Bank # NRC 2005 (Note changes or attach parent)
New ____

Question History: Last NRC Exam 2005

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

Comments:

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This annunciator will be received if the hydraulic fluid pressure to any of the turbine CVs is <530 psig. It is an expected alarm on any main turbine trip event and will also be received on a main generator load rejection and/or on a loss of EHC Control Oil pressure.

6. MAIN STOP VALVE CLOSURE (Window C5/B2)

   This annunciator will be received if any MSV is <95% open. It is an expected alarm on any main turbine trip event and will also be received on a loss of EHC Control Oil pressure.

7. TCV FAST CLOSE AND MSV TRIP BYP (Window C5/C2)

   The TCV fast closure and MSV closure inputs to the Reactor Protection System are automatically bypassed when reactor power is < 30% (as sensed by main turbine first stage pressure). This annunciator will be received if main turbine first stage pressure is <135.7 psig.

C. Control Functions and Interlocks

1. Quill Shaft Failure

Obj. 4a,14

a. Failure of the quill shaft with the main turbine in operation could result in a false indication of a stationary rotor from the low speed switch. Quill shaft failure is sensed by the presence of both of the following signals simultaneously:

   1) Turbine speed greater than 100 RPM as sensed by the magnetic speed pickups located on the driving side of the quill shaft. (Turbine Side)

   AND

   2) Turbine stopped signal as sensed by the low speed switch located on the driven side of the quill shaft. (Pump shaft)

b. The presence of both of these signals together indicates that the main turbine rotor is still rolling at high speed but the pump shaft has stopped. This signifies the failure of the quill shaft, and will cause the TURBINE GENERATOR TROUBLE annunciator to be received.

c. If main turbine speed was initially greater than 1350 RPM, the main turbine will trip on low main shaft oil pump discharge pressure (105 psig). This trip is bypassed if main turbine speed was initially less than 1350 rpm.

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#52

Question # 20 **Hope Creek RO Exam - Nov 2005**

RO *Tier #* 1 *Group #* 1 *HC Obj:*

SRO *Importance* 3.3

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

NOTE: Loss of SWC signals are:

- ***Temp 79°C = Hi stator winding cooling outlet temp***
- ***Press 64 psig = Lo stator winding cooling inlet press***
- ***Flow setpoint – 690 gpm.***
- ***These are a 2/3 logic.***

If stator cooling for the generator is lost, the load set is reduced at 26.8%/min. Turbine load will be reduced only when the load set signal is less than the pressure setpoint signal. The load set will continue to be reduced until either stator coolant system status returns to normal or generator load is reduced to less than the stator amperage of ≈ 26.8% of rated turbine load(7055 amps). Below this value the generator has the capability to operate safely without stator cooling. Therefore, further reduction of generator load is not required. The runback progress is checked after 2 minutes. If generator loading is not below ≈ 95% of rated turbine load (24683 amps) at the end of this period, protective circuitry indicates a failure of the runback circuitry and initiates a turbine trip. After 3.5 minutes, if generator loading has not been reduced to the minimum limit (26.8%) of rated turbine load (6814 amps), the circuitry will initiate a turbine trip. Restoration of stator cooling during this period will stop the turbine runback.

Currently, following RF12, the 100% turbine load will be when the generator is carrying 26314 amps.

Obj. 9a;b;c

f) If a power-to-load unbalance (PLU) (also called a load reject) occurs:

- The load signal is immediately set to minimum.
- The control valve and intercept valve fast acting solenoids are actuated
- A direct turbine trip is generated.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295037 EA2.02	
	Importance Rating	4.1	

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

Proposed Question: Common 53

While operating at 100% Reactor Power, an MSIV (NSSSS) isolation occurs and the reactor fails to scram; all rods remain at their pre-trip conditions.

The operators are in the process of deliberately lowering RPV water level.

The current plant conditions are:

- RX Power 4.5%
- RPV Pressure 900 psig
- RPV Level -120 inches
- Suppression Pool Level 79 inches and rising
- Suppression Pool Temp 175 F and rising
- Drywell Pressure 4.5 psig
- SLC Injecting with 3000 gallons remaining in SLC Tank
- Control Rods are being inserted

What action(s) are required to be performed IAW EOPs?

A. Continue to lower RPV level.

B. Open SRVs to Emergency Depressurize.

C. Restore and maintain RPV water level between +12.5 inches and +54 inches.

D. Open SRVs to depressurize.

Proposed Answer: A.

Explanation (Optional):

A. Correct - Continue to lower RPV level because power is above 4.5 %

B. Incorrect - HCTL action required and RPV water level do not meet conditions for ATWS

Emergency de-pressurization.

C. Incorrect - Can not restore RPV water level until the Reactor is shutdown under all conditions without boron, exit EOP-101A and enter EOP-101.

D. Incorrect - Suppression pool temperature can be maintained below HCTL action required area. SRVs not required

Technical Reference(s) EOP-101A (Attach if not previously provided)

Proposed references to be provided to applicants during examination: EOP-101A - no entry conditions

Learning Objective: EO101AE008 (As available)

Question Source: Bank # _____
Modified Bank # ID: Q56142 (Note changes or attach parent)
New _____

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295021 G2.1.28	
	Importance Rating	4.1	

Conduct of Operations: Knowledge of the purpose and function of major system components and controls. (Loss of Shutdown Cooling)

Proposed Question: Common 54

Plant Conditions are as follows:

- Alternate Shutdown Cooling is being implemented by using the C to A RHR Loop Cross-Tie IAW HC.OP-AB.RPV-0009.
- RHR Loop C is the only means of removing heat from the RPV.

If the operator opens HV-F007C, C RHR PMP MIN FL MOV during this operation, how will the plant initially respond?

A. RHR Pump C will lose NPSH.

B. The RPV will drain to the Suppression Pool.

C. Flow through the A RHR Heat Exchanger will rise.

D. SACS outlet temperature from A RHR Heat Exchanger will rise.

Proposed Answer: B.

Explanation (Optional):

B. Correct - Opening HV-F007 C will establish a drain path from the B Recirculation Pump Loop to the Torus via C RHR Pump Suction and HV-F007

A. Incorrect - C RHR Pump would eventually lose NPSH. The stem stipulates the selection of the first consequence

C. Incorrect - The flow which existed initially in the A RHR Heat Exchanger will lower due to a drain path being opened to the Torus.

D. Incorrect - The loss of RHR flow to the A RHR Heat Exchanger will lower the heat burden on SACS and hence the SACS outlet temperature will not rise.

Technical Reference(s) HC.OP-AB.RPV-0009, Rev.5 Caution 1.6 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: ABRPV9E004 (As available)

Question Source: Bank # ID: Q61858
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X
_____

Comments:

ATTACHMENT 2
ALTERNATE DECAY HEAT REMOVAL USING C TO A CROSS-TIE
(Page 2 of 8)

CAUTION 1.6

Manual or automatic opening of HV-F007 A(C) RHR PMP A(C) MIN FL MOV will drain the Reactor Vessel to the Suppression Pool.

1.6 **ENSURE** the following valves are CLOSED: _____

- HV-F007A RHR PMP A MIN FL VLV. _____
- HV-F007C RHR PMP C MIN FL VLV. _____

1.7 **VERIFY** the following LPCI Injection Valves are CLOSED: _____

- HV-F017A RHR LOOP A LPCI INJ MOV _____
- HV-F017C RHR LOOP C LPCI INJ MOV _____

1.8 **CLOSE** the following valves: _____

- HV-F010A RHR LOOP C TEST RET MOV _____
- HV-F024A RHR LOOP A TEST RET MOV _____
- HV-F021A RHR LOOP A SPRAY ISLN MOV _____
- HV-F027A RHR LOOP A SUPP CHAMBER SPRAY HDR ISLN MOV _____

1.9 Fully **OPEN** 1BC-V133 RHR Pmp C Suct Frm Recir Loop B (Rm 4227E). _____

1.10 **ENSURE** F077 RECIRC LOOP B TO RHR SUP MAN VLV is open. _____

1.11 IF the Shutdown Cooling suction line was isolated,
THEN **PERFORM** a fill and vent IAW HC.OP-SO.BC-0002(Q),
Decay Heat Removal Operation. _____

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295024 G2.4.1	
	Importance Rating	4.6	

Emergency Procedures / Plan: Knowledge of EOP entry conditions and immediate action steps. (High DW pressure)

Proposed Question: Common 55

A small break LOCA occurs and the following conditions exist:

- Drywell pressure is 2.5 psig
- Drywell temperature is 150°F
- Reactor Water Level is +15 inches
- Suppression pool temperature is 93 degrees F
- Suppression Pool pressure is 1.6 psig
- Suppression Pool level is 77 inches

Which of the following correctly describes the sections of Emergency Operating Procedures which have been entered and the initial step(s) required?

A. ALL sections of OP-EO.ZZ-101, Reactor/Pressure Vessel (RPV) Control, and ALL sections of OP-EO.ZZ-102, Primary Containment Control.
Lock the Mode Switch in Shutdown.

B. ALL sections of OP-EO.ZZ-101, Reactor/Pressure Vessel (RPV) Control, and ALL sections of OP-EO.ZZ-102, Primary Containment Control.
Lock the Mode Switch in Shutdown and place Drywell Spray in service.

C. Drywell Pressure (DW/P) and Drywell Temperature (DW/T) sections of OP-EO.ZZ-102, Primary Containment Control only, and ALL sections of OP-EO.ZZ-101, Reactor/Pressure Vessel (RPV) Control.
Lock the Mode Switch in Shutdown.

D. Drywell Pressure (DW/P) and Drywell Temperature (DW/T) sections of OP-EO.ZZ-102, Primary Containment Control only, and ALL sections of OP-EO.ZZ-101, Reactor/Pressure Vessel (RPV) Control.
Lock the Mode Switch in Shutdown and place Drywell Spray in service.

Proposed Answer: A

Explanation (Optional): A. Correct – IAW EOP entry conditions and initial steps. Per EOP usage guidance all legs of each EOP are performed concurrently. Conditions are not met for Drywell spray (see attached EOPs)

A. Correct.

B. Incorrect. Drywell spray conditions not met
C. Incorrect. All legs of both EOPs 101 and 102 must be entered
D. Incorrect. All legs of both EOPs 101 and 102 must be entered. Drywell spray conditions not met

Technical Reference(s) EOP 101 & 102 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: DSIL curve

Learning Objective: EO101LE003 (As available)

Question Source: Bank # 56092
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 X

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295016 G2.4.46	
	Importance Rating	4.2	

Emergency Procedures / Plan: Ability to verify that the alarms are consistent with the plant conditions.

Proposed Question: Common 56

Plant conditions are as follows:

- A fire causes an MSIV closure resulting in a scram
- HC.OP-AB.HVAC-0002, Control Room Environment is complete
- Control is being established at the Remote Shutdown Panel (RSP) IAW HC.OP-IO.ZZ-0008, Shutdown from Outside the Control Room
- All Transfer Switches are in the Emergency position
- While placing RCIC in service at the RSP the following indication is received:
  - Turbine Tripped and Low Bearing Oil Pressure alarm indicating lights illuminated

Which one of the following would cause this response?

A. RCIC System trip on high RPV water level.

B. Trip of the RCIC Turbine Mechanical Overspeed device.

C. RCIC System Steam Line break causing an automatic system Isolation.

D. Consequences of the fire because there are NO automatic actions associated with the RCIC system with control from the RSP.

Proposed Answer: B

Explanation (Optional):

IAW HC.OP-IO.ZZ-0008, Note 3.1.8.D

B. CORRECT - Trip of the RCIC Turbine Mechanical Overspeed device. With control at the RSP, all automatic trips and interlocks are disabled. One exception is the RCIC Overspeed Trip. Since it is a mechanical device, it will perform its function even with control at the RSP.

A. INCORRECT - RCIC System trip on high RPV water level. With control at the RSP, all automatic trips and interlocks are disabled.

C. INCORRECT - RCIC System Steam Line break causing an automatic system Isolation. With control at the RSP, all automatic trips and interlocks are disabled.

D. INCORRECT - Consequences of the fire, there are NO automatic actions associated with the RCIC system with control from the RSP. One exception is the RCIC Overspeed Trip. Since it is a mechanical device, it will perform its function even with control at the RSP.

Technical Reference(s) HC.OP-IO.ZZ-0008 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: IOP008E004 (As available)

Question Source: Bank # 62224
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

Comments:

#56

10) RCIC steam line inboard isolation valve bypass valve (HV-F076)

b. Indication for the following valves:

1) RCIC pump discharge valve (HV-F012)

2) Test return valve to condensate storage tank (HV-F022)

3) RCIC turbine exhaust to suppression pool valve (HV-F059)

4) RCIC condenser vacuum pump discharge valve (HV-F060)

5) RCIC turbine exhaust outboard vacuum breaker isolation valve (HV-F062)

6) RCIC turbine exhaust inboard vacuum breaker isolation valve (HV-F084)

7) RCIC condensate pot drain to main condenser valve (HV-F025)

8) RCIC vacuum tank condensate pump discharge to clean radwaste valve (HV-F004)

c. Control/indication for the following pumps:

1) RCIC vacuum tank condensate pump OP220

2) RCIC gland seal condenser vacuum pump OP219

d. Indication for ECCS (RCIC) jockey pump BP228

e. Indication for RCIC turbine speed (SI-4280-2)

f. Control/indication for RCIC system injection flow (FIC-4158)

g. Indication for RCIC turbine trip (ZA-4275)

h. Indication for the following alarm conditions:

1) RCIC turbine bearing oil pressure low (PAL-4276)

2) RCIC high pressure turbine bearing temperature high (TAH-4277)

3) RCIC low pressure turbine bearing temperature high (TAH-4278)

4) Condensate storage tank level low-low (LALL-N061)

2. Residual Heat Removal (RHR) System

a. Control/indication for the following valves:

#56

B. Major Subsystems/Components

Obj. 3c

1. Non-1E Emergency Transfer Switch

The NON 1E TRANSFER SWITCH, when positioned to EMER will initiate closure of the B reactor recirculation pump discharge valve, HV-F031B. This action prevents "short cycling" of shutdown cooling flow when established.

HV-F031B going closed provides a trip signal and a runback signal to the "B" recirc pump; thus, if the pump is running when the Emergency Transfer Switch is taken to emergency, the pump will trip.

Obj. 3c

2. The Channel "A" 1E Transfer Switch being placed in emergency causes the following:

M-51-1
SHS 1&2

a. RHR valves HV-F021A (Inboard Cont Spray), HV-F006A (Pump A suction from Recirc) auto-close.

b. The following actions will occur but will not be observed at the RSP:

1) RHR valves HV-F021A, HV-F006A become inoperative from the control room.

2) RHR valves HV-F009 (SDC Inboard Isolation) and HV-F049 (B Loop discharge to RW) become INOP from the Main Control Room.

M-11-1 SHT 2

3) SACS valves HV-2317A and HV-7922A (SACS FPC X-Conn) auto-close signals are disabled and valve control becomes INOP from Main Control Room.

4) SACS valves HV-2314A and HV-7921A (SACS FPC Supply/Return)become INOP from Main Control Room.

5) The OVERLOAD/POWER FAIL MONITOR (OPF) is disabled for all valves mentioned in steps 1, 2, 3 & 4.

6) Fuel Pool Cooling Pump AP211 becomes INOP from Main Control Room.

#56

7) The RSP/RSS TAKEOVER and the BOP SAFETY SYS OUT OF SVCE alarm annunciates.

Obj. 3c
M-51-1

3. The Channel "B" 1E Transfer Switch being placed in emergency causes the following:

   a. The following will be observed at the RSP:

M-49-1
M-50-1

1) RHR valves HV-F016B (DW Spray), HV-F017B(LPCI Injection) , HV-F021B(DW Spray), <u>and</u> HV-F027B (SP Spray)auto-close.
2) RCIC valves HV-F012(Pump Discharge), HV-F059(Exhaust Isolation), HV-F060(Vacuum Pump discharge), HV-F062(Exhaust Vacuum breaker isolation) auto-open.
3) RCIC valve HV-F022 (Min Flow valve) auto-closes.
4) The RCIC Jockey Pump BP-228 auto-starts.
5) RCIC turbine alarms are enabled at the RSP (may not be on).

   b. The following actions will occur but will not be observed at the RSP:

1) RHR valves HV-F016B, HV-F017B, HV-F021B, HV-F027, become inoperative.
2) RHR valves HV-F003B, HV-F004B, HV-F006B, HV-F007B, HV-F024B, HV-F047B, and HV-F048B become INOP from the Main Control Room.
3) RCIC valves HV-F012, HV-F059, HV-F060, HV-F062 become INOP from Main Control Room.
4) The RCIC Vacuum Pump OP-219 <u>and</u> RCIC Condenser Pump OP-220 auto-start signals are disabled.
5) RCIC turbine alarms are enabled at the RSP (may not be on).
6) RCIC valves HV-F008, HV-F010, HV-F013, HV-F019, HV-F022, HV-F031, <u>and</u> HV-4282 become INOP from the Main Control Room.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295028 EK2.01	
	Importance Rating	3.7	

Knowledge of the interrelations between HIGH DRYWELL TEMPERATURE and the following: Drywell spray: Mark-I&II.

Proposed Question: Common 57

Which one of the following describes the consequences of spraying the drywell when drywell temperature is ABOVE the drywell temperature limit of the Drywell Spray Initiation Limit Curve?

A. External pressure limits on the Secondary Containment would be exceeded.

B. Automatic Depressurization System instruments would NO longer be qualified for these conditions.

C. The relief capacity of the Suppression Chamber to Drywell vacuum breakers would be exceeded.

D. RPV water level instrumentation would become inaccurate due to rapidly lowering drywell temperatures.

Proposed Answer: C.

Explanation (Optional):

C. Correct - If unrestricted, the evaporative cooling affect of Drywell spray could result in an immediate, rapid and large reduction in Drywell pressure at a rate much faster than can be compensated for by the Primary Containment Vacuum Relief System and thus result in a negative Drywell-to-Suppression Chamber differential pressure large enough to cause a loss of Primary Containment integrity.

A. Incorrect – The limit of concern is on the Primary Containment.

B. Incorrect - There is no relationship between the DSIL Curve and ADS. The DSIL Curve permits Spray for a wide range of pressures with Drywell temperature above 340°F.

D. Incorrect - Inaccurate level indication may occur when drywell temperature exceeds the saturation temperature for the existing RPV pressure, since drywell sprays lower drywell temperature they will have the opposite affect.

Technical Reference(s) Bases for DW Spray Curve-EOP-102 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: EO102PE006 (As available)

Question Source: Bank # _____
Modified Bank # ID: Q53337 (Note changes or attach parent)
New _____

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 X

Comments:

initiated, the resulting pressure reduction opens the wetwell-to-drywell vacuum breakers, drawing noncondensibles from the suppression chamber back into the drywell.

As an added conservatism, drywell sprays are initiated when the drywell noncondensible content reaches 5%. This condition defines the Suppression Chamber Spray Initiation Pressure (SCSIP). As the drywell atmosphere is purged to the suppression chamber and replaced by steam, suppression chamber pressure increases. The SCSIP is the lowest suppression chamber pressure which can occur when 95% of the noncondensibles in the drywell have been transferred to the suppression chamber. Refer to HC.OP-EO.ZZ-LIMITS-CONV, EOP Limit Curves and Cautions for detailed discussion of the SCSIP.

The restriction regarding suppression pool water level applies to the internal suppression chamber-to-drywell vacuum breakers. These vacuum breakers will not function as designed if any portion of the valve is covered with water. The specified suppression pool water level assures that no portion of the drywell side of the valve is submerged for any drywell below wetwell differential pressure less than or equal to the valve opening differential pressure. Spray operation with vacuum breakers inoperable (i.e., with no drywell vacuum relief capability) may cause the containment differential pressure capability to be exceeded and therefore is not permitted.

Spray operation effects a drywell pressure and temperature reduction through the effects of evaporative cooling. In evaporative cooling the water spray undergoes a change of state, liquid to vapor.

Evaporative cooling refers to spray droplet heat and mass transfer which occurs when water is sprayed into a superheated atmosphere. The water in each droplet is assumed to instantaneously heat and flash to steam until the surrounding atmosphere saturates, absorbing heat energy from the atmosphere. For bounding calculations with typical drywell spray flowrates, this cooling process results in an immediate, rapid, and large reduction in drywell pressure at a rate much faster than can be compensated for by the primary containment vacuum relief system. Unrestricted operation of drywell sprays could thus result in a negative drywell-to-suppression chamber differential pressure large enough to cause a loss of primary containment integrity.

Considering the pressure drop concerns described above, the Drywell Spray Initiation Limit (DWSIL) is the highest drywell temperature at which initiation of drywell sprays will not result in an evaporative cooling pressure drop to below either:

- The drywell-below-wetwell differential pressure capability, or
- The high drywell pressure scram setpoint.

The DWSIL is a function of drywell pressure. It is utilized to preclude containment failure or de-inertion following initiation of drywell sprays.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	600000 AA1.06	
	Importance Rating	3.0	

Ability to operate and / or monitor the following as they apply to PLANT FIRE ON SITE: Fire Alarm

Proposed Question: Common 58

Given the following conditions:

- The Diesel Generator Room Carbon Dioxide Fire protection system is aligned for automatic operation
- A valid EDG room high temperature alarm has just occurred

Which of the following describes how the Diesel Generator Room Carbon Dioxide Fire protection system responds IAW HC.OP-SO.KC-0002 "CO2 Fire Protection System Operation"?

A. A discharge alarm occurs and CO2 with a wintergreen scent is discharged into the room immediately.

B. A pre-discharge alarm is activated. After a time delay, CO2 with a wintergreen scent is discharged into the room.

C. A pre-discharge alarm is activated. NO CO2 is discharged into the room.

D. A pre-discharge alarm is activated and a wintergreen scent is discharged into the room. After a time delay, CO2 is discharged into the room.

Proposed Answer: B

Explanation (Optional): IAW LP NOH01FIRPRO-03

B. CORRECT: A pre-discharge alarm is activated. After a time delay CO2 with a wintergreen scent is discharged into the room.

A. INCORRECT: A discharge alarm occurs, CO2 with a wintergreen scent is discharged into the room immediately. No there is a warning time delay.

C. INCORRECT: A pre-discharge alarm is activated. No CO2 is discharged into the room. EDG room actuates on Hi Temp

D. INCORRECT: A pre-discharge alarm is activated and a wintergreen scent is discharged into the room. After a time delay, CO2 is discharged into the room. Wintergreen comes in with the CO@, not before

Technical Reference(s) LP NOH01FIRPRO-03 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: FIRPROE015 (As available)

Question Source: Bank # 53407
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam 1998

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 X

Comments:

H58
NRC

applied to the port of the valve piston chamber forcing the valve disc downward, opening the valve. When flow is to be terminated, the 3-way pilot valve repositions to allow a flow path to release the CO_2 pressure from the piston chamber so the valve disc repositions to close the valve.

- A major difference between the master valve and selector valve is the 3-way pilot valve. For the mater valve, when the solenoid that controls the 3-way pilot valve is energized, the flow path created maintains pressure beneath the valve disc so the master valve remains closed. De-energizing the pilot solenoid valve repositions the 3-way valve so that CO_2 pressure is applied to open the valve
- For the selector valves, a normally de-energized solenoid valve provides the necessary flow path to keep the selector valve closed. Energizing the pilot solenoid valve repositions the 3-way valve to apply pressure to open the selector valve.

Obj. 8d

- In the event of a loss of power to the system, the master valve would fail open, allowing the system discharge piping to become pressurized with CO_2 up to the selector valves. However, the selector valves fail closed so that CO_2 is not inadvertently discharged. This configuration also supports fire brigade response to a single location, the location of the selector valve, in the event that CO_2 is required to be discharged during a loss of power.

d) System Operation

- Each of the CO_2 systems is provided with a Chemetron control panel (white). When a CO_2 system actuates, the

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295010 AK1.01	
	Importance Rating	3.0	

Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL PRESSURE: Downcomer submergence: Mark-I&II.

Proposed Question: Common 59

Which of the following events would result in compromising the pressure suppression function of primary containment?

(1) Uncovering the SRV T-Quenchers
(2) Downcomer openings becoming uncovered
(3) Torus to Drywell Vacuum Breakers failing closed
(4) Reactor Building to Torus Vacuum Breakers failing open

A. (1) and (2) ONLY

B. (2) and (3) ONLY

C. (1) and (4) ONLY

D. (3) and (4) ONLY

Proposed Answer: A.

Explanation (Optional): IAW EOP102 bases SP/L-7

A. Correct – (1) and (2) would result in pressurizng directly the containment atmosphere therby potentially compromising high pressure limit in the containment

B. Incorrect – SRV t-quenchers correct
C. Incorrect – downcomer openings correct, Rx bldg to torus vac bkrs failing open would relieve torus pressure.
D. Incorrect – would not compromise Pressure Supression function, suppression pool would still be available.

Technical Reference(s) EOP 102 bases (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: EOP102E009 (As available)

Question Source: Bank # _____
Modified Bank # _____ (Note changes or attach parent)
New X

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 X
_____

Comments:

SP/L-6

RUNBACK Recirc AND **INITIATE** a manual scram

Enter and Execute Concurrently EOP-101

Discussion

Entering RPV Control at Step RC-1 assures that, if possible, the reactor is scrammed and shutdown is assured by control rod insertion before RPV depressurization is initiated. Entry into RPV Control must be explicitly stated because conditions requiring entry into Primary Containment Control do not necessarily require entry into RPV Control . Therefore, a scram may not yet have been initiated. Directing that RPV Control be entered, rather than explicitly stating here "Initiate a reactor scram," coordinates actions currently being executed if RPV Control has already been entered. In addition, entry to the RPV Control guideline must be made because it is through this guideline that the transfer to "Emergency RPV Depressurization," is effected.

SP/L-7

Can supp pool level be maintained above 38.5 in.

YES ⇒ Return to SP/L-1

NO ⇒ Go to SP/L-8

Discussion

When suppression pool water level decreases to the elevation of the downcomer openings, any further drop in water level could result in direct exposure of the drywell atmosphere to the suppression chamber airspace thus compromising the pressure suppression function of the primary containment. Suppression pool water level should therefore be maintained above this elevation.

SP/L-8

EMERGENCY RPV DEPRESSURIZATION IS REQUIRED

Enter and Execute Concurrently EOP-202

Discussion

The RPV is not permitted to remain at pressure if suppression of steam discharged from the RPV into the drywell cannot be assured. When the downcomer vent openings are not adequately submerged, any steam discharged from the RPV into the drywell may not condense

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295022 AK2.03	
	Importance Rating	3.4	

Knowledge of the interrelations between LOSS OF CRD PUMPS the following: Accumulator pressure

Proposed Question: Common 60

Given the following:

- The plant is at 37% power
- Both CRD pumps are tripped on low suction pressure
- The Reactor Building Operator is swapping CRD suction filters
- CRD ACCUM TROUBLE Overhead Annuciator C6-D4 is clear

For the two minutes following the CRD pump trip, what will be the response of HCU Accumulator Gas pressure?
(Assume NO other operator actions)

HCU Accumulator Gas pressure ...

A. stays the same because reactor pressure holds the charging water check valve closed.

B. stays the same because cooling water pressure holds the charging water check valve closed.

C. lowers because the accumulator piston moves when charging water header pressure is lost.

D. lowers because the cooling water pressure lowers when charging water header pressure is lost.

Proposed Answer: C

Explanation (Optional): HC.OP-IS.BF-0103

Charging water check valve 115 maintains water volume on a loss of charging pressure from the CRD pumps initially. However, Accumulator gas pressures will begin to lower immediately after pump trip depending on the leak rate of the check valves. Actual plant experience demonstrated that the first alarm comes in at 2.1 minutes. N2 gas pressure will remain the same as long as the check valve holds. When the check valve begins to leak, the piston will stroke and N2 pressure will drop causing low accumulator pressure alarm.

C. Correct.

A. Incorrect. Lowers due to check valve leak by. Also reactor pressure does not hold the check valve closed
B. Incorrect. Lowers due to check valve leak by. Also water pressure does not hold the check valve closed.
D. Incorrect. The pressure lowers due to check valve leak by.

Technical Reference(s) HC.OP-IS.BF-0103 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: CRDHYDE017 (As available)

Question Source: Bank # 68914 NRC2002
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

Comments:

#60
NRC

1.0 PURPOSE

The purpose of this procedure is to demonstrate during plant refuel, the operability of the Control Rod Drive Accumulator Charging Water Check Valve, 1-BF-V115, as required by Technical Specification 4.0.5. This is performed by verifying each individual accumulator check valve maintains the associated accumulator pressure above the alarm setpoint for greater than or equal to 2 minutes, starting at normal operating pressure, with no Control Rod Drive Pump operating. [**CD-270A**]

2.0 PREREQUISITES

2.1 Charging Water Check Valve Exercise Test

2.1.1. Permission to perform this test has been obtained from the SM/CRS as indicated by the completion of Attachment 1, Section 1.0. ____

2.1.2. All personnel involved in the performance of this procedure, should complete Attachment 1, Section 3.0, prior to performing any part of this procedure. ____

2.1.3. No other testing OR maintenance is in progress that would adversely effect the performance of this test. ____

NOTE

All Control Rod Drive Scram Accumulators need not be OPERABLE to perform this test provided INOPERABLE accumulators are tracked IAW OP-HC-108-115-1001, Removal and Return of Equipment to Service, and this surveillance is listed in Part B of Attachment 1, Action Statement Log Sheet as required to restore the equipment to operability.

2.1.4. The Control Rod Drive system is in service AND all OPERABLE Hydraulic Control Units are charged to normal operating pressure IAW HC.OP-SO.BF-0001(Q), CRD Hydraulic System Operation. ____

2.1.5. All insertable control rods are inserted except for rods removed IAW Technical Specifications 3.9.10.1 and/or 3.9.10.2. ____

2.1.6. The plant is in Condition 3, 4 or 5. ____

2.1.7. Radiation Protection should be contacted prior to performing venting OR draining in this procedure. The individuals performing the venting OR draining shall obtain instructions AND approval from the Radiation Protection Shift Technician OR Radiation Protection Supervisor. ____

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295032 EK3.02	
	Importance Rating	3.6	

Knowledge of the reasons for the following responses as they apply to HIGH SECONDARY CONTAINMENT AREA TEMPERATURE: Reactor SCRAM.

Proposed Question: Common 61

Given the following conditions:

- An unisolable steam line leak has occurred in the RCIC room
- RCIC Equipment Room Area Temperature is 207°F and rising

Which of the following is the reason for initiating a Reactor Scram with the above conditions?

A. Emergency Depressurization is anticipated.

B. The scram will begin to reduce the energy that the RPV will discharge to the RCIC room.

C. A scram will reduce the driving head and flow through the break in the RCIC room to prevent the blowout panel from opening.

D. Failure of Secondary Containment due to high temperatures must be assumed and the scram will stop the radioactive release.

Proposed Answer: B.

Explanation (Optional):

From EOP-103 bases:
If temperatures or floor levels in any one of the ROOMS listed in Table 1 or 2 of Reactor Building Control approach their maximum safe operating value, adequate core cooling, containment integrity, safety of personnel, or continued operability of equipment required to perform EOP actions can no longer be assured. EOP-101 must be entered to make certain the reactor is scrammed. Scramming the reactor reduces to decay heat levels the energy that the RPV may be discharging to the reactor building.

A. Incorrect – the scram is not performed for anticipating an ED. Levels may not reach the ED requirement
B. Correct
C. Incorrect. – The blowout panel is not a concern per the bases.
D. Incorrect – The failure of Secondary Containment is not a concern at this point in the event with the conditions stated.

Technical Reference(s) EOP-103 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: EOP103E006 (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

Comments:

#61

NRC

system is the source of radioactivity, the action that is directed in RB-14 should be adequate to terminate any further increase in reactor building radiation levels.

If temperatures or floor levels in any one of the ROOMS listed in Table 1 or 2 of Reactor Building Control approach their maximum safe operating value, adequate core cooling, containment integrity, safety of personnel, or continued operability of equipment required to perform EOP actions can no longer be assured. EOP-101 must be entered to make certain the reactor is scrammed. Scramming the reactor reduces to decay heat levels the energy that the RPV may be discharging to the reactor building. However, an explicit direction to scram the reactor is not provided in this step.

Rather, entry (or re-entry) to EOP-101 is required. This accommodates the parallel execution of the reactor power control steps of RPV Control. if needed, and at the same time avoids unnecessarily cycling the control rod drive hydraulic system.

RB-18 and 19

WHEN the same parameter exceeds Max Safe Op Limit in 2 or more areas,

EMERGENCY RPV DEPRESSURIZATION IS REQUIRED.

Discussion

Should reactor building temperatures radiation levels or floor levels exceed their maximum safe operating values in more than one area, the RPV must be depressurized to preclude further temperature, increases. RPV depressurization places the primary system in its lowest possible energy state, rejects heat to the suppression pool in preference to outside the containment, and reduces the driving head and flow of primary systems that are unisolated and discharging into the reactor building.

The criteria of "more than one area" specified in this step identifies the rise in reactor building temperature, reactor building radiation level or reactor building water level as a wide-spread problem which may pose a direct and immediate threat to reactor building integrity, equipment located in the reactor building, and continued safe operation of the plant.

One parameter (e.g., temperature) above its maximum safe operating value in one area [example "A" & "C" Core Spray rooms] and a different parameter (e.g., radiation or water level) above its maximum safe operating value in the same or another area [example "A" and "C" RHR rooms] is not a condition which requires emergency RPV depressurization. A combination of parameters exceeding maximum safe operating values in one area does not necessarily indicate that control of a given parameter cannot be maintained or that previous actions have not been effective in confining the trouble to one area. Expanding the application of "more than one area" to encompass multiple parameters might lead to depressurization of the RPV when such action is not, in fact, appropriate or needed.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295029 EA1.01	
	Importance Rating	3.4	

Ability to operate and/or monitor the following as they apply to HIGH SUPPRESSION POOL WATER LEVEL: HPCI: Plant-Specific.

Proposed Question: Common 62

HPCI and RCIC both started and are injecting in response to a valid low reactor water level. Current plant conditions are as follows:

- Reactor water level is +25 inches, steady
- Reactor pressure is 845 psig, rising slowly
- Drywell pressure is 1.1 psig, steady
- RCIC has been aligned to Full Flow Recirc operation (CST to CST) for pressure control
- HPCI is injecting to the reactor for level control

After 10 minutes of operation, suppression pool level reaches 78.5"

Which of the following would be the response of HPCI & RCIC for the given conditions?

A. HPCI will continue to inject and RCIC will operate on minimum flow.

B. HPCI will continue to inject and RCIC will trip on low suction pressure.

C. HPCI will trip on low suction pressure and RCIC will operate on minimum flow.

D. HPCI will trip on low suction pressure and RCIC will trip on low suction pressure.

Proposed Answer: A.

A. Correct - The F011 closes on the HPCI Suppression Pool Suction Valve (F042) opening. HPCI will continue to inject, RCIC has no discharge path, Min. flow opens.

B. Incorrect - RCIC Suction flow path will remain on the CST.

C. Incorrect - HPCI will continue to inject, AP-HV-F011 closes in the return line to the CST.

D. Incorrect - HPCI will continue to inject, RCIC has no discharge path, Min. flow opens.

Technical Reference(s) HC.OP-SO.BJ-0001(Q), Sect 3.3 Interlocks (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: HPCI00E012 (As available)

Question Source: Bank # X
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

Comments:

#62 -NRC

3.3.4 (continued)

- BJ-HV-8278 (HV-F105) PMP DSCH TO FW ISLN MOV
  Auto closes IF FD-HV-F001 HPCI TURB STM SPLY is full closed (K57) AND * LINE FILL keyswitch is in NORM (K100)
  OR Test Selector switch is in HV-F006/HV-8278 (HV-F105) (AND Test Jack installed) (K78) AND * LINE FILL keyswitch is in NORM (K100)
  OR FD-FV-4880 HPCI TURB STOP is full closed (K79) AND * LINE FILL keyswitch is in NORM (K100). Auto opens on HPCI Initiation (K80) AND FD-FV-4880 HPCI TURB STOP VLV is not full closed (K79)
  AND FD-HV-F001 HPCI TURB STM SPLY not full closed (K57)
  AND HV-F006/HV-8278 not in test (K78). ____

- LINE FILL keyswitch interlock requires Reactor steam pressure < 100 psig on PT E41-N058A & N058E (K16A & K17A) AND keyswitch in BYPASS to defeat auto closure. ____

- BJ-HV-F008 TEST BYP TO CST MOV Auto closes on HPCI Initiation (K34) OR BJ-HV-F006 PMP DSCH TO CS ISLN not full closed (limit switch) OR BJ-HV-8278 (HV-F105) PMP DSCH TO FW ISLN MOV not full closed (limit switch). Opens manually IF no HPCI Initiation (K34) present AND both BJ-HV-8278 (HV-F105) PMP DSCH TO FW ISLN MOV AND BJ-HV-F006 PMP DSCH TO CS ISLN are full closed (limit switches). ____

- AP-HV-F011 HPCI/RCIC RET TO CST Auto closes IF BD-HV-F031 RCIC TORUS SUCT VLV is full open (K76) OR HPCI Initiation (K34) OR BJ-HV-F042 PMP SUCT FROM SUPP CHB is full open (K43). Opens manually IF no HPCI Initiation (K34) AND BJ-HV-F042 PMP SUCT FROM SUPP CHB is not full open. ____

Continued on next page

#62-NRC

3.3.4. HPCI valve interlocks are as follows:

- FD-HV-F002 INBD HPCI STM ISLN MOV Auto Closes on HPCI Div 3 Isolation signal (K50C). Opens manually if no HPCI Div 3 Isolation signal (K50C). IF Isolation signal was sealed in (K92) with FD-HV-F002 handswitch in OPEN THEN handswitch must be placed in CLOSE THEN OPEN. ____

- FD-HV-F003 OUTB HPCI STM ISLN MOV Auto closes on HPCI Div 1 Isolation signal (K50A). Opens manually if no HPCI Div 1 Isolation signal (K50A). IF Isolation signal was sealed in (K93) with FD-HV-F003 handswitch in OPEN THEN handswitch must be placed in CLOSE THEN OPEN. ____

- FD-HV-F001 HPCI TURB STM SPLY Auto opens on HPCI Initiation (K33), IF FD-HV-F071 TURB EXH ISLN is full open. Opens manually IF FD-HV-F071 TURB EXH ISLN is full open. ____

- FD-HV-F100 HPCI W/U VLV Auto closes on HPCI Div 3 Isolation signal (K50C & K52C). Opens manually IF no HPCI Div 3 Isolation signal (K50C). ____

- BJ-HV-F004 PMP SUCT FROM CST Auto closes IF BJ-HV-F042 PMP SUCT FROM SUPP CHB is full open (K43). Auto opens IF BJ-HV-F042 PMP SUCT FROM SUPP CHB is not full open (K43) AND HPCI Initiation (K33). ____

- BJ-HV-F042 PMP SUCT FROM SUPP CHB Auto closes on HPCI Div 1 Isolation signal (K51A). Auto opens on CST low level OR Suppression Chamber high level (K42), IF BJ-HV-F042 handswitch is not in AUTO OPEN OVRD AND no HPCI Div 1 Isolation signal (K51A). Opens manually IF no HPCI Div 1 Isolation signal (K51A). ____

- BJ-HV-F007 PMP DSCH ISLN Auto closes IF Test Selector switch is in HV-F007 position AND Test Jack installed (K66). Auto opens on HPCI Initiation (K32) AND not in test (K66). ____

- BJ-HV-F006 PMP DSCH TO CS ISLN Auto closes IF FD-HV-F001 HPCI TURB STM SPLY is full closed (K44)
  OR Test Selector switch is in HV-F006/HV-8278 (HV-F105) (AND Test Jack installed) (K65)
  OR FD-FV4880 HPCI TURB STOP VLV is full closed. Auto opens on HPCI Initiation (K90)
  AND FD-FV4880 HPCI TURB STOP VLV is not full closed (K39)
  AND FD-HV-F001 HPCI TURB STM SPLY is not full closed (K44) AND not in test (K65). ____

Continued on next page

H#63

SUBSEQUENT OPERATOR ACTIONS

CONDITION	ACTION
A. Unexpected rise in Drywell Pressure. Date/Time: _____	☐ A.1 **TERMINATE** Containment Makeup AND Inerting. A.2 **MAXIMIZE** Drywell Cooling by **ENSURING**: ☐ • All Drywell Fan Cooling Coils are Open. ☐ • All Drywell Fans are running in Fast Speed. ☐ **NOTE 1** ☐ • Proper TBCW system operation A.3 **PERFORM** the following: ☐ • Check Reactor Recirc. Pump Seals. ☐ • Check SRV Tailpipe Temperatures. ☐ • Drywell Leakage Source Detection IAW GP.ZZ-0005.
B. Turbine Bldg. Chill Water System is lost to the Drywell. Date/Time: _____	☐ ☆ **CAUTION 1** ☆ B.1 **ALIGN** RACS to the Chill Water System for Drywell Cooling as follows: a. **ENSURE** RACS to the out of service Off-Gas Train is ISOLATED as follows: ☐ • IF the Common Off-Gas Train is in service, THEN **CLOSE** HV-2577. ☐ • IF Unit 1 Off-Gas Train is in service, THEN **CLOSE** HV-7712A1. ☐ b. **CLOSE** HV-9532-1 AND HV-9532-2. ☐ c. **PRESS** LOOP A SPLY/RTN OPEN RACS PB. ☐ d. **PRESS** LOOP B SPLY/RTN OPEN RACS PB. e. **OBSERVE** the following indications: ☐ • HV-9530A1/A3 CLOSED ☐ • HV-9530B1/B3 CLOSED ☐ • HV-9530A2/A4 OPEN ☐ • HV-9530B2/B4 OPEN ☐ f. **OPEN** HV-9532-1 AND HV-9532-2.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295012 AA2.02	
	Importance Rating	3.9	

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

Proposed Question: Common 63

Given the following conditions:

- The plant is operating at 100% power.
- The COMPUTER PT IN ALARM A4-F5 alarm is received.
- Drywell pressure is 1.1 psig.
- HC.OP-AB.CONT-0001, Drywell Pressure abnormal is entered.

Which of the following events, BY ITSELF, could be the cause of the pressure rise?

A. Failure of the "A" Reactor Recirculation Pump #2 Seal

B. FV-4971 Nitrogen Flow Control Valve fails open

C. Torus Vent Valve Isolation Valve HV-11541 fails open

D. Loss of power to multiple Drywell Fans

Proposed Answer: D.

Explanation (Optional):

D. Correct: A reduction in cooling will raise temperature and therefore pressure in the Drywell.

A. Incorrect: Assuming the #1 seal is intact, no change in DW conditions will occur.
B. Incorrect: During normal operation, the nitrogen FCV is isolated from the DW.
C. Incorrect: A vent valve opening would result in a reduction in pressure. However the rupture disk downstream of HV-11541 should be intact resulting in no effect on DW pressure.

Technical Reference(s) HC.OP-AB.CONT-0001 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: ABCNT1E004 (As available)

Question Source: Bank # 61761

Modified Bank # _____ (Note changes or attach parent)

New _____

Question History: Last NRC Exam 2007

Question Cognitive Level: Memory or Fundamental Knowledge _____

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

_____

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295009 G2.1.20	
	Importance Rating	4.6	

Conduct of Operations: Ability to interpret and execute procedure steps. (Low reactor water level)

Proposed Question: Common 64

Given the following conditions:

- A startup following a refueling outage was in progress when a loss of offsite AC Power occurred.
- Only A, C, and D Emergency Diesel Generators are running.
- HPCI and RCIC are NOT available.
- All control rods are at 00.
- RPV water level is stable at (-35) inches.
- RPV pressure is stable at 910 psig.
- NO operator actions have been taken.

Which of the following statements describes the actions required for the conditions above?

A. Restore and maintain level to +12.5 to +54 inches by maximizing CRD flow.

B. Lower reactor pressure to 600 psig and restore level using the Secondary Condensate Pumps.

C. Emergency Depressurize the reactor and restore level using the low pressure ECCS systems.

D. Override 1E Breakers and restore RFPTs to raise RPV level to between +12.5 and + 54 inches.

Proposed Answer: A

Explanation (Optional):

A. Correct - No ATWS exists and Level is not lowering. A, C and D diesels allows use of RACS and 2 CRD pumps. CRD is a Preferred Table 1 system 0-1500 psig.

B. Incorrect – This would be the normal method, but condensate has lost power.

C. Incorrect - ED not required. RPV level still too high.

D. Incorrect – No power for secondary and primary condensate pumps

Technical Reference(s) HC.OP-EO.ZZ-0101, Steps RC/L-2 thru RC/L-5 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: EO101LE006 (As available)

Question Source: Bank # ID: Q76668
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295013 AK3.02	
	Importance Rating	3.6	

Knowledge of the reasons for the following responses as they apply to HIGH SUPPRESSION POOL TEMPERATURE: Limiting heat additions.

Proposed Question: Common 65

The plant is operating at 80% power with RCIC quarterly testing in progress.

Which one of the following is the reason for the Technical Specification temperature limitation for the suppression pool during the test?

A. To assure primary containment integrity following a stuck open Safety Relief Valve.

B. To assure that excessive steam condensing loading does NOT occur during the test.

C. To assure that sufficient RHR and Core Spray NPSH exists following a containment blowdown in conjunction with containment overpressure.

D. To assure sufficient RHR and Core Spray NPSH exists during LOCA conditions without overpressure.

Proposed Answer: D.

Explanation (Optional):

D. Correct - TS requires any testing that adds heat to the SP be terminated at 105°F the TS bases is PC integrity during a LOCA and NPSH without overpressure for RHR and Core Spray.

A. Incorrect - the bases is for a LOCA.

B. Incorrect - not a concern during the test

C. Incorrect - NPSH without overpressure for RHR and Core Spray.

Technical Reference(s) T.S. 3.6.2 Bases (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: _____ (As available)

Question Source: Bank # _____
Modified Bank # _____ (Note changes or attach parent)
New X

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 X

Comments:

#65 NRC

DEPRESSURIZATION SYSTEMS (Continued)

Under full power operating conditions, blowdown from an initial suppression chamber water temperature of 95°F results in a water temperature of approximately 135°F immediately following blowdown which is below the 200°F used for complete condensation via mitered T-quencher devices. At this temperature and atmospheric pressure, the available NPSH exceeds that required by both the RHR and core spray pumps, thus there is no dependency on containment overpressure during the accident injection phase. If both RHR loops are used for containment cooling, there is no dependency on containment overpressure for post-LOCA operations.

Experimental data indicates that excessive steam condensing loads can be avoided if the peak local temperature of the suppression pool is maintained below 200°F during any period of relief valve operation. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings.

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature normally changes very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be frequently recorded during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be the points of highest stress.

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a safety-relief valve inadvertently opens or sticks open. As a minimum this action shall include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling, (3) initiate reactor shutdown, and (4) if other safety-relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open safety relief valve to assure mixing and uniformity of energy insertion to the pool.

In conjunction with the Mark I containment Long Term Program, a plant unique analysis was performed which demonstrated that the containment, the attached piping and internal structures meet the applicable structural and mechanical acceptance criteria for Hope Creek. The evaluation followed the design basis loads defined in the Mark I Load Definition Report, NEDO-21888, December 1978, as modified by NRC SER NUREG 0661, July 1980 and Supplement 1, August 1982, to ensure that hydrodynamic loads, appropriate for the life of the plant, were applied.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	1	
	K/A #	G2.1.30	
	Importance Rating	4.4	

Ability to locate and operate components, including local controls.

Proposed Question: Common 66

Given the following conditions:

- Control Room has been abandoned
- Control has been transferred to the Remote Shutdown Panel (RSP)
- RPV level is +10"
- RPV pressure is 80 psig

Which one of the following describes the ability to operate the BC-HV-F009 SDC Suction Isolation valve and whether the valve would close automatically if reactor pressure exceeded 82 psig?

A. The valve CAN be opened at the RSP.
If already open, the valve would automatically close.

B. The valve CAN be opened at the RSP.
If already open, the valve would NOT automatically close.

C. The valve CANNOT be opened at the RSP.
If already open, the valve would NOT automatically close.

D. The valve CANNOT be opened at the RSP.
If already open, the valve would automatically close.

Proposed Answer: B

Explanation (Optional): Note and a Caution in HC.OP-IO.ZZ-0008 (5.9.6 of Rev 28) - CAUTION WHEN the RSP Transfer Switch is placed in EMER, RHR S/D Cooling interlocks for overpressure AND low Reactor level are inoperable. RX pressure of 80 psig should NOT be exceeded WITH Suction Valves F008 & F009 open.

B. Correct. When control is transferred to the RSP, both the Low RPV Water Level AND high RPV Pressure isolations for the BC-HV-F009 are defeated. There remains a pressure switch permissive in series with the opening contactor that requires reactor pressure to be below 82 psig to open the valve (this is NOT a function of NSSSS). This is identified in a

Note and a Caution in HC.OP-IO.ZZ-0008 (5.9.6 of Rev 23).

A. Incorrect - The valve will NOT isolate if reactor pressure exceeds 82 psig.
C. Incorrect. The valve CAN be opened, since the RPV Low Water Level isolation is defeated and there is NO Low Water Level opening permissive.
D. Incorrect. The valve can be opened. The valve will NOT isolate if reactor pressure exceeds 82 psig.

Technical Reference(s) HC.OP-IO.ZZ-0008 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: IOP008E006 (As available)

Question Source: Bank # 53914
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

Comments:

5.9.5. BEFORE Reactor Pressure decreases to < 660 psig, THEN **VERIFY** all Primary AND Secondary Condensate Pumps are secured. (Local) ____

NOTE

IF the A RHR Loop is in Suppression Pool Cooling:

- The A RHR Loop should be maintained in operation while the B RHR Loop is placed in Shutdown Cooling operation.
- IF time AND personnel are available, the B RHR Loop can be drained and filled with CST water (locally) prior to reaching 65 psig to 80 psig.

High Reactor pressure will prevent opening of the shutdown cooling valves but will NOT isolate the valve IF pressure rises above setpoint.

CAUTION

WHEN the RSP Transfer Switch is placed in EMER, RHR S/D Cooling interlocks for overpressure AND low Reactor level are inoperable. RX pressure of 80 psig should NOT be exceeded WITH Suction Valves F008 & F009 open.

5.9.6. At < 80 psig, **PERFORM** the following: **[CD-987X, CD-370X]**

A. IF B RHR Loop is operating in Suppression Pool Cooling, THEN **REMOVE** it from operation per Step 5.8.2. ____

B. IF a LOP has occurred AND continued cooldown is required **INITIATE** alternate S/D Cooling IAW Att. 10. **[PR 981228174]** ____

C. **PLACE** B RHR in Shutdown Cooling Operation IAW Attachment 4. ____

D. IF DESIRED to place "A" RHR Loop in Shutdown Cooling, **PLACE** A RHR in Shutdown Cooling Operation IAW Attachment 11. ____

5.9.7. **MAINTAIN** a cooldown rate < 90°F/hr by throttling open/closed HV-F048B RHR HX B SHELL SIDE BYP MOV. ____

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	1	
	K/A #	G2.1.4	
	Importance Rating	3.3	

Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, "no-solo" operation, maintenance of active license status, 10CFR55, etc.

Proposed Question: Common 67

You are a licensed Reactor Operator. Due to illness, you have worked the following schedule over the past quarter (July thru September).

July 1 - Off
July 2 - Off
July 3 - 12 hour day shift as RO
July 4 -12 hour day shift as RO
July 8 -12 hour night shift as RO
July 9 - 12 hour night shift as RO
July 10 Through September 30 – Off Shift due to illness.

All licensed operator training is up to date.
You're received medical clearance to stand watch.

Which one of the following describes the status of your license and additional requirements, if any, to stand watch on October 1st IAW OP-AA-105-102 "NRC ACTIVE LICENSE MAINTENANCE?

A. Your license is Active because you stood watch for at least 40 hours the previous quarter, no additional requirements are needed to stand watch on 10/1.

B. Your license is Inactive. You must reactivate your license by performing shift functions under the sole direct supervision of an active licensed RO or SRO for at least 40 hours.

C. Your license is Inactive. You must reactivate your license by performing shift functions under the sole direct supervision of ONLY an active licensed SRO for at least 40 hours.

D. Your license is Inactive. You must reactivate your license by performing shift functions under the sole direct supervision of ONLY an active licensed RO for one additional 12 hour shift.

Proposed Answer: B

Explanation (Optional):

Explanation (Optional): IAW OP-AA-105-102 "NRC ACTIVE LICENSE MAINTENANCE, Steps 4.1.1. & 4.2.1

MAINTAIN an active license by actively performing the functions of RO, SRO, or LSRO.
1. RO licenses by performing the duties of the Unit RO and/or Unit Assist RO for a minimum of seven 8-hour or five 12-hour shifts per calendar quarter, including turnover to the next shift.

REACTIVATE an RO or SRO license to an "active status" by performing 40 hours of shift functions in the presence and under the sole direct supervision of an active RO or SRO, as appropriate and in the position to which the individual will be assigned.

A. Incorrect. License is inactive. Previous quarter requirements not met
B. Correct.
C. Incorrect. An RO is required
D. Incorrect. Previous quarter requirements not met with one additional shift on 10/1. One day too late.

Technical Reference(s) OP-AA-105-102 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: NOH04ADM062C-01 (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 X

Comments:

#67
NRC

4.2. License Reactivation

NOTE: If more than 3 months have passed between the time of NRC examination results being issued and issuance of the NRC license, then the license must be activated in accordance with the requirements of this procedure and 10CFR55.53(f).

4.2.1. **REACTIVATE** an RO or SRO license to an "active status" by performing 40 hours of shift functions in the presence and under the sole direct supervision of an active RO or SRO, as appropriate and in the position to which the individual will be assigned. The 40 hours will include completion of a position-specific activation guide containing specific and detailed activation requirements (if required), a plant tour in the presence and under the sole direct supervision of an active RO or SRO, as appropriate, and a review of the position-specific turnover procedures. LSRO's (or SRO licenses that will be activated for fuel handling only) need to complete only one 8-hour shift under the sole direction of an active SRO or LSRO. All parts of the reactivation for LSRO, or SRO licenses activated for fuel handling only, will be performed with the accompaniment of an active SRO or LSRO.

4.2.2. **DOCUMENT** the reactivation on Attachment 2, Reactivation of License Log .

1. The Shift Manager shall signify that the required OJT hours were completed.

2. The Operations Training Manager shall signify that the position-specific activation guide was completed (if required).

3. **NOTIFY** the Security Shift Supervisor to update the Main Control Room SRO tracking program, if appropriate.

4. **If** an SRO license is reactivated as LSRO, **then RESTRICT** LSRO from performance of the SRO duties.

4.3. **NOTIFY** the Shift Operations Superintendent in the event an individual fails to meet the Active License requirements defined in Section 4.1.

4.3.1. **REMOVE** the individual from the "Active License List".

4.3.2. **NOTIFY** the Security Shift Supervisor to remove the individual from the Main Control Room SRO Tracking Program, if appropriate.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	2	
	K/A #	G2.2.6	
	Importance Rating	3.0	

Knowledge of the process for making changes to procedures

Proposed Question: Common 68

Given the following conditions:

- The plant is operating at rated power
- The Common Offgas train is experiencing problems and must be swapped before vacuum starts degrading.

While performing the evolution brief, a critical procedure step was found to be missing.

Which of the following describes the requirement, if any, to continue the evolution?

A. Complete the evolution as written then perform a permanent revision change after the evolution is complete.

B. A procedure change request is required and an on-the-spot-change can be made.

C. Obtain verbal concurrence from the CRS to change the sequence of steps and continue.

D. A full procedure revision to the Offgas system operating procedure is required. An on-the-spot change CANNOT be performed.

Proposed Answer: B

Explanation (Optional): IAW AD-AA-101-101

B. Correct. An on-the-spot-change (OTSC) may used.

A. Incorrect. If an error is found in the procedure actions must be taken to correct the issue before proceeding.
C. Incorrect. Written documentation is required (On-the-spot change)
D. Incorrect. An on-the-spot change may be used

Technical Reference(s) AD-AA-101-101 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: ADMPROE002 (As available)

Question Source: Bank # _____
Modified Bank # _____ (Note changes or attach parent)
New X

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 X

Comments:

#68
NRC

PSEG IMPLEMENTING PROCEDURE ON THE SPOT CHANGE (OTSC) PROCESS

1. PURPOSE

1.1. This procedure provides instructions for performing On-The-Spot-Changes (OTSCs) to PSEG Nuclear's Department Implementing Procedures (DIPs). OTSCs are allowed for PSEG Station Implementing Procedures ONLY.

1.2. On-The-Spot-Change (OTSC) process may be used to revise a procedure **when** the change is of an urgent nature **and** routine procedure revision process will not support resolution of condition in a timely manner.

1.3. OTSC process is subject to restrictions listed in 1.3.1 and 1.3.2, for conditions found during pre-job review, or during field or shop use of a DIP in conjunction with a work activity that would require a procedure revision be made before beginning activity or prior to continuing activity because procedure cannot be performed as written (cannot use procedure as-is).

1.3.1. **Permitted Uses**

- OTSC allowed to modify procedure for temporary conditions which would not require a permanent change to procedure.
- OTSC allowed to modify procedure in accordance with an approved calculation, engineering procedure or engineering evaluation that does not change intent of procedure as described in Attachment 1, Change of Intent Criteria.
- OTSC to Q, F and R designated procedure is subject to same level of review and approval as required for procedure being revised.

1.3.2. **Prohibited Uses [80073759]**

- OTSC shall **not** be used for Administrative Procedures.
- OTSC shall **not** change intent of procedure as described in Attachment 1, Change of Intent Criteria.
- OTSC shall **not** be used to change procedures:
  - For rewording or clarification of statements based on personnel preferences
  - For reordering of steps, except when failure to do so will prevent task completion

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	2	
	K/A #	G2.2.12	
	Importance Rating	3.7	

Knowledge of surveillance procedures

Proposed Question: Common 69

The CRS has directed you to perform a RCIC inservice test following maintenance using HC.OP-IS.BD-0001, Reactor Core Isolation Cooling (RCIC) Pump - OP203 - Inservice Test.

Which one of the following describes a surveillance procedure requirement due to the performance of the test?

A. Suppression Pool Temperature Monitoring prior to starting the RCIC pump.

B. Suppression Pool Cooling when Suppression Pool temperature exceeds 95 degrees F.

C. RCIC must be secured when Suppression Pool temperature reaches 110 degrees F.

D. Remote Shutdown System Suppression Pool Temperature Instrumentation Channel Check.

Proposed Answer: A

Explanation (Optional):

A. Correct

B. Incorrect – SPC is required prior to placing the pump in service
C. Incorrect – RCIC must be secured when temperature reaches 105 dgrees F.
D. Incorrect – not required by the procedure.

Technical Reference(s) HC.OP-IS.BD-0001 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: _____ (As available)

Question Source: Bank # INPO 19132

Modified Bank # _____ (Note changes or attach parent)

New _____

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 X

_____

Comments:

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NRC

NOTE

Installation in the following step must be performed by qualified personnel.

5.12 **DIRECT** qualified personnel to perform the following:

5.12.1. **ENTER** test equipment data on Attachment 3, Inplant Data Sheet. ____

5.12.2. **INSTALL** the Measuring and Test Equipment (M&TE), specified in Section 4.0, IAW Attachment 4, M&TE Installation and Removal. ____

5.12.3. **INSTALL** Key Phasor Tach. at Turbine coupling end AND **POSITION** RPM indicator near page for communication with the Control Room. ____

5.12.4. **INITIAL** Attachment 3, Inplant Data Sheet performer/verifier. ____

5.13 IF GETARS is to be used,
REQUEST STA or System Engineer **START** GETARS. ____

5.14 **PERFORM** pre-start checks and alignment as follows:

5.14.1. **RECORD** RCIC Pump lubrication level on Attachment 3. ____

NOTE

The following step requires establishment of communication between the Control Room and operator stationed locally at valve to observe movement.

5.14.2. WHILE verifying approximately 11 - 13 seconds of valve movement locally **THROTTLE OPEN** HV-F022 RCIC TEST BYP TO CST ISLN VLV approximately 11 - 13 seconds (Control Room) AND **INITIAL** Attachment 3. ____

NOTE

Tech Spec 3.6.2.1 addresses the Suppression Pool Average Water temperature requirements.

5.15 **PERFORM** Attachment 3m, Containment Systems TS 4.6.2.1.b.1, of HC.OP-DL.ZZ-0026(Q), Surveillance Log. [**TS 3.6.2.1**] ____

THIS IS TEMPERATURE MONITORING

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NRC

5.16 IF the following Suppression Pool Average Water Temperature(s) are reached during the performance of this test the indicated actions shall be taken:

5.16.1. 95°F, **ENSURE** RHR is in the Suppression Pool Cooling mode. (Entry into HC.OP-EO.ZZ-0102(Q) is not required at this time). ____

5.16.2. 105°F; **IMMEDIATELY TERMINATE** this test AND **STOP** all testing that adds heat to the Suppression Pool. ____

5.16.3. IF Suppression Pool Average Water Temperature can not be maintained below 105°F THEN **ENTER** HC.OP-EO.ZZ-0102(Q). ____

NOTE

Tech Spec 3.6.2.1 and 3.5.3 address the Suppression Pool Water Level requirements.

5.17 **MAINTAIN** Suppression Pool level at 75-77 inches during the performance of this test using either one of the following procedures as appropriate: ____

- HC.OP-SO.EE-0001(Q), Torus Water Cleanup System Operation.
- HC.OP-SO.BC-0001(Q), RHR System Operation

5.18 IF any of the following occur during operation of RCIC THEN **TRIP** the Turbine immediately: ____

- Excessive vibration
- Excessive oil temperature
- Sudden drop in oil pressure
- Other unusual operating conditions are noted.
- Pump discharge pressure reaches the indicated maximum range of 1500 psig.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	3	
	K/A #	G2.3.4	
	Importance Rating	3.2	

Knowledge of radiation exposure limits under normal or emergency conditions.

Proposed Question: Common 70

A Hope Creek operator has received the following dose:

- November 1, 2008 thru November 21, 2008 - 350 mrem - while visiting a foreign nuclear plant as part of a Technical Exchange Program.
- July 1, 2008 thru December 31, 2008 – 175 mrem - while working at Hope Creek.
- January 1, 2009 thru January 31, 2009 – 125 mrem while working at Hope Creek.

Which of the following describes the MAXIMUM additional non-emergency Total Effective Dose Equivalent (TEDE) that this individual could receive at Hope Creek through October 31, 2009?

A. 1350 mrem

B. 1700 mrem

C. 1875 mrem

D. 2375 mrem

Proposed Answer: C

Explanation (Optional): IAW

C. Correct – 2000 – 125 = 1875

A. Incorrect. – prior year is not included
B. Incorrect. – prior year is not included
D. Incorrect. – limit is 2000 not 2500 per year

Technical Reference(s) RP-AA-203 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: NOH04ADM024C-01 (As available)

Question Source: Bank # _____
Modified Bank # 77351 (Note changes or attach parent)
New _____

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X
_____

Comments:

70

NAC

NOTE: Any request to raise the administrative dose control level for a minor shall be approved and documented by the Radiation Protection Manager.

4.1.2. Administrative dose control levels have been established for Total Effective Dose Equivalent Limits as follows:

- 2000 mrem routine cumulative TEDE/yr.
- 100 mrem TEDE for minors.

4.1.3. The Radiation Protection Manger shall review an individual's occupational exposure when the dose equivalent reaches 80% of the NRC Limits for Lens Dose Equivalent (LDE), Shallow Dose Equivalent (SDE), and Total Organ Dose Equivalent (TODE). The 80% threshold values are as follows:

- 12 rem LDE.
- 40 rem SDE.
- 40 rem TODE.

4.1.4. **If** an individual's current year dose history documentation includes an absent/ **no** record (A) dose type, **then REDUCE** the individual's allowable exposure (normally 2000 mrem TEDE for the year) by 1250 mrem TEDE for each quarter of the current year for which dose history documentation is absent/ **no** record (A), until all of that dose is resolved. All reductions require RPM approval.

4.1.5. **If** an individual is suspected of exceeding any of the NRC exposure limits in Table 1, **then PROHIBIT** the individual from entering the RCA until a detailed evaluation of the individual's actual dose equivalent has been conducted. Future access will depend upon the results of the evaluation.

1. **If** an exposure in excess of the applicable exposure limit has occurred, **then PROHIBIT** the individual from entering the RCA until the end of the current calendar year.

2. If an exposure in excess of the applicable exposure limit has **not** occurred, then the individual may be permitted to re-enter the RCA.

4.1.6. During a condition where the Generating Stations Emergency Plan has been initiated, emergency exposure authorizations shall be performed in accordance with the station's Emergency Plan Implementing Procedures.

#70

In one calendar year, a NEO received 450 mrem while visiting a foreign nuclear plant as part of a Technical Exchange Program. The Operator's prior exposure at Hope Creek was 175 mrem for that same year.

If no exposure limit extensions have been authorized, which of the following list the MAXIMUM additional non-emergency Total Effective Dose Equivalent (TEDE) that this individual could receive at Hope Creek for the remainder of that year?

A. 1375 mrem

B. 1825 mrem

C. 3375 mrem

D. 3825 mrem

Answer: A

Associated objective(s):

NOH04ADM024C-01 From Memory Describe what the worker is acknowledging when signing a RWP prior to use. IAW NC.NA-AP.ZZ-0024, Radiation Protection Program

License

Question 286 Details					
Topic:	L/B/TEDE calculation				
System ID:	33658	User ID:		Q77351	
Difficulty:	3.00	Time to Complete:		3	
Point Value:	1.00	Cross Reference:			
KA	2..3.1	RO Value	2.6	SRO Value	3.0

Question 286 Comments

NC.NA-AP.ZZ-0024, Attachment 1, administrative dose level (TEDE) is 2000 mREM/year.
In one calendar year, a NEO received 450 mrem while visiting a foreign nuclear plant as part of a Technical Exchange Program. The Operator's prior exposure at Hope Creek was 175 mrem for that same year.
450 + 175 = 625 mrem
2000 - 625 = 1375

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	3	
	K/A #	G2.1.3	
	Importance Rating	3.7	

Knowledge of shift or short-term relief turnover practices

Proposed Question: Common 71

Which one of the following describes Reactor Operator pre-and post-shift relief actions that should be implemented by the oncoming operator IAW OP-AA-112-101 "Shift Relief and Turnover"?

A. PRIOR to relief, read the Control Room logs through the last previous date on shift, or the preceding four (4) days logs, whichever is less.
PRIOR to relief, review the Daily Orders.
POST relief, tour the main control room back panels.

B. PRIOR to relief, read the Control Room logs through the last previous date on shift, or the preceding seven (7) days logs, whichever is less.
PRIOR to relief, tour the main control room back panels.
POST relief, review the Daily Orders.

C. PRIOR to relief, read the Control Room logs through the last previous date on shift, or the preceding four (4) days logs, whichever is less.
PRIOR to relief, tour the main control room back panels.
POST relief, review the Daily Orders.

D. PRIOR to relief, read the Control Room logs through the last previous date on shift, or the preceding seven (7) days logs, whichever is less.
PRIOR to relief, review the Daily Orders.
POST relief, tour the main control room back panels.

Proposed Answer: C

Explanation (Optional): C. Correct

IAW OP-AA-112-101 "Shift Relief and Turnover" – Section 4.8.3. - Prior to relief, the on-coming Reactor Operators should PERFORM the following:

– READ the Control Room logs through the last previous date on shift, or the preceding four days logs, whichever is less.

– DISCUSS with the off-going Reactor Operator all items listed on the turnover sheet, Shiftly and Daily Surveillance, and any other information pertinent to

proper continuity of operations.
- TOUR Main Control Room back panel areas
After relief, the on-coming RO's should PERFORM the following:
- ANNOUNCE shift turnover and relief to the Unit Supervisor.
- CONFER with the Unit Supervisor to determine the scope of planned shift activities and their responsibilities for that shift.
- REVIEW Daily Orders.
- REVIEW Standing Orders for new entries.

A. Incorrect – the back panels are toured prior to relief, the daily orders are reviewed post relief.
B. Incorrect – only the preceeding 4 days logs should be reviewed
D. Incorrect - the back panels are toured prior to relief, the daily orders are reviewed post relief. Only the preceeding 4 days logs should be reviewed.

Technical Reference(s) IAW OP-AA-112-101 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: ADMPRO102E004 (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 X

Comments:

Q.71

4.7. Field Supervisor Shift Turnover Responsibilities:

4.7.1. **INITIATE** a turnover sheet **and ENTER** the current status for their respective responsibilities (Off-going Field Supervisor). Examples of items that may be included are:

- Systems status in plant.
- Alarms in plant.
- Ground & Test Devices installed.
- Abnormal controllers in plant.
- Abnormal instrumentation in plant.
- Special chemistry or radwaste evolutions in progress or pending.
  - Releases
- Switchyard evolutions in progress or pending switching orders.
- Radiation Precautions.
- Secondary/Balance of Plant (BOP) steam leaks.
- Miscellaneous Secondary/BOP items.
- Fire Brigade Leader Duties
- Fire Protection System status and impairments.

4.7.2. **CONFER** with the Shift Manager concerning the scope of the shift and applicable responsibility for that shift.

4.8. Reactor Operator Shift Turnover Responsibilities:

4.8.1. **PERFORM** turnover in the Main Control Room. The Reactor Operator position must be filled by a qualified licensed reactor operator.

4.8.2. **INITIATE** a turnover sheet **and ENTER** the current status for their respective Unit (Off-going Reactor Operator). Examples of items that may be included are:

- Unit status,
- LCOAR / LCO status,
- Major Out-Of-Services,
- Abnormal positioned components,
- Daily orders,
- Surveillances,
- Major evolutions in progress or planned.

4.8.3. Prior to relief, the on-coming Reactor Operators should **PERFORM** the following:

- **READ** the Control Room logs through the last previous date on shift, or the preceding four days logs, whichever is less.
- **DISCUSS** with the off-going Reactor Operator all items listed on the turnover sheet, Shiftly and Daily Surveillance, and any other information pertinent to proper continuity of operations.

Q. #71

- **TOUR** the Main Control Boards with the off-going Reactor Operators and **DISCUSS** the following as a minimum:
  - Status of safety-related systems.
  - Running equipment and safety train alignments.
  - Inoperable equipment, including instrumentation, and Limiting Conditions of Operations, including surveillance requirements.
  - Contingency actions / abnormal procedures when redundant pieces of equipment are unavailable (e.g., stator cooling pump)
  - Reasons for new annunciator alarms.
  - Out-of-service equipment including any surveillance or equipment work in progress at time of the shift relief.
  - Abnormal events occurring during the last 24 hours.
- **TOUR** Main Control Room back panel areas

4.8.4. After relief, the on-coming RO's should **PERFORM** the following:

- **ANNOUNCE** shift turnover and relief to the Unit Supervisor.
- **CONFER** with the Unit Supervisor to determine the scope of planned shift activities and their responsibilities for that shift.
- **REVIEW** Daily Orders.
- **REVIEW** Standing Orders for new entries.

4.9. <u>Radwaste Panel Operator Turnover Responsibilities</u>

4.9.1. **PERFORM** turnover at the radwaste panel as appropriate.

4.9.2. **INITIATE** a turnover sheet and ENTER the current status for their respective responsibilities (Off-going Radwaste panel operator). Examples of items that may be included are:

- Liquid Systems.
- Gaseous Systems.
- Waste Water Treatment status.
- Sewage Treatment status.
- Makeup Demins status.
- CP Demins status.
- Shipments status.
- LCO / TRM / RETS / DEL status.
- Need to Order status.
- Gas Decay Tanks status.
- Abnormal Radwaste Controllers.
- Abnormal Radwaste Instrumentation.
- Water Inventory status in Radwaste.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	4	
	K/A #	G2.4.43	
	Importance Rating	3.2	

Knowledge of emergency communications systems and techniq ues.

Proposed Question: Common 72

An Alert has been declared at Hope Creek. You have been designated as the Secondary Communicator (CM2).

IAW the ECG Secondary Communicator Log, you are required to_____.

A. activate ERDS within 30 minutes from the Shift Manager OR Control Room SPDS terminal.

B. activate ERDS within 60 minutes from the Shift Manager OR Control Room SPDS terminal.

C. establish communications with state and local organizations using ERDS within 30 minutes.

D. establish communications with state and local organizations using ERDS within 60 minutes.

Proposed Answer: B

Explanation (Optional):

B. Correct – IAW ECG Att.8 section A.4.b.

A. Incorrect - required within 60 minutes

C. Incorrect – ERDS is not used for this purpose

D. Incorrect - ERDS is not used for this purpose

Technical Reference(s) ECG, ATT 8. Section A.4.b. (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: _____ (As available)

Question Source: Bank # _____

Modified Bank # _____ (Note changes or attach parent)

New X

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 X

Comments:

#72
NRC

A. NOTIFICATIONS (cont'd)

b. ACTIVATE **ERDS** within 60 minutes from EITHER the SM Office or the CR SPDS terminal;
   1) PRESS <**TOP**> button.
   2) Under Displays, PRESS <**ERDS**> button.
   3) Under ACTION, PRESS <**START ERDS LINK**> button

NOTE
"Link Denied" may be displayed until NRC accepts transmission of data, which may take several minutes.

Initials

4) ERDS LINK STATUS will read "TRANSMITTING DATA". _____ CM2

NOTE
See pages 11 & 12 of this attachment as needed for fax machine information.

5. OBTAIN a copy of the **ICMF** and FAX the ICMF to Group A (EOF2 – FAX to Group C).

_____ CM2/TSC2/EOF2

6. COMPLETE a **Station Status Checklist (SSCL)** Form, Pg. 8 or Common Site UNUSUAL EVENT **Station Status Checklist (SSCL)** Form, Pg. 10;

( ) a. OBTAIN SM (TSS/SSM) assistance, as needed for Pg. 1.
( ) b. OBTAIN SRPT (RAC/RSM) assistance, as needed for Pg. 2. (N/A for Common Site)
( ) c. FAX to Group B. (EOF2 – FAX to Group D)
( ) d. IF fax transmission of the SSCL is incomplete, THEN CONTACT the State Agencies listed below, READ the data, AND DOCUMENT on SSCL, Pg. 2.

DEMA	Delaware Emergency Management Agency	302-659-2290
BNE	NJ Bureau of Nuclear Engineering	609-984-7700

( ) e. REPEAT Steps 6a - d approximately every half hour OR IMMEDIATELY for significant changes in Station status, until either Turnover or relief.

_____ CM2/TSC2/EOF2

7. OBTAIN a completed **NRC Data Sheet** from the Primary Communicator (CM1/TSC1/EOF1) and FAX form to Group B (EOF2 – FAX to Group D)

_____ CM2/TSC2/EOF2

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	4	
	K/A #	G2.4.18	
	Importance Rating	3.3	

Knowledge of the specific bases for EOPs.

Proposed Question: Common 73

EOP 102 PRIMARY CONTAINMENT CONTROL, contains the following retainment override.

IF while executing the following steps:	THEN:
All entry conditions have cleared	**EXIT** this procedure
Drwl sprays have been initiated	BEFORE Drwl press reaches **0 psig**, **TERMINATE** drwl sprays
Supp chamber sprays have been initiated	BEFORE suppression chamber press reaches **0 psig**, **TERMINATE** supp chamber sprays
SAG entry is required.	**EXIT** this procedure and **ENTER** SAG

Which one of the following statements describes the bases for terminating drywell spray before drywell pressure reaches 0 psig?

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

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

C. It ensures a drywell temperature below 212 degrees F, therefore there is NO need to continue drywell sprays.

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

Proposed Answer: D

Explanation (Optional):

IAW EOP 102 Bases for step PCC-1 - It prevents drawing a negative pressure in the

containment, which would open the vacuum breakers and draw air into the containment.

D. Correct.

A. Incorrect - Concern is de-inerting containment.
B. Incorrect - a negative pressure will open the SC to RB vacuum breakers and de-inert containment. Thermal stress is not a concern.
C. Incorrect – Concern is de-inerting containment.

Technical Reference(s) _____ (Attach if not previously provided)
EOP 102 Bases for step PCC-1

Proposed references to be provided to applicants during examination: None

Learning Objective: EO101PE008 (As available)

Question Source: Bank # 80632
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam 2003

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 X
_____

Comments:

IF while executing the following steps,	THEN
All entry conditions have cleared	**EXIT** this procedure

Discussion

If actions taken to control reactor building conditions are successful, steps in this EOP have performed their intended mitigation function and exit to normal procedures is appropriate. This step helps to promote consistent EOP implementation.

IF while executing the following steps,	THEN
Drwl sprays have been initiated	BEFORE Drwl Press reaches 0 psig, **TERMINATE** Drwl Sprays

Discussion

The operation of drywell sprays must be terminated by the time drywell pressure decreases to 0 psig to ensure that primary containment pressure is not reduced below atmospheric.

Terminating the sprays "*before...0 psig*" permits use of the sprays for fission product scrubbing at low pressures or if the containment has failed, yet still avoids negative containment pressures. Consistent with the definition of "before," the actual pressure value at which the sprays should be secured is event-specific:

- Reducing containment pressure below the scram setpoint will clear the scram logic and maximize the margin to containment pressure limits.
- If the containment has failed or if primary containment venting is anticipated, it may be advisable to continue spray operation at low pressures to scrub the containment atmosphere.
- Reducing primary containment pressure will also reduce the NPSH available for pumps drawing suction from the suppression pool. As stated in the discussion of Caution #2, NPSH limits should be observed if possible, but may be exceeded if warranted by event-specific conditions. If there is no need for continued spray operation, however, sprays may be terminated at higher pressures if NPSH limits are approached. The reference to Caution #6 emphasizes the relationship between primary containment pressure and available NPSH.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	2	
	K/A #	G2.2.13	
	Importance Rating	4.1	

Knowledge of tagging an clearance procedures

Proposed Question: Common 74

IAW the Safety Tagging Program procedures, which of the following statements is correct regarding Worker's Blocking Tags (WBTs)?

A. The Clearing Agent will place a label designating the Worker and Job Technician on the WBT.

B. A WBT may be used to isolate a high voltage energy source (>600 volts).

C. A Work Clearance Document (WCD) containing WBTs may also contain Yellow Permissive Tags (YPTs).

D. Two WBTs may be simultaneously installed on the same blocking point.

Proposed Answer: C

Explanation (Optional): SH.OP-AP.ZZ-0015, rev 20 Att. 3

C. CORRECT- A Work Control Document (WCD) containing WBTs may also contain Yellow Permissive Tags (YPTs). WCDs containing WBTs may contain other tag types such as RBTs and YPTs.

A. INCORRECT - A label designating the Worker and the Clearing Agent shall be placed on the WBT by the Worker.

B. INCORRECT - A WBT may not be used to isolate a high voltage energy source.

D. INCORRECT - The WBT shall not be installed on any blocking point that is already tagged with any safety tag except for a WCT.

Technical Reference(s) SH.OP-AP.ZZ-0015, rev 20 Att. 3 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: NA0015E004 (As available)

Question Source: Bank # 62253
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 X
_____

Comments:

#74 - NRC

ATTACHMENT 3
TAGGING RULES
(Page 4 of 6)

(continued) WORKER'S BLOCKING TAG (WBT)	• WBTs shall not be used if work scope changes from that which the WBT use was initially intended. • Work shall not be performed on the component to which the WBT is affixed. • WBTs shall not be used to isolate equipment from a high Voltage (600 volts and above) energy source. • WBTs shall not be placed on any blocking point that is tagged with any other Safety Tag except a White Caution Tag. • WCDs containing WBTs will only allow activation of one Job Supervisor and one Worker at a time; only one worker shall be designated to operate the equipment (or hold responsibility for the tag) at a time. ➢ The individual worker specified on WCD shall be the lead worker for the job (the lead worker may designate another worker directly involved with the job to operate the equipment). ➢ After the Clearing Agent has signed onto the request, the Worker shall place a label designating the Worker and the Clearing Agent on the WBT. ➢ When work is completed, components tagged with WBTs shall be returned to the tagged position as specified on the WCD. IF conditions warrant, THEN the SM/CRS/WCCS may grant permission to leave components in an alternate position. This permission shall be documented in the long text or the untagging text of the WCD. • A WCD containing WBTs can be turned over from one Clearing Agent/worker to another. This allows jobs to continue from one shift to another without the necessity of swapping tags. CORRECT • WCDs containing WBT(s) may contain other tag types such as red blocking tags and yellow permissive tags. • Electrical T&D (Substation Mechanics) personnel performing work on station switchyard equipment under the tagging jurisdiction of ESO may apply T&D WBT(s)provided tags are within tagged boundaries and approved by SM prior to application. • Electrical T&D (Relay Department) personnel may apply T&D WBT(s) if all of the following conditions are met: ➢ The circuits are used for relay protection (ex. breaker failure, generator protection). ➢ The SM/CRS/WCCS and/or ESO approve manipulation prior to hanging tag(s). ➢ Relay Department shall use an approved procedure that has specific steps to log all tag(s) applied and released.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	3	
	K/A #	G2.3.13	
	Importance Rating	3.4	

Knowledge of Radiological Safety procedures pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

Proposed Question: Common 75

The following conditions exist for a job to be performed on a system.

- The general area radiation levels are 10 mrem/hr in the room.
- The hot spot in the room is a pipe elbow that has a radiation level of 100 mrem/hr.
- The job will be performed near the hot spot area.

(Assumptions: ALL 4 cases below have the same transition time to and from destinations. All shielding placement and removal is at 100 mrem/hr. The hot spot with shielding in place is 10 mrem/hr)

Which one of the following methods would comply with ALARA procedural requirements for performance of the task?

A. The job is performed by using 2 operators for 3 hrs each on the job at the hot spot.

B The job is performed by 3 operators for 1 hr each on the job at the hot spot and a fourth operator reading instructions in the general room area for 1 hr.

C. The job is performed by 2 operators for 2 hrs each on the job at the hot spot and a third operator reading instructions in the general room area for 2 hrs.

D. Two Radiation Protection personnel hang and remove 1 tenth thickness of lead shielding on the hot spot in 1.5 hours for the job. The job is performed after the lead shielding is in place by using 2 operators for 3 hrs each on the job.

Proposed Answer: B

Explanation (Optional):

B. Correct: The job is performed by 3 operators for 1 hr each on the job at the hot spot and a fourth operator reading instructions in the general room area for 1 hr.(3 operators X 100 mrem/hr x 1hr) + (1 operators X 10 mrem/hr x 1hr) = 310 mrem/hr.

A. Incorrect: The job is performed by using 2 operators for 3 hrs each on the job at the hot spot. (2

operators X 100 mrem/hr x 3hrs) = 600 mrem/hr.

C. Incorrect: The job is performed by 2 operators for 2 hrs each on the job at the hot spot and a third Operator reading instructions in the general room area for 2 hrs. (2 operators X 100 mrem/hr x 2hr) + (1 operators X 10 mrem/hr x 2hrs) = 420 mrem/hr.

D. Incorrect: Two Radiation Protection personnel hang and remove 1 tenth thickness of lead shielding on the hot spot in 1.5 hours for the job. The job is performed after the lead shielding is in place by using 2 operators for 3 hrs each on the job. (2 rad techs X 100 mrem/hr x 1.5hrs) + (2 operators X 10 mrem/hr x 3hr) = 360 mrem/hr.

Technical Reference(s) RP-AA-400 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: _____ (As available)

Question Source: Bank # WTS Bank
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 X

Comments:

#75
NRC

Exelon Nuclear

ALARA PROGRAM

1. PURPOSE

1.1. This procedure establishes the requirements and responsibilities for the effective implementation of the ALARA Program. The objective of the ALARA program is to ensure that occupational radiation exposure, both individually and collectively, is maintained ALARA.

2. TERMS AND DEFINITIONS

2.1. **ALARA**: Acronym for "as low as reasonably achievable." ALARA means making every reasonable effort to maintain exposure to radiation as far below the dose limits, as defined in 10 CFR 20, as is practical consistent with the purpose for which the licensed activity is undertaken, taking into account the state of technology, the economics of improvements in relation to benefits to the public health and safety, and other societal and socioeconomic considerations, and in relation to utilization of nuclear energy and licensed material in the public interest.

2.2. **ALARA Plan**: A documented job evaluation. The evaluation will consider the radiological conditions expected during each phase of the job and the methods and controls to minimize contamination and collective radiation exposure (person-rem). The term "ALARA Plan" is equivalent to "ALARA Action Review" and "ALARA Review."

2.3. **Station ALARA Committee (SAC)**: Station committee responsible for the overall coordination of the Station ALARA Program and for advising site management in matters relating to ALARA and other pertinent Radiation Protection programs.

3. RESPONSIBILITIES

NOTE: Sites may utilize position titles that are **not** identical to those outlined in this procedure. In those cases, the responsibilities should be performed by the site individual with responsibilities closest to those described in this procedure.

3.1. Plant Manager/Site Vice President

3.1.1. **ACT** as the Chairperson for the Station ALARA Committee under normal circumstances. If the Plant Manager or Site Vice President is **not** available to fulfill this duty, a senior member of station management staff may be appointed as an alternate to chair the meeting.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295021 AA2.01	
	Importance Rating		3.6

Ability to determine and/or interpret the following as they apply to LOSS OF SHUTDOWN COOLING: Reactor water heatup/cooldown rate.

Proposed Question: SRO 76

The plant is shutting down for a refuel outage. Shutdown Cooling has been in service for 1 hour.

T= 12:00 - RPV temperature is 149 degrees F.

Then, a complete loss of Shutdown Cooling occurs.
After 20 minutes, the operators determine that RPV temperature is rising at 16 degrees every 10 minutes.

T= 12:20 - RPV temperature is 186 degrees F.

Which one of the following describes how the heatup, if it continues at the rate stated above, will affect the plant Operational Condition and Technical Specification (TS) heatup limits?

A. After T=12:30, a mode change will occur. At T=13:00, the TS heatup rate limit will be exceeded.

B. Before T=12:30, a mode change would occur. At T=13:00, the TS heatup rate limit will be exceeded.

C. After T=12:30, a mode change will occur. At T=13:00, the TS heatup rate limit will NOT be exceeded.

D. Before T=12:30, a mode change would occur. At T=13:00, the TS heatup rate limit will NOT be exceeded.

Proposed Answer: B

Explanation (Optional): Mode change occurs at >200 dgrees F. per TS definitions. The TS heatup limit is 100 degrees in a one hour period. Although the rate is >100 degrees per hour the limit is not exceeded until the one hour time period has been met (1200 – 1300)

TS 3.4.6.1.a. & TS definitions of operational conditions

B. Correct. Mode change has already occurred. (202 F @12:30) Limit has been exceeded.

101 degrees F at 1300. (149 @ 12:00 to 250 @ 13:00)

A. Incorrect. Mode change has already occurred. (202 F @12:30)
C. Incorrect. The TS limit has been exceeded. 101 degrees @13:00
D. Incorrect. A mode change would occur prior to 1230 (202 F @12:30) The TS limit has been exceeded. 101degrees @13:00

Technical Reference(s) TS 3.4.6.1.a. & TS definitions (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: RXVESSEE007 (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ____
55.43 2

Comments:

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.6.1 The reactor coolant system temperature and pressure shall be limited in accordance with the limit lines shown on Figure 3.4.6.1-1 (hydrostatic or leak testing), and Figure 3.4.6.1-2 (heatup by non-nuclear means, cooldown following a nuclear shutdown and low power PHYSICS TESTS), and Figure 3.4.6.1-3 (operations with a critical core other than low power PHYSICS TESTS), with:

a. A maximum heatup of 100°F in any one hour period,

b. A maximum cooldown of 100°F in any one hour period,

c. A maximum temperature change of less than or equal to 20°F in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves, and

d. The reactor vessel flange and head flange metal temperature shall be maintained greater than or equal to 79°F when reactor vessel head bolting studs are under tension.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system; determine that the reactor coolant system remains acceptable for continued operations or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1.1 During system heatup, cooldown and inservice leak and hydrostatic testing operations, the reactor coolant system temperature and pressure shall be determined to be within the above required heatup and cooldown limits and to the right of the limit lines of Figures 3.4.6.1-1, 3.4.6.1-2, and 3.4.6.1-3 as applicable, at least once per 30 minutes.

#76
NRC

TABLE 1.2

OPERATIONAL CONDITIONS

CONDITION	MODE SWITCH POSITION	AVERAGE REACTOR COOLANT TEMPERATURE
1. POWER OPERATION	Run	Any temperature
2. STARTUP	Startup/Hot Standby	Any temperature
3. HOT SHUTDOWN	Shutdown[#,***]	> 200°F
4. COLD SHUTDOWN	Shutdown[#,##,***]	≤ 200°F[+]
5. REFUELING[*]	Shutdown or Refuel[**,#]	≤ 140°F

[#]The reactor mode switch may be placed in the Run, Startup/Hot Standby, or Refuel position to test the switch interlock functions and related instrumentation provided that the control rods are verified to remain fully inserted by a second licensed operator or other technically qualified member of the unit technical staff. If the reactor mode switch is placed in the Refuel position, the one-rod-out interlock shall be OPERABLE.

[##]The reactor mode switch may be placed in the Refuel position while a single control rod drive is being removed from the reactor pressure vessel per Specification 3.9.10.1.

[*]Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

[**]See Special Test Exceptions 3.10.1 and 3.10.3.

[***]The reactor mode switch may be placed in the Refuel position while a single control rod is being recoupled or withdrawn provided that the one-rod-out interlock is OPERABLE.

[+]See Special Test Exception 3.10.8.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295023 AA2.05	
	Importance Rating		4.6

Ability to determine and/or interpret the following as they apply to REFUELING ACCIDENTS : Entry conditions of Emergency plan

Proposed Question: SRO 77

Given the following:

- The plant is in OPCON 5.
- Core offload is in progress.
- A spent fuel bundle is full up on the main hoist over the core.
- The refuel bridge spotter notices the fuel bundle has become unlatched and has fallen into the vessel.

A short time later, the following Refuel Floor Rad Monitors are in HIGH alarm:

- Spent Fuel Pool ARM
- Refuel Floor Exhaust Channels A, B, C
- General Area radiation surveys have NOT been performed
- All other plant systems are operating as designed

Which one of the following describes actions required IAW AB-CONT-0005 "Irradiated Fuel Damage" and the Emergency Plan?

Suspend the handling of Irradiated Fuel/Components...

A. re-establish Secondary Containment and declare an Alert.

B. re-establish Secondary Containment and perform a 4 hour report to the NRC.

C. evacuate all unnecessary personnel from the refuel floor and declare an Alert.

D. evacuate all unnecessary personnel from the refuel floor and perform a 4 hour report to the NRC.

Proposed Answer: C

Explanation (Optional): ECG Section 6.4.2.b. – with the alarms noted in stem an Alert declaration is warranted.

C. Correct.

A. Incorrect. Secondary Containment was not lost based on stem conditions

B. Incorrect. Secondary Containment was not lost based on stem conditions
D. Incorrect. May be correct if Alert is believed to be incorrect. Based on voluntary/courtesy report IAW ECG Section 11.10.2

Technical Reference(s) AB-CONT-0005 (Attach if not previously provided)
ECG Section 6.4.2.b

Proposed references to be provided to applicants during examination: ECG – not the attachments

Learning Objective: ABCNT5E007 (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New x

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 ____
55.43 1

Comments:

#77

NRC

6.0 Radiological Releases/Occurrences

6.4 Irradiated Fuel Event

Initiating Condition

OPCON

EAL #

EMERGENCY ACTION LEVELS

Action Required

Major Damage to Irradiated Fuel

All

6.4.2.a

IF

Major Damage to Irradiated Fuel has occurred

AND

Valid High Alarm received from ANY one of the following RMS channels:

- Refuel Floor Exhaust Channel A (9RX627)
- Refuel Floor Exhaust Channel B (9RX628)
- Refuel Floor Exhaust Channel C (9RX629)

Events that have or may result in uncovering Irradiated Fuel outside the Reactor Vessel

All

6.4.2.b

IF

Unplanned rise on ANY one of the following Area Rad Monitors or by general area rad survey indicates **≥ 2000 mRem/hr**:

- Spent Fuel Storage Pool Area (9RX707)
- New Fuel Criticality Storage Channel A (9RX612)
- New Fuel Criticality Storage Channel B (9RX613)

All

6.4.2.c

IF

Visual observation of Irradiated Fuel uncovered

THEN

Refer to Attachment 2
ALERT

#77
NRC

SUBSEQUENT OPERATOR ACTIONS

CONDITION	ACTION
A. Irradiated Fuel Damage. Date/Time: _____	A.1 **PERFORM** the following: ☐ • **ENSURE** either RBVS OR FRVS is in service • **PERFORM** ONE of the following: ☐ 1. **ENSURE** Secondary Containment Integrity is in effect. ☐ 2. WITHIN 30 minutes, **IMPLEMENT** SH.OP-AP.ZZ-0108 Attachment 10 to seal secondary containment breaches. ☐ • **DIRECT** Radiation Protection to take air samples AND control access to the Reactor Bldg and Refuel floor. ☐ • **EVACUATE** all UNNECESSARY personnel from the affected area.
B. Damaged irradiated fuel is attached to the fuel handling grapple. Date/Time: _____	☐ B.1 IF Radiation levels permit, **PLACE** the damaged fuel in the Spent Fuel Pool.
C. Damaged irradiated fuel is in the Spent Fuel Pool, Reactor Cavity OR Cask Loading Pit. Date/Time: _____	☐ C.1 **DIRECT** Radiation Protection to survey the affected Spent Fuel Pool Cooling AND Residual Heat Removal systems piping as appropriate.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295019 AA2.02	
	Importance Rating		3.7

Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR : Status of safety-related instrument air system loads

Proposed Question: SRO 78

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.

The operators insert a manual scram.

What will the Reactor Pressure Vessel (RPV) level control and pressure control strategy be for the loss of Instrument Air?

A. IAW EOP-101 "RPV Control", SRVs for pressure control, HPCI/RCIC for level control.

B. IAW EOP-101 "RPV Control", SRVs for pressure control, Maximize CRD for level control.

C. IAW AB.ZZ-0000 "Reactor SCRAM", Bypass Valves for pressure control, HPCI/RCIC for level control.

D. IAW AB.ZZ-0000 "Reactor SCRAM", Bypass valves for pressure control, Maximize CRD for level control.

Proposed Answer: A

Explanation (Optional):

A 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.

B INCORRECT - CRD flow control valves fail closed on a loss of air
C INCORRECT - Condenser is NOT available for pressure control
D INCORRECT - Condenser is NOT available and NO condensate line up is possible due to level control valves fail closed on a loss of air.

Technical Reference(s): EOP-101 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: INSAIRE016 (As available)

Question Source: Bank # INPO 25895
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam 2005

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 _____
55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295025 G2.1.23	
	Importance Rating		4.4

(K&A Statement) Conduct of Operations: Ability to perform specific system and integrated plant procedures during all modes of operation. (High Reactor Pressure)

Proposed Question: SRO 79

The plant is operating at 22% power when an EHC failure raises RPV pressure to 1052 psig. The following events occur:

- Main Turbine trips
- Mode Switch locked in the Shut Down position
- Scram Air header pressure lowers to 72 psig
- NO control rod motion

Which one of the following EOP entries correctly describes the required operator action(s) and the basis for the action(s)?

A. Enter EOP-101A ATWS-RPV Control because the SDV is full.

B. Enter EOP-101A ATWS-RPV Control because manual ARI is required.

C. Enter EOP-101 Reactor Pressure Vessel Control because a scram reset is required.

D. Enter EOP-101 Reactor Pressure Vessel Control because the Main Turbine is tripped.

Proposed Answer: B.

Explanation (Optional):

B. Correct – EOP-101 is initially entered on RPV press >1037 psig, but with the Mode Switch in SD and all rods not in the SRO must transition to EOP-101A, where the verification of ARI is the next step (the Turbine has already tripped).

A. Incorrect – EOP-101A is entered because the rods did not fully insert, additionally scram air header pressure has not lowered therefore the SDV Vents and Drains are open, there is no confirmation in the question stem that the SDV is full.

C. Incorrect – A scram reset is not required at this time because scram air pressure has not lowered (failure to scram), additionally EOP-101 is exited and EOP-101A is entered.

D. Incorrect – The turbine tripping is not an entry condition into EOP-101, additionally EOP-101 is exited and EOP-101A is entered.

Technical Reference(s): EOP-101A (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: EA101AE002 (As available)

Question Source: Bank # ID: Q56465

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295026 G2.1.7	
	Importance Rating		4.7

Conduct of Operations: Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation. (Suppression Pool High Water Temperature)

Proposed Question: SRO 80

While operating at 60% Reactor power, a Reactor Scram on low reactor water level occurs but all rods remain at their pre-trip conditions.

Plant conditions thirty minutes after the transient start are:

- SLC tank level 2600 gal
- Rx power <4 %
- RPV pressure 900 psig being controlled using SRVs
- RPV level Intentionally lowered to -135 inches and steady
- Suppression pool level 79 inches
- Suppression pool temp 185°F and rising at 1°F/5 min
- Drywell pressure 4.5 psig
- Main steam tunnel temperature 170°F and rising at 1°F/2 min

Which one of the following is required for the conditions above?

A. Maintain RPV water level between +54" and -185".

B. Bypass interlocks to open the MSIV's and reduce RPV pressure.

C. Reduce RPV pressure to prevent exceeding the Heat Capacity Temperature Limit curve.

D. Emergency Depressurize to prevent exceeding the Pressure Suppression Pressure curve.

Proposed Answer: C

Explanation (Optional):

C. Correct - HCTL limit is being approached and will reach Action Required line within 15 minutes. RPV pressure must be reduced to move away from limit.

A. Incorrect - Wrong level band. Upper end of band limited to - 50 inches with an ATWS.

B. Incorrect - Incorrect action based on evidence of leak in the Main Steam Tunnel.
D. Incorrect - Action to be taken if RPV pressure cannot be lowered.

Technical Reference(s) EOP-102, 101A (Attach if not previously provided)

Proposed references to be provided to applicants during examination: SCP-L, SPT-P

Learning Objective: EO101AE008 (As available)

Question Source: Bank # 55997
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 _____
55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295031 G2.4.3	
	Importance Rating		3.9

(K&A Statement) Emergency Procedures / Plan: Ability to identify post-accident instrumentation. (Reactor Low Water Level).

Proposed Question: SRO 81

A plant event has occurred. RPV level is +20" and lowering. You are informed by I & C that the following level instruments are inoperable.

- Fuel Zone Range Level Recorder LR-R615
- Upset Range Level Recorder LR-R608

Which one of the following sections of Technical Specifications must be entered for these inoperable instruments once plant conditions stabilize?

A. 3.3.7.5, Accident Monitoring Instrumentation ONLY

B. 3.3.4, Recirculation Pump Trip Actuation Instrumentation ONLY

C. 3.2.1, Isolation Actuation Instrumentation and 3.3.3 Emergency Core Cooling Actuation Instrumentation ONLY

D. 3.3.7.5 Accident Monitoring Instrumentation and 3.3.3 Emergency Core Cooling Actuation Instrumentation ONLY.

Proposed Answer: A.

Explanation (Optional):

A. Correct – The fuel zone instruments provide post accident indication only

B. Incorrect - These TS instruments apply to normal and wide range instruments.
C. Incorrect - These TS instruments apply to normal and wide range instruments.
D. Incorrect - These TS instruments apply to normal and wide range instruments.

Technical Reference(s): T.S. 3.3.7.5 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: RXINSTE021 (As available)

Question Source: Bank # _____

Modified Bank # _____ (Note changes or attach parent)

New X

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge _____

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 _____

55.43 2

Comments:

TABLE 3.3.7.5-1

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. Reactor Vessel Pressure	2	1	1,2,3	80
2. Reactor Vessel Water level	2	1	1,2,3	80
3. Suppression Chamber Water level	2	1	1,2,3	80
4. Suppression Chamber Water Temperature*	2	1	1,2,3	80
5. Suppression Chamber Pressure	2	1	1,2,3	80
6. Drywell Pressure	2	1	1,2,3	80
7. Drywell Air Temperature	2	1	1,2,3	80
8. Deleted				
9. Safety/Relief Valve Position Indicators	2/valve**	1/valve**	1,2,3	80
10. Drywell Atmosphere Post-Accident Radiation Monitor	2	1	1,2,3	80
11. North Plant Vent Radiation Monitor #	1	1	1,2,3	81
12. South Plant Vent Radiation Monitor #	1	1	1,2,3	81
13. FRVS Vent Radiation Monitor #	1	1	1,2,3	81
14. Primary Containment Isolation Valve Position Indication ##	2/valve	1/valve	1,2,3	82

* Average bulk pool temperature.

** Acoustic monitoring and tail pipe temperature.

\# High range noble gas monitors.

One channel consists of the open limit switch, and the other channel consists of the closed limit switch.

#81
NRC

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.5 The accident monitoring instrumentation channels shown in Table 3.3.7.5-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3.7.5-1.

ACTION:

With one or more accident monitoring instrumentation channels inoperable, take the ACTION required by Table 3.3.7.5-1.

SURVEILLANCE REQUIREMENTS

4.3.7.5 Each of the above required accident monitoring instrumentation channels shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.7.5-1.

NOTE: I & C training has an optical isolator assembly that may be available to demonstrate the following points.

Figure 14

7. Optical Isolator (GE Systems)
   a. Optical isolators are used to transfer signals between 1E and non-1E electronic systems to maintain electrical isolation. This prevents an electrical fault in a non-1E system from propagating to a 1E system. There are two general types of optical isolators: analog and digital.
   b. The optical isolator (designated AT on GE elementary drawings) has two circuits. The first is an infrared light emitting diode with its own power supply. The second circuit has a phototransistor and its own power supply. The digital and analog isolators are functionally equivalent, although the analog isolator converts the data input into a serialized digital format for transmission via the LED.
   c. The infrared light output of the input isolator travels through a 'light pipe'—a ¼ inch cylindrical quartz about one inch in length—to the phototransistor in the output isolator module. The digital and analog output isolators provide an output equivalent to the input. The digital outputs are on-off and the analog isolator converts the serialized data into an analog output.
   d. Both the input and output isolator cards require power. The digital isolators are typically powered by the interfacing system. The analog isolators require an external source of power.
8. Post Accident Monitoring System (PAMS)

Objective 14a

   a. The PAM System utilizes existing reactor vessel level and pressure instrumentation for post accident monitoring.
      1) Six level and two pressure transmitters supply signals to indicators and recorders in the control room on 10C650A and C.
         a) Two channels each (A & B) of fuel zone level, wide range level, shutdown range level and wide range pressure
            (1) Fuel zone indicator LI-R610 and recorder LR-R615
            (2) Wide range recorders LR-R623A and B
            (3) Shutdown range recorders LR-3622A and B
            (4) Wide range reactor pressure recorders PR-3684A and B

FSAR Sections 1.8.1.97 and 7.5.1.3.4
Tech Spec 3.3.7.5
HC.OP-ST.SH-0001

b) Although other level and pressure instrumentation channels are provided, only the above instruments meet the requirements of Reg Guide 1.97 (FSAR Table 7.5-1) and Technical Specification 3.3.7.5, Accident Monitoring Instrumentation.

NOTE: Both the indicating and recording functions are required for these instruments to be considered operable. They are channel-checked IAW HC.OP-ST.SH-0001.

Figure 15
Objective 14b

b. The PAMS instruments utilize the 'A' and 'B' channel reference legs. These reference legs have a short elevation drop within the drywell, making them less susceptible to the effects of elevated drywell temperature.

c. LR-R623A and B have two modes for trending parameters:

1) Normal Operation – Slow (typically 20 mm/hour)

2) Accident Conditions – High-Speed (1200 mm/hour)

3) They automatically shift to the high-speed trend to provide increased resolution of the recorded traces of wide range reactor level and pressure.

a) The recorders will shift to the fast-speed trend when level drops to 12.5" or pressure reaches 1037 psig.

b) A 'HIGH SPEED CHART DRIVE RESET' pushbutton is provided to return the chart drives to the low-speed trend after the initiating conditions have cleared.

Figure 8
Objectives 7c, 15a, 15b, 15c, 15d, 16

IV. INSTRUMENTATION AND CONTROL

A. Instrumentation

1. Main Control Room

Table 1

a. Reactor Vessel Level

See Table 1

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295028 EA2.03	
	Importance Rating		3.9

Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE : Reactor water level

Proposed Question: SRO 82

Given the following conditions:

- A Large Break LOCA has occurred in the Drywell concurrent with a LOP
- Only "C" EDG is running
- All control rods are fully inserted
- Wide Range RPV level indicator LR-623A is reading +20 inches
- Wide Range RPV level indicator LR-623B is reading -55 inches
- Drywell pressure is 29 psig and rising
- Drywell temperature is 300 F and rising
- Reactor pressure is 25 psig and steady
- Suppression Pool Level is 80 inches and rising
- Suppression Chamber pressure is 30 psig and rising
- "C" RHR Pump has been injecting LPCI flow for 3 minutes

Based on the above conditions, which one of the following actions is required?

A. Continue LPCI injection and enter EOP-206 "RPV Flooding".

B. Continue LPCI injection and continue in EOP-101 "RPV Control" in all control legs.

C. Stop LPCI injection, Emergency Depressurize IAW EOP-202, and then resume LPCI injection.

D. Continue LPCI injection, Emergency Depressurize IAW EOP-202, and then enter EOP-206 "RPV Flooding".

Proposed Answer: A

Explanation (Optional): with High drywell temps and low RPV pressure, per EOP caution 1 level is unreliable. Therefore it is not known and RPV flooding is required

A. Correct. Level is unknown due to unreliability of level instruments with high drywell temperature. RPV flooding is required. LPCI injection would continue

B. Incorrect. RPV flooding is required. Pressure Control Leg of 101 is exited when 206 is required.
C. Incorrect. Would not stop LPCI injection
D. Incorrect. ED already performed, RPV flooding required

Technical Reference(s) EOP-102 retainment step (Attach if not previously provided)
EOP caution 1

Proposed references to be provided to applicants during examination: Caution 1

Learning Objective: EOP206E008 (As available)

Question Source: Bank # 56161
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 _____
55.43 5

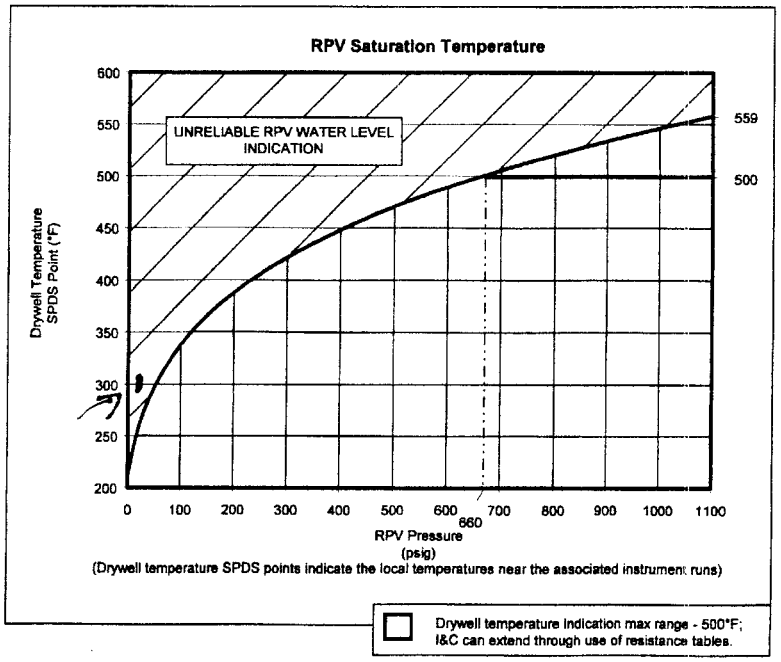
Comments:

#82
NRC

EOP CAUTION 1

A. Under conditions of elevated drywell temperature, channels A and B of the wide, narrow and upset RPV water level instruments provide the most reliable indications.

B. If any drywell temperature SPDS point exceeds the RPV saturation temperature, the associated level instruments may be unreliable.



Drywell Temperature SPDS Points	Associated RPV Water Level Instruments
A2266	• LR-R623A-B21 Wide Range A (-150 to +60 in.) • LR-3683A Narrow Range A (0 to +60 in.) • LR-3683B Narrow Range B (0 to +60 in.) • LR-P615-B21 Fuel Zone A (-311 to -111 in.)
A2274	• LR-R623A-B21 Wide Range A (-150 to +60 in.) • LR-3683A Narrow Range A (0 to +60 in.) • LR-P615-B21 Fuel Zone A (-311 to -111 in.)
A2277	• LR-R623B-B21 Wide Range B (-150 to +60 in.) • LR-3683B Narrow Range B (0 to +60 in.) • LI-R610-B21 Fuel Zone B (-311 to -111 in.)
A2280	• LR-P615-B21 Fuel Zone A (-311 to -111 in.)
A2281	• LI-R610-B21 Fuel Zone B (-311 to -111 in.)
A2283	• LR-R623A-B21 Wide Range A (-150 to +60 in.) • LR-R623B-B21 Wide Range B (-150 to +60 in.) • LR-3683A Narrow Range A (0 to +60 in.) • LR-P615-B21 Fuel Zone A (-311 to -111 in.)
A2284	• LR-R623B-B21 Wide Range B (-150 to +60 in.) • LR-3683B Narrow Range B (0 to +60 in.) • LI-R610-B21 Fuel Zone B (-311 to -111 in.)
A2287	• LI-R610-B21 Fuel Zone B (-311 to -111 in.)

EOP CAUTION 1	
HOPE CREEK GENERATING STATION	
REV. 01	REV. DATE 5/21/99

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	295033 EA2.01	
	Importance Rating		3.9

EA2.01 - Ability to determine and/or interpret the following as they apply to HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS: Area Radiation levels

Proposed Question: SRO 83

The plant is operating at rated power.

A Backwash of Clean-Up Filter Demineralizer AF-203 has just been completed. Transfer of the RWCU Backwash Receiver Tank to Radwaste is in progress. A catastrophic failure of Backwash Transfer pump 1AP-214 suction line causes a spill into the Reactor Building. All attempts to isolate the leak have been unsuccessful.

Reactor Building Area Radiation conditions are as follows:

Reactor Building Area Radiation Monitor	**Beginning of Shift**	**Current Conditions**
9RX706 Reactor Cleanup Demin. Sys. Equipment	2 mr/hr	2400 mr/hr - In Alarm
9RX723 Outside Reactor Bldg. Sample Station	3 mr/hr	1100 mr/hr - In Alarm
9RX708 Sample Station	3.5 mr/hr	4500 mr/hr - In Alarm
Other Reactor Building Area Radiation Monitors	2 to 5 mr/hr	3 to 7mr/hr - NOT In Alarm

Which one of the following is the required action?

A. Commence a normal reactor shutdown to cold shutdown IAW IO.ZZ-0004.

B. Continue reactor operation and attempt to stop the tank drain line leakage IAW SO.BG-0001.

C. IAW EOP-0103/4, Runback Recirc, Initiate a Manual Scram and Emergency Depressurize the RPV.

D. IAW EOP-0103/4, Runback Recirc and Initiate a Manual Scram. Emergency Depressurization is NOT required.

Proposed Answer: A

Explanation (Optional): RWCU Backwash Receiving tank is not a primary System, with 2 areas > Max Safe Operating Limit, Plant shutdown and cooldown per IO-004 is applicable.

A. Correct.

B. Incorrect. Per EOP 103/4 and since the leak is not from a primary system, plant shutdown and cooldown applies
C. Incorrect. Only applicable for a primary system leak
D. Incorrect. Only applicable for a primary system leak

Technical Reference(s) EOP 103/4 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: EOP 103 no entrys no retainment steps

Learning Objective: EOP103E006 (As available)

Question Source: Bank # 54264
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 _____
55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	295017 G2.4.21	
	Importance Rating		4.6

Emergency Procedures / Plan: Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.

Proposed Question: SRO 84

Given the following conditions:

- A High Off-site release event is in progress
- Various RMS points are indicating elevated values but less than the RM-11 ALERT values
- The Iodine release rate must be determined

Which of the following describes the procedure to be entered and the method used to determine the total release rate?

A. Enter AB.CONT-0004 "Radioactive Gaseous Release". Use RMS values for all inputs to the release rate formula.

B. Enter AB.CONT-0004 "Radioactive Gaseous Release". Use RMS values for FRVS and HTV. An Iodine sample must be taken for the NPV and SPV.

C. Enter OP-AR.SP-0001 "Radiation Monitoring System Alarm Response". Samples must be taken for all inputs to release rate formula.

D. Enter OP-AR.SP-0001 "Radiation Monitoring System Alarm Response". Use RMS values for the NPV, SPV, and FRVS. An Iodine sample must be taken for the HTV.

Proposed Answer: A

Explanation (Optional): IAW AB.CONT-0004 , OP-AR.SP-0001 may be entered but the parameters are specified in the abnormal

A. CORRECT - The FRVS, NPV, SPV & HTV sample skids all have Iodine Monitors that can be used for the calculation.

B. INCORRECT - The NPV and SPV have iodine monitors and sampling is not specified in

the procedure.

C. INCORRECT - The NPV, SPV & HTV sample skids all have Iodine Monitors that can be used for the calculation. Samples are not required for those values.

D. INCORRECT - The FRVS sample skid does not have an Iodine Monitor so a sample must be taken. The HTV sample skid has an Iodine Monitor.

Technical Reference(s) AB.CONT-0004 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: ABCNTE005 (As available)

Question Source: Bank # 64416
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 _____
55.43 4

Comments:

ADDITIONAL INFORMATION:

Procedures:

- HC.RP-AR.SP-0001(Q), Radiation Monitoring System Alarm Response
- HC.OP-SO.GU-0001(Q), Filtration Recirculation and Ventilation System Operation

TABLE "A"

(Iodine release rates can be calculated by substituting Iodine values for Noble Gas)

CALCULATE the total Noble Gas release rate from Hope Creek Generating Station by adding all gaseous effluent channels:

_____ +	_____ +	_____ +	_____ =	_____
µCi/sec SPV (9RX580)	µCi/sec NPV (9RX590)	µCi/sec FRVS (9RX680)	µCi/sec HTV (9RX518)	µCi/sec Total

IF the effluent (µCi/sec) channel on the RM-11 is NOT operating for a specific plant vent, THEN **CALCULATE** the Noble Gas release rate for that vent using the following:

_____ *	_____ *	472 =	_____
µCi/cc (n.g.)	Plant Vent Exh Flow in cfm		µCi/sec (n.g.)

Where:

µCi/cc (n.g.)	The concentration of Noble Gas obtained from the RM-11 (the operable channel will be highlighted in GREEN) OR from an actual sample of the plant vent
472	The conversion factor in units of cc/sec/cfm
µCi/sec (n.g.)	The calculated release rate from the specified plant vent (Noble Gas)

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	295015 G2.4.30	
	Importance Rating		4.1

Emergency Procedures / Plan: Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as State, the NRC, or the transmission system operator (incomplete scram)

Proposed Question: SRO 85

The plant was operating at 100% when an existing leak in the drywell worsened.

The operators scrammed the plant prior to reaching the High Drywell scram setpoint.

The following conditions now exist:

- Two peripheral control rods are at position 48
- ALL APRM downscale lights are lit
- Drywell Pressure is 1.80 psig and slowly rising
- RPV level is +20 inches and slowly rising
- RPV pressure is 910 psig

Which one of the following describes the NRC notification required?

A. 50.72 - 1 Hour Report

B. 50.72 - 4 Hour Report

C. Alert

D. SAE

Proposed Answer: C

Explanation (Optional): ECG Section 5.1.2.b manually initiated scram unsuccessful = Alert

C. Correct

A. Incorrect. Does not meet ECG Section 11.1 for 1 hour reports

B. Incorrect. Correct for actuation of RPS ONLY - ECG 11.3.2

D. Incorrect. Correct only if power remained about 4%

Technical Reference(s) ECG section 5.1.2.b (Attach if not previously provided)

Proposed references to be provided to applicants during examination: ECG, no attachments

Learning Objective: _____ (As available)

Question Source: Bank # _____
Modified Bank # _____ (Note changes or attach parent)
New X

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 _____
55.43 1

Comments:

USER RESPONSIBLE FOR VERIFYING REVISION, STATUS AND CHANGES
PRINTED 20060221

5.0 Failure to Scram

5.1 ATWS

Initiating Condition	Failure of the Reactor Protection System (RPS) to Successfully Complete a Reactor Scram (Automatic or Manual)	Failure of the Reactor Protection System (RPS) to Successfully Complete a Reactor Scram (Automatic and Manual) and Reactor Power is above 4%	Failure of the Reactor Protection System (RPS) to Successfully Complete a Reactor Scram (Automatic and Manual) and there is indication of an Extreme Challenge to the Ability to Cool the Core
OPCON	1, 2	1, 2	1, 2	1, 2
EAL #	5.1.2.a	5.1.2.b	5.1.3	5.1.4
EMERGENCY ACTION LEVELS	IF An Automatic Reactor Scram Condition exists AND An Automatic Reactor Scram (RPS) IS NOT successful	IF ANY Manually Initiated Reactor Scram (RPS) from the Control Room IS NOT successful	ALL Reactor Scram attempts from the Control Room (RPS and ARI) DID NOT REDUCE and MAINTAIN Reactor Power to ≤ 4%	EITHER one of the following: • Reactor Water Level CANNOT BE RESTORED and MAINTAINED above -185" • The combination of Suppression Pool Temperature and RPV Pressure CANNOT BE MAINTAINED below the HCTL Curve
	THEN		AND THEN	AND THEN
Action Required	Refer to Attachment 2 ALERT		Refer to Attachment 3 SITE AREA EMERGENCY	Refer to Attachment 4 GENERAL EMERGENCY

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	206000 2.2.37	
	Importance Rating		4.6

Equipment Control: Ability to determine operability and/or availability of safety related |equipment. (HPCI)

Proposed Question: SRO 86

Given the following:

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

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

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

B. HPCI is "operable but degraded" because it has lost its testing capacity.

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

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

Proposed Answer: C

Explanation (Optional): TS 3.5.1

C. Correct – HPCI must be in AUTO with a setpoint of 5600 gpm and capable of rated flow and discharge pressure.

C. Correct.

A. Incorrect – It must develop rated flow in AUTO

B. Incorrect – The case could be made if flow in AUTO remained stationary at 5600 gpm.

D. Incorrect - operable but non-conforming does not apply with flow at 4000 gpm in AUTO.

Technical Reference(s) TS 3.5.1 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: HPCI00E018 (As available)

Question Source: Bank # _____
Modified Bank # 55949 (Note changes or attach parent)
New _____

Question History: Last NRC Exam 2003

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 _____
55.43 2

Comments:

H86
NRC

4.5.1 The emergency core cooling systems shall be demonstrated OPERABLE by:

a. At least once per 31 days:

1. For the core spray system, the LPCI system, and the HPCI system:

a) Verifying by venting at the high point vents that the system piping from the pump discharge valve to the system isolation valve is filled with water.

b) Verifying that each valve, manual, power operated or automatic, in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct* position.

c) Verify the RHR System cross tie valves on the discharge side of the pumps are closed and power, if any, is removed from the valve operators.

2. For the HPCI system, verifying that the HPCI pump flow controller is in the correct position.

b. Verifying that, when tested pursuant to Specification 4.0.5:

1. The two core spray system pumps in each subsystem together develop a flow of at least 6150 gpm against a test line pressure corresponding to a reactor vessel pressure of ≥105 psi above suppression pool pressure.

2. Each LPCI pump in each subsystem develops a flow of at least 10,000 gpm against a test line pressure corresponding to a reactor vessel to primary containment differential pressure of ≥20 psid.

3. The HPCI pump develops a flow of at least 5600 gpm against a test line pressure corresponding to a reactor vessel pressure of 1000 psig when steam is being supplied to the turbine at 1000, +20, -80 psig.**

c. At least once per 18 months:

1. For the core spray system, the LPCI system, and the HPCI system, performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence and verifying that each automatic valve in the flow path actuates to its correct position. Actual injection of coolant into the reactor vessel may be excluded from this test.

---

*Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in position for another mode of operation.

**The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	209001 G2.2.40	
	Importance Rating		4.7

Equipment Control: - Ability to apply Technical Specifications for a system.

Proposed Question: SRO 87

The plant is operating at rated power. The following alarm is then received:

- B3-C1 "CORE SPRAY LOOP A TROUBLE"

I&C Technicians report that 1-BE-PT-N054A, Core Spray Loop A Header Pressure transmitter is failed.

Which one of the following describes the Technical Specification action time(s) of this failure?

A. Declare Core Spray Loop A inoperable within 24 hours.

B. Declare Core Spray Loop A inoperable within seven days.

C. Restore the transmitter to operable status within thirty days and verify Core Spray Loop A pressure is less than 475 psig every 24 hours.

D. Restore the transmitter to operable status within seven days or verify Core Spray Loop A pressure is less than 475 psig every 12 hours.

Proposed Answer: D

Explanation (Optional):

D Correct: Restore to operable status within seven days or verify Core Spray Loop A pressure less than 475 psig every 12 hours for thirty days. T.S. 3.5.1, Action f. including *. Then T.S. 3.4.3.2, Action d. The stated transmitter feeds both Hi-Lo pressure interface alarm and Keepfill low pressure alarm.

A. Incorrect. Declare CS loop A inoperable within 24 hours. Not required if pressure verified every 12 hours.

B. Incorrect: Declare CS loop A inoperable within 7 days. 30 days to restore provided

pressure verified every 12 hours.

C. Incorrect: Restore to operable within 30 days or verify pressure less than 475 every 24 hours. Verify every 12 hours.

Technical Reference(s) TS3.5.1. & 3.4.3.2 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: TS 3.5.1 & 3.4.3.2
M-52 Sht.1

Learning Objective: CSSYS0E014 (As available)

Question Source: Bank # 55139
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 _____
55.43 2

Comments:

#87

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ECCS - OPERATING

LIMITING CONDITION FOR OPERATION

3.5.1 The emergency core cooling systems shall be OPERABLE with:

a. The core spray system (CSS) consisting of two subsystems with each subsystem comprised of:

   1. Two OPERABLE core spray pumps, and

   2. An OPERABLE flow path capable of taking suction from the suppression chamber and transferring the water through the spray sparger to the reactor vessel.

b. The low pressure coolant injection (LPCI) system of the residual heat removal system consisting of four subsystems with each subsystem comprised of:

   1. One OPERABLE LPCI pump, and

   2. An OPERABLE flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel.

c. The high pressure coolant injection (HPCI) system consisting of:

   1. One OPERABLE HPCI pump, and

   2. An OPERABLE flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel.

d. The automatic depressurization system (ADS) with five OPERABLE ADS valves.

APPLICABILITY: OPERATIONAL CONDITION 1, 2*,** #, and 3*,**,##.

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*The HPCI system is not required to be OPERABLE when reactor steam dome pressure is less than or equal to 200 psig.

**The ADS is not required to be OPERABLE when reactor steam dome pressure is less than or equal to 100 psig.

#See Special Test Exception 3.10.6.

##Two LPCI subsystems of the RHR system may be inoperable in that they are aligned in the shutdown cooling mode when the reactor vessel pressure is less than the RHR shutdown cooling permissive setpoint.

87

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.3.2 Reactor coolant system leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE.

b. 5 gpm UNIDENTIFIED LEAKAGE.

c. 25 gpm IDENTIFIED LEAKAGE averaged over any 24-hour period.

d. 0.5 gpm leakage per nominal inch of valve size up to a maximum of 5 gpm from any reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1, at rated pressure.

e. 2 gpm increase in UNIDENTIFIED LEAKAGE within any period of 24 hours or less.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With any reactor coolant system leakage greater than the limits in b and/or c, above, reduce the leakage rate to within the limits within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With any reactor coolant system pressure isolation valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least one other closed manual or deactivated automatic or check* valves, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. With one or more of the high/low pressure interface valve leakage pressure monitors shown in Table 3.4.3.2-2 inoperable, restore the inoperable monitor(s) to OPERABLE status within 7 days or verify the pressure to be less than the alarm setpoint at least once per 12 hours; restore the inoperable monitor(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

e. With any increase in UNIDENTIFIED LEAKAGE exceeding the limit in e above, implement preplanned leak location and isolation actions and either verify that the source of the leakage is not service-sensitive type 304 or 316 stainless steel or reduce the leakage rate-of-change to less than the limit within 4 hours or be in HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

* Which have been verified not to exceed the allowable leakage limit at the last refueling outage or after last time the valve was disturbed, whichever is more recent.

TABLE 3.4.3.2-2

REACTOR COOLANT SYSTEM INTERFACE VALVES

LEAKAGE PRESSURE MONITORS

SERVICE	INSTRUMENT	ALARM SETPOINT (psig)	ALARM ALLOWABLE VALUE (psig)
Core Spray	1-BE-PISH-N654A	475	≤ 500
Core Spray	1-BE-PISH-N654B	475	≤ 500
LPCI/RHR	1-BC-PISH-N653A	380	≤ 410
LPCI/RHR	1-BC-PISH-N653B	380	≤ 410
LPCI/RHR	1-BC-PISH-N653C	380	≤ 410
LPCI/RHR	1-BC-PISH-N653D	380	≤ 410
RHR	1-BC-PISH-N657	130	≤ 155

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	211000 G2.4.47	
	Importance Rating		4.2

Emergency Procedures / Plan: Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.(SLC)

Proposed Question: SRO 88

Given the following:

- A feedwater line break in the drywell has occurred
- Reactor Power is 3.5% and steady
- Reactor Pressure is cycling between 800 to 1000 psig
- RPV level is -100 inches and lowering at 1" per minute
- HPCI is in service
- Drywell Pressure is 3.8 psig and rising at 0.1 psig per minute
- Drywell Temperature is 168 degrees F and rising at 3 degrees per minute
- Suppression Pool Temperature is 112 degrees and rising at 4 degrees per minute

Assuming the trends continue as above and all systems are operable, which one of the following is/will be required?

A. Immediately lower RPV Level until it reaches -129" IAW EOP-101A "ATWS-RPV Control"

B. Inject SLC before 7 minutes has elapsed IAW EOP-101A "ATWS-RPV Control"

C. Emergency Depressurize in 5 minutes IAW EOP-202 "Emergency Depressurization"

D. Terminate & Prevent Injection in 6 minutes IAW EOP-101A "ATWS-RPV Control", once that is complete Emergency Depressurize IAW EOP-202 "Emergency Depressurization"

Proposed Answer: B

Explanation (Optional): IAW EOP-101A

B. Correct – in 7 minutes, SP temperature will be 140 degrees. IAW the EOP101A, Step RC/Q-10, SLC must be injected before SP temp reaching 140 degrees

A. Incorrect. With power <4%, level is not lowered

C. Incorrect. No parameters will have met the ED requirement

D. Incorrect. Terminate & Prevent would occur only if level could not be maintained above --
-185 inches

Technical Reference(s) EOP-101A (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: EO101AE006 (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ____
55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	205000, 2.2.22	
	Importance Rating		4.7

(K&A Statement) Equipment Control: Knowledge of limiting conditions for operations and safety limits. (Shutdown Cooling)

Proposed Question: SRO 89

The plant is 3 days into a shutdown. Reactor cavity level is 23 feet above the RPV head flange.

Which one of the following is the minimum Shutdown Cooling systems/components required by Technical Specifications for these conditions.

A. One loop of Shutdown Cooling consisting of one OPERABLE RHR pump and one OPERABLE RHR Heat Exchanger in operation.

B. Two loops of Shutdown Cooling each consisting of one OPERABLE RHR pump and one OPERABLE RHR Heat Exchanger in operation.

C. Two loops of Shutdown Cooling OPERABLE and one Recirculation Pump in operation.

D. NO Shutdown Cooling loops OPERABLE and one Recirculation Pump in operation.

Proposed Answer: A

Explanation (Optional):

A. Correct - IAW T/S 3.9.11.1

B. Incorrect – this would be correct for low water level TS 3.9.11.2
C. Incorrect – not correct for given conditions, not the minimum.
D. Incorrect – this is the action requirements for TS 3.4.9.2

Technical Reference(s): T.S. 3.9.11.1 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: IOP009E006 (As available)

Question Source: Bank #

Modified Bank # _____ (Note changes or attach parent)

New X

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 _____
55.43 2

Comments:

#89
NRC

REFUELING OPERATIONS

3/4.9.11 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.11.1 At least one shutdown cooling mode loop of the residual heat removal (RHR) system shall be OPERABLE and in operation* with:

a. One OPERABLE RHR pump, and

b. One OPERABLE RHR heat exchanger.

APPLICABILITY: OPERATIONAL CONDITION 5, when irradiated fuel is in the reactor vessel and the water level is greater than or equal to 22 feet 2 inches above the top of the reactor pressure vessel flange and heat losses to ambient** are not sufficient to maintain OPERATIONAL CONDITION 5.

ACTION:

a. With no RHR shutdown cooling mode loop OPERABLE, within one hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternate method capable of decay heat removal. Otherwise, suspend all operations involving an increase in the reactor decay heat load and establish SECONDARY CONTAINMENT INTEGRITY within 4 hours.

b. With no RHR shutdown cooling mode loop in operation, within one hour establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature at least once per hour.

SURVEILLANCE REQUIREMENTS

4.9.11.1 At least one shutdown cooling mode loop of the residual heat removal system or alternate method shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

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* The shutdown cooling pump may be removed from operation for up to 2 hours per 8-hour period.

** Ambient losses must be such that no increase in reactor vessel water temperature will occur (even though REFUELING conditions are being maintained).

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	218000 A2.05	
	Importance Rating		3.6

(K&A Statement) Ability to (a) predict the impacts of the following on the AUTOMATIC DEPRESSURIZATION SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of A.C. or D.C. power to ADS valves

Proposed Question: SRO 90

The reactor scrammed from 100% power due to a loss of offsite power with a LOCA.

The following conditions exist:

- All EDGs started, however busses 10A401, 10A402, 10A403, and 10A404 all have bus lockouts (due to ground faults).
- Reactor pressure is 400 psig and lowering
- Reactor water level has reached -185 inches and is lowering at 1 inch/minute.
- RCIC is injecting at rated flow.
- HPCI is injecting and reaching rated flow.
- Drywell pressure is 12.5 psig and slowly rising.

Which one of the following is the result of these conditions and what actions are required?

A. The ADS valves are available, enter EOP-206 "RPV Flooding" and open five SRVs and flood the RPV to the Main Steam Lines.

B. The ADS valves are available, enter EOP-202, "Emergency RPV Depressurization", and open five SRVs and depressurize the RPV.

C. The ADS valves are NOT available, enter EOP-202 "Emergency RPV Depressurization", and use Alternate Depressurization Systems to depressurize.

D. The ADS valves are NOT available, enter EOP-206 "RPV Flooding" and use Alternate Depressurization Systems to depressurize and flood the RPV to the Main Steam Lines.

Proposed Answer: B

Explanation (Optional):

B. Correct – although no RHR or Core Spray systems are available, HPCI and RCIC are both preferred systems and although they will go away when the RPV is depressurized EOP-101 directs a blowdown when an injection system is available and RPV level cannot be maintained >-185 inches.

A. Incorrect – There is no need to enter EOP-206 at this time.
C. Incorrect – The ADS valves are available (the auto ADS function is NOT available but the SRVs can be manually opened.
D. Incorrect - The ADS valves are available (the auto ADS function is NOT available but the SRVs can be manually opened. There is no need to enter EOP-206 at this time.

Technical Reference(s): EOP-101 step AL/C-9,10,11 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: EO101LE006 (As available)

Question Source: Bank # _____
Modified Bank # _____ (Note changes or attach parent)
New X

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 _____
55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	245000 A2.02	
	Importance Rating		3.5

Ability to (a) predict the impacts of the following on the MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of lube oil

Proposed Question: SRO 91

The plant was operating at 100% power in a normal system alignment when a steam leak in the Drywell causes pressure to rise. The Reactor is shutdown prior to Drywell pressure reaching 1.68 psig and all SCRAM actions are carried out by the RO.

Ten minutes after the scram the following conditions exist:

- Drywell Pressure is 2.1 psig and slowly rising
- RPV level is –40" and rising, with HPCI and RCIC injecting
- RPV pressure is 800 psig and lowering slowly
- Turbine Speed is 200 RPM and lowering

Further investigation identifies all Lift Pumps and the Turning Gear Oil Pump are NOT running.

Which one of the following describes the plant response and actions required, if any?

A. The TGOP failed to AUTO start and must be manually started to allow the Lift Pump to start IAW HC.OP-AB.BOP-0002 "Main Turbine".

B. The Lube Oil system has an apparent leak causing the TGOP and lift pumps to trip. The NEO should be sent to investigate IAW HC.OP-SO.CB-0001 "Main Turbine and Generator Lube Oil System Operation".

C. The plant responded as designed due to the High Drywell pressure. The operator must verify the EBOP is running IAW the overhead alarm response for window C8-F3, Digital Point D5573, "Turning Gear Oil Pump Trouble".

D. The plant responded as designed due to the Main Turbine speed coasting down. The operator must verify the TGOP and Lift Pumps start when Turbine speed reaches < 100 RPM IAW HC.OP-SO.CB-0001 "Main Turbine and Generator Lube Oil System Operation".

Proposed Answer: C

Explanation (Optional):

C. **Correct**-: IAW HC.OP-SO.SM-0001 the 10B323 load center is stripped on a LOCA 1 signal the TGOP is de-energized resulting in the Lift pumps tripping. Therefore the EBOP must be verified as running for the turbine lube oil system IAW the alarm response procedure.

A. **Incorrect-** IAW HC.OP-SO.SM-0001 the 10B323 load center is stripped on a LOCA 1 signal the TGOP is de-energized.
B. **Incorrect**- without the TGOP running and suction pressure > 1 psig the Lift pumps trip.
D. **Incorrect**- the LIFT pumps and TGOP due not rely on Turbine speed for a start signal, the TGOP starts on pressure and the lift pump starts on the TGOP and > 1 psig suction pressure.

Technical Reference(s) HC.OP-AB.BOP-0002 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: MTLO00E011 (As available)

Question Source: Bank # 56902
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 _____
55.43 5

Comments:

SUBSEQUENT OPERATOR ACTIONS (continued)

CONDITION	ACTION
E. Hydrogen Seal Oil System Malfunction. Date/Time: _____	___ **NOTE 9** ___ E.1 IF the Emergency Seal Oil Pump fails to auto start, THEN **START** the Emergency Seal Oil Pump.
F. Generator H_2 pressure drops below 63 psig. Date/Time: _____	F.1 IF Reactor Power ≥24% THEN **PERFORM** the following: ___ a. **REDUCE** Recirc. Pump speed to MINIMUM. ___ b. **LOCK** the Mode Switch in Shutdown. ___ F.2 **TRIP** the Turbine AND **LOCK OUT** the Generator. ___ ★ **CAUTION 2** ★ ___ F.3 **PLACE** BOTH Stator Water Cooling Pump Control Switches to PULL TO LOCK. **[CD-197G]** ___ F.4 **OPEN** Breaker 52-112113 Stator Coolant Water Heater 10E113 power supply.
G. Both the Emergency Seal Oil Pump AND The Main Seal Oil Pump CANNOT be started, OR BEFORE Generator H_2 pressure drops below 30 psig. Date/Time: _____ **(Continued on Page 19)**	G.1 IF Reactor Power ≥24% THEN **PERFORM** the following: ___ a. **REDUCE** Recirc. Pump speed to MINIMUM. ___ b. **LOCK** the Mode Switch in Shutdown. ___ G.2 **TRIP** the Turbine AND **LOCK OUT** the Generator. G.3 **PERFORM** the following: ___ • **EVACUATE** unnecessary personnel from Turbine Generator Area ___ • **ENSURE** Turbine Building Ventilation in Purge ___ • **START** all available Turbine Building Exhaust / Supply Fans. **(Continued on Page 19)**

#91

RC

DIGITAL ALARM POINT D5573

NOMENCLATURE TURNING GEAR OIL PUMP TROUBLE **SETPOINT** 10 psig

DESCRIPTION TGOP auto start failure trouble **ORIGIN** PSH-3109

AUTOMATIC ACTION:

1. Turning Gear Oil Pump auto start has occurred.
2. Loss of Lift Pumps running signal causes Turning Gear to trip.
3. Loss of discharge pressure signal from TGOP causes Lift Pumps to trip.

OPERATOR ACTION:

1. IF Main Turbine is on Turning Gear,
   ENSURE Turning Gear trips (TURN GEAR MOT AMPS indicates 0).
2. **ENSURE** LIFT BEARING PUMPS STOP is ON.
3. **ENSURE** Motor Suction Pump
   OR Emergency Bearing Oil Pump is running and supplying bearing header.
4. IF no bearing oil header pressure is available,
   STOP Main Turbine shaft rotation as quickly as possible.

CAUSE	CORRECTIVE ACTION
1. Discharge pressure low WHEN a start signal is present	1A. **PRESS** START on Lube Oil Reservoir Pumps Turning Gear. (10C651C)
a. Failure of TGOP to start on auto start signal	1B. **CHECK** Turning Gear Oil Pump is not in LOCKOUT.
b. TGOP LOCKOUT	1C. IF TGOP starts, **PRESS** LOW DISCH PRESS to clear amber light. (10C651C)
c. Loss of bus power	
d. Loss of control power	1D. **DETERMINE** cause of auto start failure. **REQUEST** SM/CRS to initiate corrective action.
1. Blown control power fuse	

Associated with Annunciator C8 F3

REFERENCES: J-200-Q-181-10 E-0252-0 Sht.1
J-19-0 Sht. 4

#91
NRC

DIGITAL ALARM POINT D5573

CAUSE	CORRECTIVE ACTION
Discharge pressure low WHEN a start signal is present (continued)	1E. **DISPATCH** an operator to MCC 10B323. Check TGOP breaker on (52-323074). **RESET** thermal overload AND **RESTART** TGOP. 1F. **ENSURE** all Lift Pumps are on. **CHECK** bearing oil is supplied by visually inspecting Turbine Bearing Sightglasses.

Associated with Annunciator C8 F3

REFERENCES: J-200-Q-181-10 E-0252-0 Sht. 1
J-19-0 Sht. 4

H91
NRC

OBSERVE the Equipment or Valves listed in Table **SM-020** have met their required Action under the Manual or Automatic Isolation Signals

TABLE SM-020 (1 of 8)							
EQUIPMENT NUMBER	NOMENCLATURE	ACTION	MANUAL ISOLATION	AUTO ISOLATIONS			
				A	B	C	D
"52-232094	INSTRUMENT GAS CPRSR AK202	TRIP	+C CNTMT	X		X	X
"52-242094	INSTRUMENT GAS CPRSR BK202	TRIP	+D CNTMT	X		X	X
"52-41011	RACS PMP AP209 BRKR	TRIP	A CS	X		X	
"52-42011	RACS PMP BP209 BRKR	TRIP	B CS	X		X	
"52-43014	CRD PMP AP207 BRKR	TRIP	C CS	X		X	
"52-44014	CRD PMP BP207 BRKR	TRIP	D CS	X		X	
"52-41014	MCC 10B313 BRKR	TRIP	A CS	X		X	
"52-42014	MCC 10B323 BRKR	TRIP	B CS	X		X	
"52-47011	MCC 10B272 BRKR	TRIP	C CS	X		X	
"52-48011	MCC 10B282 BRKR	TRIP	D CS	X		X	
"52-45011	MCC 10B252 BRKR	TRIP	A CS	X		X	
"52-46011	MCC 10B262 BRKR	TRIP	B CS	X		X	
"52-47031	MCC 00B474 BRKR	TRIP	C CS	X		X	
52-451023	INVERTER 10D496 B/U PWR (PA SYST)	TRIP	A CS	X		X	

ISOLATION	SETPOINT
A - REACTOR VESSEL WATER LEVEL 1	-129"
C - DRYWELL PRESSURE - HIGH	1.68 psig
D - REACTOR BUILDING EXHAUST RADIATION - HIGH	1 X 10^{-3} uCi/cc

+ - Also Isolates from the corresponding Core Spray Manual Initiation PB
* - Can receive a Half Isolation from the corresponding NSSSS Manual Isolation
" - Isolation can be bypassed by TRIP OVRD PB

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	202001 G2.4.49	
	Importance Rating		4.4

Emergency Procedures / Plan: Ability to perform without reference to procedures those actions that require immediate operation of system components and controls. (recirculation)

Proposed Question: SRO 92

The plant is operating at 100% power, when a lighting strike caused the "A" Reactor Recirc pump to trip, the "B" Reactor Recirc Pump to runback to it's intermediate speed and isolation of 6A FW Heater. All other plant equipment is operable.

Currently Rx power is 56% and Core flow is 41%.

Which of the following describes the action required, if any, to ensure core stability?

A. Unintentional operation in this region is allowed, NO further actions are required.

B. IAW OP-SO.BB-0001 "Recirculation System Operation", Start the "A" Reactor Recirc Pump to exit this region.

C. Exit Region 1 using Enhanced Stability Guidance IAW OP-AB.RPV-0003 "Recirculation System/Power Oscillations".

D. Immediately lock the mode switch in shutdown IAW OP-AB.RPV-0003 "Recirculation System/Power Oscillations".

Proposed Answer: C

Explanation (Optional): HC.OP-AB.RPV-0003

C. Correct - Insert control rods IAW RE guidance to exit this region.

A. Incorrect Unintentional operation in this region is not allowed, immediately Lock the Mode Switch in Shutdown. Incorrect – Mode switch is not required Locked in Shutdown and rod insertion is method to exit this region

B. Incorrect - Start the "A" Reactor Recirc pump to exit this region - Starting an idle Recirc pump to exit this region is NOT IAW with the AB.

D. Incorrect - Unintentional operation in this region is allowed, no further actions are required. Unintentional operations in this region are not allowed and actions to exit this region are required.

Technical Reference(s) HC.OP-AB.RPV-0003 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: Power–to-flow map OPRMs operable/inoperable – reference does not state the action required but is needed to determine the action

Learning Objective: IOP003E005 (As available)

Question Source: Bank # 56978
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 _____
55.43 5

Comments:

#92

CONDITION	ACTION
B. Unplanned entry resulting in Reactor Power ≥ 26.1% AND Core Flow ≤ 68% (nominal). Date/Time: _____ **Continued on Page 11**	❑ B.1 **MONITOR** for Power Oscillations. (Thermal Hydraulic Instabilities) ❑ **NOTE 3** ❑ B.2 **VERIFY** OPRM TRIP ENABLE (C3-F1) alarm. ❑ ★ **CAUTION 2** ★ B.3 IF OPRM's are OPERABLE, THEN **PERFORM** the following: ❑ A. IF in REGION 1 of Figure 2 THEN **EXIT** REGION 1 IAW Enhanced Stability Guidance. (RE.ZZ-0001) B. IF OPRM ALARM (C3-F2) is received THEN **PERFORM** the following: ❑ • **SUSPEND** Control Rod withdrawal AND Recirculation Pump speed reduction. ❑ • IF applicable, **CONTINUE** operation WITH Enhanced Stability Guidance. (RE.ZZ-0001) ❑ C. IF OPRM ALARM (C3-F2) will NOT remain Clear, OR MORE THAN one OPRM Channel has an "OPRM Alarm" (**REFER** to CRIDS Page 248) THEN **CONTINUE** Rod insertion IAW Enhanced Stability Guidance UNTIL the alarm is clear. (RE.ZZ-0001) **NOTE 4** ❑ D. WHEN OPRM ALARM (C3-F2) is CLEAR, **PRESS** the OPRM LED reset at Panel 10C608. **Continued on Page 11**

RETAINMENT OVERRIDE	
CONDITION	**ACTION**
I. DANGER Limit is reached on Recirc Pump/Motor Vibrations analog points: • A(B) Radial PMP A2601(A2603) – 16.0 mils • A Radial MTR A2602 – 4.5 mils • B Radial MTR A2604 – 8.0 mils Date/Time: _____	❑ **I.a** **TRIP** the affected Recirc Pump. ❑ **I.b** **ENTER** Condition A.
II. Affected Recirc Pump seal cavity temperature greater than 200 degrees F Date/Time: _____	❑ **II.a** **REDUCE** the affected Recirc pump speed to minimum, **TRIP** the affected Recirc pump and enter Condition A

CAUTIONS:

2. **DO NOT intentionally enter the REGION 1 while changing power. Passing through this region or lowering flow once in REGION 1 is acceptable if the intent is to TAKE the Mode Switch to shutdown. [PR 971229163]**

NOTES:

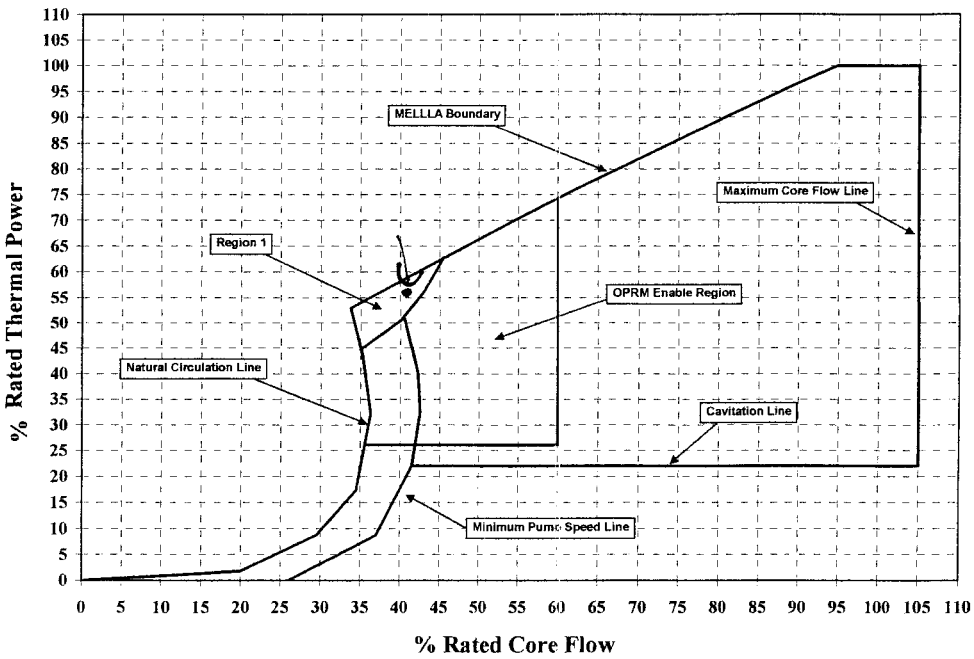
3. 68% core flow is the nominal value to enable the alarm. To ensure the OPRM enables during all conditions, e.g. SLO, the value has been set at 68% core flow. It is possible to receive the Trip Enable Alarm at a higher value, and may require up to 73% core flow to clear the alarm.
4. The OPRM ALARM overhead alarm (C3-F2) and associated CRIDS alarms will clear automatically when the condition clears. The OPRM Alarm LED on Local Panel 10C608 is a "lock in" indication and requires physically resetting when the condition clears.

ADDITIONAL INFORMATION:

Indications:

- CRIDS PAGE 248
- OPRM Alarm, Red LED @ Panel 10C608

Figure 2 - POWER TO FLOW MAP (OPRM OPERABLE)



Power Oscillations (Thermal Hydraulic Instabilities) are defined as any of the following:

- 10% peak to peak Power Oscillations on APRM Recorders
- Periodic Upscale OR Downscale LPRM Alarms (on the Full Core Display)
- Strong positive/negative swings on the Period Meters

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	215001 A2.07	
	Importance Rating		3.7

Ability to (a) predict the impacts of the following on the TIPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Failure to retract during accident conditions

Proposed Question: SRO 93

A Traversing Incore Probe (TIP) trace is in progress when a high drywell pressure event (> 1.68 psig) occurs due to a leak in the recirculation system. Three minutes following the event, the Reactor Operator reports the following indications on the TIP Valve Control Monitor:

- "SQUIB MONITOR" lights - both illuminated
- "SHEAR VALVE MONITOR" lights - both extinguished
- "BALL VALVE OPEN" lights - both illuminated
- "BALL VALVE CLOSED" lights - both extinguished

Which of the following describes the status of the TIP system, the next required operator action(s), if any, and status of Primary Containment Isolation (in regard to TIPS ONLY)? (Assume NO operator actions have been taken)

A. The system has responded as designed. Operator action is required to close the ball valves IAW OP-SO.SE-0002 "TIP System Operation", to ensure Primary Containment isolation.

B. The system has responded as designed. IAW OP-SO.SE-0002 "TIP System Operation", direct the operators to fire the shear valves to ensure Primary Containment has been isolated.

C. The TIP detectors may NOT have withdrawn. IAW OP-AB.CONT-0002, Withdraw the detectors and verify the ball valves close. Primary Containment will be isolated once the ball valves are closed.

D. The TIP detectors may NOT have withdrawn. IAW OP.AB-CONT-0002, Fire the shear valves, withdraw the detectors and then close the ball valves. Primary Containment is considered isolated ONLY after the ball valve has closed and shear valve has fired.

Proposed Answer: C

Explanation (Optional):

C. Correct: The ball valve should be closed. A failure to auto retract could be the problem. The next action would be to attempt to withdraw the detectors and verify the ball valve closes.

A. Incorrect: Did not respond as designed. The ball valve should be closed and the squib monitor light extinguished.
B. Incorrect: The system did not respond as designed. The ball valve should be closed and the squib monitor light extinguished. Primary Containment is not isolated.
D. Incorrect: The next action is to attempt a manual withdrawal and close the ball valve. The squib valves have lost continuity and will not fire. Primary Containment Isolation will occur once the ball valves are closed.

Technical Reference(s) IAW OP.AB-CONT-0002 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: ABPRCUE003 (As available)

Question Source: Bank # ____
Modified Bank # NRC 2007 (Note changes or attach parent)
New ____

Question History: Last NRC Exam 2007

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ____
55.43 5

Comments:

#93
NRC

SUBSEQUENT OPERATOR ACTIONS (continued)

CONDITION	ACTION
F. TIP CANNOT be isolated manually AND it is a source of leakage **[T/S 3.6.3]** **[CD-118X]** Date/Time: _____	___ ** **NOTE 2** ___ F.1 At the discretion of the SM/CRS, **FIRE** the shear valve. F.2 **VERIFY** the following indications lit: ___ • SQUIB MONITOR ___ • SHEAR VALVE MONITOR ___ • TIP SHEAR VALVE CLOSED/ INOP
G. TIP fails to retract. **[T/S 3.6.3]** Date/Time: _____	___ G.1 **RETRACT** the TIP AND **CLOSE** the Ball Valve manually OR ___ G.2 **DIRECT** I&C to manually retract the TIP IAW HC.IC-GP.SE-0005(Q).

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		3
	K/A #	G2.1.35	
	Importance Rating		3.9

Knowledge of refueling responsibilities of SRO

Proposed Question: SRO 94

Plant conditions are as follows:

- The Reactor Vessel is prepared for refueling operations IAW "Cold Shutdown to Refueling", HC.OP-IO.ZZ-0005.
- Prerequisite plant conditions have been verified IAW "Refueling Operations" HC.OP-IO.ZZ-0009.
- Spiral Fuel offload is in progress per directions of Reactor Engineers and Fuel Handling Control Core Alteration forms HC.RE-FR.ZZ-0001.
- Multiple Control Rod blades and drive mechanisms are being removed IAW Technical Specification 3.9.10.2

Then, Reactor Engineering reports Shutdown Margin CANNOT be demonstrated.

Which of the following are required?

A. Stop fuel handling in the fuel pool and return the Control Rod Blades to the reactor vessel.
Then, remove the shorting links prior to resuming any fuel or Control Blade movement.

B. Stop Control Rod Blade removal from the reactor vessel. Fuel handling in the fuel pool may continue.
Control Rod Blade removal from the reactor vessel may continue once the shorting links are removed.

C. Stop fuel handling in the fuel pool and return the Control Rod Blades to the reactor vessel.
Then, install the shorting links prior to resuming either of the above activities.

D. Stop Control Rod Blade removal from the reactor vessel. Fuel handling in the fuel pool may continue.
Control Rod Blade removal from the reactor vessel may continue once the shorting links are installed.

Proposed Answer: B

Explanation (Optional): B. Correct

B. Correct. T.S. action 3.1.1 c requires suspension of Core Alterations and other activities that could reduce the Shutdown margin, and insert all insert able control rods. Returning CRBs to the RPV would be Core Alterations (as defined in TS)
TS 3.9.10.2.c both require SDM demonstration or suspend control rod removal
T.S. 3.9.2.c allows RPS shorting Link removal in place of SDM demonstration, or suspend all Core Alterations and insert all insert able control rods.

A. Incorrect- Fuel Pool fuel handling is allowed.
C. Incorrect. Shorting links must be removed. Fuel Pool fuel handling is allowed
D. Incorrect. Shorting links must be removed.

Technical Reference(s) HC.OP-IO-ZZ-0009 (Attach if not previously provided)
TS 3.9.10.2, 3.1.1

Proposed references to be provided to applicants during examination: None

Learning Objective: IOP009E006 (As available)

Question Source: Bank # 76882
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 _____
55.43 2, 7

Comments:

NOTE

If adequate Shutdown Margin has been demonstrated IAW T/S 3.1.1, then removal of the RPS shorting links is NOT required.

5.1.3. WHEN **within 8 hours prior** to the start of CORE ALTERATIONS requiring control rod withdrawal (Except for control rods removed IAW T/S 3.9.10.1 OR 3.9.10.2), THEN **PERFORM** the following: **[T/S 4.9.2.d]**

A. **VERIFY** EITHER Reactor Protection System shorting links are removed
OR
Adequate Shutdown Margin has been demonstrated IAW T/S 3.1.1. ____

B. **RECORD** the time, and date, the applicable surveillance requirement was satisfied
AND "N/A" for other
AND **ENTER** initials on Attachment 1. ____

#94

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.1.1 The SHUTDOWN MARGIN shall be equal to or greater than:

a. 0.38% delta k/k with the highest worth rod analytically determined, or

b. 0.28% delta k/k with the highest worth rod determined by test.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4 and 5.

ACTION:

With the SHUTDOWN MARGIN less than specified:

a. In OPERATIONAL CONDITION 1 or 2, reestablish the required SHUTDOWN MARGIN within 6 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 3 or 4, immediately verify all insertable control rods to be inserted and suspend all activities that could reduce the SHUTDOWN MARGIN. In OPERATIONAL CONDITION 4, establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

c. In OPERATIONAL CONDITION 5, suspend CORE ALTERATIONS and other activities that could reduce the SHUTDOWN MARGIN and insert all insertable control rods within 1 hour. Establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

SURVEILLANCE REQUIREMENTS

4.1.1 The SHUTDOWN MARGIN shall be determined to be equal to or greater than specified at any time during the fuel cycle:

a. By measurement, prior to or during the first startup after each refueling.

b. By measurement, within 500 MWD/T prior to the core average exposure at which the predicted SHUTDOWN MARGIN, including uncertainties and calculation biases, is equal to the specified limit.

c. Within 12 hours after detection of a withdrawn control rod that is immovable, as a result of excessive friction or mechanical interference, or is untrippable, except that the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod.

1794

CORE ALTERATION

1.7 CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. The following exceptions are not considered to be CORE ALTERATIONS:

a. Movement of source range monitors, local power range monitors, intermediate range monitors, traversing incore probes, or special movable detectors (including undervessel replacement), and

b. Control rod movement, provided there are no fuel assemblies in the associated core cell.

Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

1.8 DELETED

CORE OPERATING LIMITS REPORT

1.9 The CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.9. Plant operation within these limits is addressed in individual specifications.

CRITICAL POWER RATIO

1.10 The CRITICAL POWER RATIO (CPR) shall be the ratio of that power in the assembly which is calculated by application of the applicable NRC- approved critical power correlation to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.

DOSE EQUIVALENT I-131

1.11 DOSE EQUIVALENT I-131 shall be that concentration of I-131, microcuries per gram, which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

$\bar{E}$-AVERAGE DISINTEGRATION ENERGY

1.12 $\bar{E}$ shall be the average, weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling, of the sum of the average beta and gamma energies per disintegration, in MeV, for isotopes, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

#94

REFUELING OPERATIONS

MULTIPLE CONTROL ROD REMOVAL

LIMITING CONDITION FOR OPERATION

3.9.10.2 Any number of control rods and/or control rod drive mechanisms may be removed from the core and/or reactor pressure vessel provided that at least the following requirements are satisfied until all control rods and control rod drive mechanisms are reinstalled and all control rods are inserted in the core.

a. The reactor mode switch is OPERABLE and locked in the Shutdown position or in the Refuel position per Specification 3.9.1, except that the Refuel position "one-rod-out" interlock may be bypassed, as required, for those control rods and/or control rod drive mechanisms to be removed, after the fuel assemblies have been removed as specified below.

b. The source range monitors SRM are OPERABLE per Specification 3.9.2.

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied.

d. All other control rods are either inserted or have the surrounding four fuel assemblies removed from the core cell.

e. The four fuel assemblies surrounding each control rod or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.

f. All fuel loading operations shall be suspended.

APPLICABILITY: OPERATIONAL CONDITION 5.

ACTION:

With the requirements of the above specification not satisfied, suspend removal of control rods and/or control rod drive mechanisms from the core and/or reactor pressure vessel and initiate action to satisfy the above requirements.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		2
	K/A #	G2.2.7	
	Importance Rating		3.6

Knowledge of the process for conducting special or infrequent tests.

Proposed Question: SRO 95

Given the following conditions:

The plant is in Operational Condition 1. It's been determined that work must be performed on the 6A Feedwater Heater level controller. It has been determined that the work could result in an unplanned load reduction of 40 MWe.

This evolution is...

A. a production risk activity and an IPTE brief is required.

B. a production risk activity and a HLA/IPA brief is required.

C. NOT a production risk activity but an IPTE brief is still required.

D. NOT a production risk activity but an HLA/IPA brief is still required.

Proposed Answer: B

Explanation (Optional): IAW WC-AA-104 – Step 2.4 defines production risk activity as >20 MWe. Then an HLA/IPA

B. Correct
A. Incorrect - an HLA/IPA briefing is required. IPTE briefings are no longer performed for these evolutions.
C. Incorrect – this is a production risk activity, an HLA/IPA briefing is required.
D. Incorrect - this is a production risk activity. IPTE briefings are no longer performed for these evolutions.

Technical Reference(s) OP-AA-108-110 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: OPSBRIEFE005 (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 ____
55.43 3

Comments:

H95

REVIEW AND SCREENING FOR PRODUCTION RISK

1. PURPOSE

1.1. This procedure provides the proper steps required to screen, plan and oversee work activities to manage production risk.

1.2. Applicability

1.2.1. This procedure applies to all on-line work activities ***pre-outage activities, and common unit outage work*** unless excluded by this procedure.

1.2.2. It is Management's expectation that personnel understand their work activities and the effects of those activities on plant/system operation. This is best accomplished through a combined effort of involved parties.

1.2.3. Work activities screened prior to the implementation of this revision, are **not** required to be revised to the new forms/process.

2. TERMS AND DEFINITIONS

2.1. **Critical Step or Task** – Any action that when performed improperly will lead to an unintentional change that adversely impacts plant, systems, or personnel status.

2.2. **Power Block** - Includes all structures, systems and components which perform a direct function in the production of, transport of, or storage of heat energy, electrical energy or radioactive wastes. Also included are structures, systems and components that monitor, control, protect or otherwise support the above equipment.
Hope Creek – Reactor Building, Turbine Building, Control and D/G areas, Service Water Intake, Circulating Water and Cooling Tower, and Switchyard.
Salem (1&2) – Containment Building, Aux Building, Turbine Building, Control and D/G areas, Fuel Handling Building, Service Water Intake, Circulating Water Intake, and Switchyard.
Salem 3 – all areas.

2.3. **Procedure Risk Matrix** – A listing of frequently performed procedures that have been evaluated for Production Risk.

2.4. **Production Risk Activity** - Any activity that has the potential to derate the plant. A derate constitutes a > 20 MWE load reduction from the power level(s) committed to the electric operations dispatcher.

2.5. **Production Risk Systems** – A list of systems determined by each site. Work activities on those systems could cause a > 20 MWE load reduction from the power level(s) committed to the electric operations dispatcher, see Attachment 4, Hope Creek Production Risk System Matrix and Attachment 5, Salem Production Risk System Matrix..

#95

4.1.3. The individual evaluating an activity **ENSURES** the SLM concurs with the results of their evaluation.

4.1.4. **If** an activity is determined to be a special test or evolution, **then** the SLM shall **ASSIGN** a Special Test or Evolution Coordinator to perform sections 4.2 and 4.3. **If** **not**, **then** **no** further action is required under this procedure.

4.1.5. The SLM shall **ENSURE** proper plant management approval has been received to perform the special test or evolution.

4.2. Implementing Special Test Or Evolution Controls

4.2.1. **If** an evolution is determined to be a special test or evolution and **no** procedure exists, **then PREPARE** a special procedure in accordance with approved plant guidance.

4.2.2. The SQR shall **DETERMINE** the cross discipline review requirements of a special test or evolution procedure for technical adequacy.

1. If the special test or evolution has the potential to affect Reactivity Management, then this review shall be performed by Reactor Engineering (OP-AA-300-1540).

4.3. Special Test Or Evolution Performance

4.3.1. **CONSIDER** Just in Time training to **ENSURE** plant conditions and the activity are understood by those involved.

4.3.2. **ENSURE** the special test or Evolution is reviewed by senior management prior to the start of the evolution.

4.3.3. The SLM or his designee shall **CONDUCT** a HLA/IPA briefing prior to performing the special test or evolution. This briefing shall cover all the applicable topics from HU-AA-1211 and should be attended by all personnel performing the activity.

4.3.4. **If** the duration of the special test or evolution is greater than one shift, **PERFORM** the HLA/IPA briefing for each shift.

4.3.5. The SLM and the Special Test Or Evolution Coordinator shall **ENSURE** appropriate station management involvement is present prior to and during special test or evolution performance.

4.3.6. Operations Shift Management shall **ENSURE** required plant conditions are maintained as required for the special test or evolution.

5. **DOCUMENTATION** - None

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3.3. Special Test Or Evolution Coordinator

3.3.1. Utilizes the guidance provided in Attachment 1 and Attachment 2 to determine if the activity requires special test or evolution controls.

3.3.2. Ensures Senior Line Manager and Shift Management are aware of the status of a special test or evolution.

3.4. Operations Shift Management

3.4.1. Approves implementation of the special test or evolution.

3.4.2. Ensures required plant conditions are maintained during performance of the special test or evolution.

3.5. Site Reactor Engineering

3.5.1. Reviews each special test or evolution that involves plant systems potentially affecting reactivity or changes to the core or fuel pool (OP-AA-300-1540).

3.6. Training

3.6.1. Develops and validates appropriate training material to support the evolution or test as necessary.

3.6.2. Provides challenging, relevant, and effective training necessary to perform the special evolution or test.

3.6.3. Reinforces line management expectations and department fundamentals in areas applicable to the training setting.

4. MAIN BODY

4.1. Identification of Special Test or Evolutions

4.1.1. **REVIEW** the following types of activities to determine if screening under this procedure is warranted:

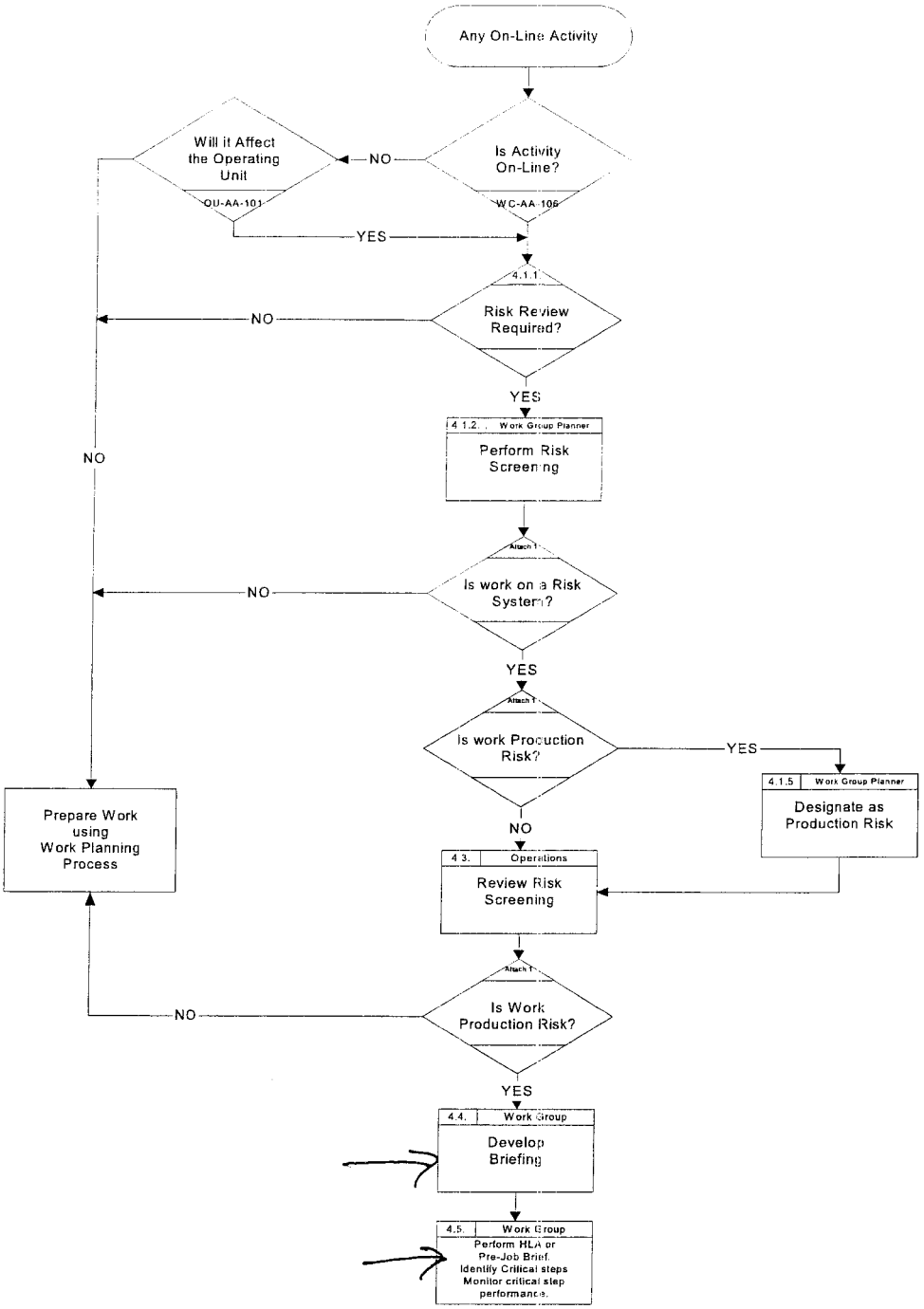
1. Work activities screened as "high-risk" in the work control process (WC-AA-104).
2. Activities similar in nature to those "example" activities discussed in Attachment 1.

4.1.2. The individual evaluating an activity to determine if it is a special test or evolution shall **USE** Attachment 1 and Attachment 2 to determine if special test or evolution controls are required.

H95

4. **MAIN BODY**

Top Level



Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		3
	K/A #	G2.2.18	
	Importance Rating		3.9

Knowledge of the process for managing maintenance activities during shutdown operations, such as risk assessments, work prioritization, etc.

Proposed Question: SRO 96

The plant is in Cold Shutdown for a forced outage.

You are approving work to be performed during the outage.

Which one of the following describes an example of systems/components which would become an "Operation With Potential To Drain The Reactor Vessel" (OPDRV) if NOT isolated by a barrier IAW HC.OM-AP.ZZ-0001 "Shutdown Safety Management Program"?

A. SRVs

B. HPCI

C. RPV Instrumentation

D. Reactor Water Cleanup

Proposed Answer: D

Explanation (Optional): HC.OM-AP.ZZ-0001 Step 7.2

D. Correct.

A. Incorrect. This would be considered an Operation with a Potential for Draining the **Reactor Cavity as defined in step 7.3 of the procedure**

B. Incorrect. This would be considered an Operation with a Potential for Draining the **Reactor Cavity as defined in step 7.3 of the procedure**

C. Incorrect. This would be considered an Operation with a Potential for Draining the **Reactor Cavity as defined in step 7.3 of the procedure**

Technical Reference(s) HC.OM-AP.ZZ-0001 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: _____ (As available)

Question Source: Bank # _____
Modified Bank # _____ (Note changes or attach parent)
New X

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 _____
55.43 2

Comments:

Involves opening piping with an inner diameter of greater than one (1) inch or multiple pipes which, when summed, exceed an inner cross-sectional area equivalent to one (1) inch diameter pipe. Exceptions to this are equipment that:

a) Is protected by at least one isolation valve (and associated functional instrumentation and logic) capable of automatically closing on a low reactor water level signal; or

b) is isolated from the RPV by at least one approved device controlled in the required position. Approved devices are closed manual valves, back seated valve, or a deactivated automatic valve secured in the closed position. Any other device must be recommended for approval by SORC; or

c) Has a MOV capable of being operated from the control room; or

d) Has a valve capable of being closed locally by a dedicated equipment operator or technician, as appropriate, with no concurrent duties who are maintaining communications with the control room.

The following systems/components are examples of those, which could become subject to OPDRVs if not isolated by a barrier:

- Reactor Water Cleanup
- Reactor Recirculation
- Residual Heat Removal (RHR)
- Control Rod Drive (CRD)
- Reactor Recirculation Sample

2. Involves control rod drive mechanism (CRDM) removal unless:

a) Prior to complete removal of the CRDM, communications between the under vessel area and the control room are established and it is verified that the control rod is back seated against the CRDM guide tube. Following removal of the CRDM (if a CRDM is not immediately placed in the guide tube), a blank flange is placed over the removed CRDM guide tube; and

b) Any Control Rod Blade Exchanges performed concurrently with CRDM Removal shall be performed in accordance with HC.OP-SP.BF-0001(Q), Control Rod Drive Mechanism/Blade Simultaneous Removal and shall be considered an Infrequently Performed Activity or Evolution in accordance with HU-AA-1211 Conduct of Infrequently Performed Activities.

7.3 **Hope Creek Operations with a Potential for Draining the Reactor Cavity (OPDRC)** - An OPDRC is any operations or maintenance activity that:

1. Has the potential to uncover irradiated fuel when the Fuel Pool Reactor Cavity Gates are removed, or Core Alterations are in progress; and involves opening piping with an inner diameter of greater than one (1) inch or multiple pipes which, when summed, exceed an inner cross-sectional area equivalent to one (1) inch diameter pipe. Exceptions to this are:

   - Is protected by at least one isolation valve (and associated functional instrumentation and logic) capable of automatically closing on a low reactor water level signal; or
   - Is isolated from the RPV cavity by at least one approved device controlled in the required position. Approved devices are closed manual valves, back seated valve, or a deactivated automatic valve secured in the closed position. Any other device must be recommended for approval by SORC; or
   - Has a motor operated valve capable of being operated from the control room; or
   - Has a valve capable of being closed locally by a dedicated equipment operator or technician, as appropriate, with no concurrent duties who are maintaining communications with the control room.

   The following systems/components are examples of those, which could become subject to OPDRCs if not isolated by a barrier:

   - All systems/components for OPDRVs
   - MSIVs
   - SRVs
   - Main Steam
   - Feedwater
   - Core Spray
   - HPCI
   - RCIC
   - RPV Instrumentation
   - Fuel Pool Cooling

7.4 **ECCS Injection Source** - An RHR or Core Spray subsystem which consists of a minimum of one AVAILABLE pump; and AVAILABLE flow path capable of being lined up from the Torus or the CST to the vessel.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		4
	K/A #	G2.4.6	
	Importance Rating		4.7

Knowledge of EOP mitigation strategies.

Proposed Question: SRO 97

Given the following conditions:

- Reactor Power is at 6%
- Reactor Pressure is being controlled by SRV's at 950 psig
- Reactor Water Level is (-10) inches, slowly lowering
- Drywell Temperature is 355°F, and rising
- Drywell Pressure is 23 psig, and rising
- Suppression Pool Temperature is 115°F, and rising
- Suppression Pool Level is 85 inches, steady
- Suppression Chamber Pressure is 21.7 psig, and rising
- NO operator actions have been taken

Which one of the following action(s) is(are) are required?

A. ONLY initiate Drywell Sprays IAW EOP-102.

B. ONLY initiate Drywell Sprays and Suppression Pool Cooling/Sprays IAW EOP-102.

C. Enter EOP-202 and Emergency Depressurize.

D. Place Suppression Pool Cooling/Sprays in service then Emergency Depressurize IAW EOP-202.

Proposed Answer: C

Explanation (Optional): C. Correct

IAW EOP-102 Step DWT-8, If DW temp cannot be maintained below 340 degrees F., ED is required.

C. Correct.

A. Incorrect. – ED is required

B. Incorrect. – ED is required

D. Incorrect. – All RPV injection must be secured prior to ED (EOP-202, step ED-3)

Technical Reference(s) EOP-102 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: EO102PE007 (As available)

Question Source: Bank # _____
Modified Bank # _____ (Note changes or attach parent)
New X

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 _____
55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		4
	K/A #	G2.4.40	
	Importance Rating		4.5

Knowledge of SRO responsibilities in emergency plan implementation

Proposed Question: SRO 98

An event has occurred at the plant. The TSC and EOF are manned and NOT activated.

IAW NC.EP-EP.ZZ-0102 "Emergency Coordinator Response", which one of the following describes the individual responsible for escalating an emergency event level from a SAE to a GE?

A. The Shift Manager.

B. The Emergency Duty Officer.

C. The Emergency Response Manager.

D. The Site Vice President.

Proposed Answer: A

Explanation: The SM is the Emergency Coordinator until the TSC is ACTIVATED. Until then the SM (as the EC) is responsible for escalating an event

A. Correct.

B. Incorrect. Correct if the TSC was activated and the EOF manned

C. Incorrect. Correct if the EOF was activated

D. Incorrect. Site VP is not a designated EC

Technical Reference(s) NC.EP-EP.ZZ-0102 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: _____ (As available)

Question Source: Bank # _____

Modified Bank # _____ (Note changes or attach parent)

New X

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 _____
55.43 1

Comments:

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NRC

ATTACHMENT 8
Page 2 of 2

2. Direct RAC to "take turnover" of dose assessment from the SRPT _____

3. Contact the Shift Manager and complete EC turnover. _____

   - VERIFY current plant/emergency status
   - DETERMINE which steps in Section 5, (page 3) of this procedure are completed and which steps will be turned over for completion.
   - ENSURE timing and responsibility for next SSCL and/or NRC data sheet are coordinated
   - With SM concurrence, ASSUME the EC function and declare the TSC activated.

2.0 Upon Assuming Emergency Coordinator Duties, the EDO Should:

- ANNOUNCE to the staff that the TSC is activated, the effective time and that you are the emergency coordinator. _____
- ANNOUNCE (over the plant page) "THE TSC IS ACTIVATED AND _____IS THE EMERGENCY COORDINATOR" _____
- NOTIFY ERM of TSC activation and provide a brief status update. _____
- IMPLEMENT Sections 5.2, 5.3, & 5.4 of this procedure. _____

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NRC

ATTACHMENT 8
Page 1 of 2
ACTIVATION of the TSC

Initials

1.0 Prior To TSC Activation (i.e., Before Assuming Emergency Coordinator Duties):

- ESTABLISH and MAINTAIN a chronological log of activity and events. (If needed, obtain TSC Administrative staff member to maintain logbook) _____

- If not already done, OBTAIN a briefing on the status of the emergency from the Shift Manager (SM). Refer to Attachment 2, EC Emergency Status Briefing Form for turnover points of discussion. _____

- DIRECT/ENSURE TSC section leads are making preparations to assume emergency response functions while ensuring adequate staffing: _____
  - Radiological Assessment Coordinator (RAC)
  - Technical Support Supervisor (TSS)
  - Administrative Support Supervisor
  - EPA or CM-1
  - Security

- PERFORM initial briefing to the TSC staff on emergency conditions and the following issues: (Full TSC staff briefing should be done after TSC has activated) _____
  - Plant/emergency conditions – Why are we here?
  - Establish TSC Activation target time – Should be within 75 minutes of Alert or higher classification.

- PREPARE to activate the TSC and ASSUME the duties and responsibilities of the Emergency Coordinator as follows: _____

  Ensure key roles for TSC activation are covered:
  - TSC has either an EDO or TSS to provide command & control
  - TSC has at least one qualified communicator
  - TSC can support Radiological and Engineering assessment

- ACTIVATE the TSC as follows: (Communicator, RP and EC turnovers should be done coincident with each other) _____
  1. Direct TSC communicators to "take turnover" from the Control Room Communicators. Communications turnover **does not** have to be completed prior to TSC activation but should be in progress.

(continued on next page)

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		3
	K/A #	G2.3.12	
	Importance Rating		3.7

Knowledge of Radiological Safety Principles pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high radiation areas, aligning filters, etc.

Proposed Question: SRO 99

A Planned Special Exposure (PSE) is required today for a containment entry.

The individual involved and their supervisor have agreed to the PSE. The individual has never performed a PSE. Their lifetime dose is 2 Rem and has been fully documented.

Which one of the following describes the additional approvals required and the maximum dose that may be received IAW RP-AA-203 "Exposure Control and Authorization"?

A. The maximum dose permitted for the PSE is 3 Rem. It requires additional approvals by the RPM and Plant Manager ONLY.

B. The maximum dose permitted for the PSE is 3 Rem. It requires additional approvals by the RPM, Plant Manager and Site VP.

C. The maximum dose permitted for the PSE is 25 Rem. It requires additional approvals by the RPM and Plant Manager ONLY.

D. The maximum dose permitted for the PSE is 25 Rem. It requires additional approvals by the RPM, Plant Manager and Site VP.

Proposed Answer: D

Explanation (Optional):IAW RP-AA-203, Section 4.3

For a PSE, the max annual NRC limit is treated separately. Therefore 25 rem is permissible. It requires additional approvals by the RPM, Plant Manager and Site VP.

D. Correct.

A. Incorrect. If subtracting the lifetime dose (2 R) from the annual limit (5R) this would be correct. However the PSE is treated separately with a 25 R limit. Site VP is also required for approval .

B. Incorrect. If subtracting the lifetime dose (2 R) from the annual limit (5R) this would be

correct. However the PSE is treated separately with a 25 R limit

C. Incorrect. Site VP is also required for approval.

Technical Reference(s) RP-AA-203, Section 4.3 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: NOH04ADM024C-01 (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 ____
55.43 4

Comments:

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4.2. Authorization To Raise Administrative Dose Control Levels (ADCLs)

4.2.1. **USE** Attachment 1, Dose Control Level Extension Form, or a computerized equivalent, to authorize exposures for adult individuals in excess of 2000 mrem routine TEDE in a year.

4.2.2. A supervisor from the department requesting approval shall complete Section I of Attachment 1 and submit the request to the Radiation Protection Department indicating:

- The name, identification number, and signature of the individual for whom a dose extension is being requested.
- Whether or **not** other qualified individuals with lower dose are available to perform the work.
- A detailed explanation of why the dose extension is necessary.
- The requested annual TEDE limit for the individual (expressed in 500 mrem increments, i.e. 2500 mrem, 3000 mrem, etc.)

4.2.3. The Radiation Protection Department shall complete Section II Attachment 1 or computerized equivalent.

4.2.4. Pending investigations or calculations of internal exposure shall be reviewed and evaluated to determine the individuals TEDE.

NOTE: An individual shall **not** be approved to receive greater than 2000 mrem TEDE if that person has any absent/ **no** record (A) dose equivalent for the year.

4.2.5. To raise the ADCL to 3000 mrem TEDE in a calendar year, written approval is required by the Radiation Protection Manager and the work group supervisor.

4.2.6. To raise the ADCL to 4000 mrem TEDE in a calendar year, written approval is required by the Radiation Protection Manger, a work group supervisor, and the Station/Plant Manager.

NOTE: An individual being considered for approval greater than 4000 mrem TEDE for the year should **not** have significant estimated dose equivalent.

4.2.7. To raise the ADCL above 4000 mrem, **not** to exceed 5000 mrem, written approval is required by the Site Vice President.

4.3. Planned Special Exposures (PSEs)

NOTE: PSEs are **not** the same as emergency doses. PSEs only apply to adult workers.

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4.3.1. PSEs are to be authorized only in exceptional situations when alternatives that might avoid the dose estimated to result from the PSE are **not** available or are deemed impractical.

4.3.2. A manager from the department requesting a PSE shall submit a request to the Radiation Protection Department, indicating:

- The name and identification number of each individual for whom a PSE is being requested, and
- The nature of the task for which a PSE is being requested, and
- A detailed explanation of why the PSE is necessary.

4.3.3. Prior written approval is required before the PSE occurs.

4.3.4. Prior to participating in a PSE, individuals involved shall be:

1. Informed of the purpose of the PSE.
2. Informed of the estimated dose and associated potential risk or conditions involved in performing the PSE.
3. Instructed in dose reduction measures and techniques for the PSE.

4.3.5. PSE approval is granted when the PSE document is signed (by hand) and dated by:

- The individual(s) for whom a PSE is being requested, and
- The work group manager or other level of supervisory authority for the individual as chosen by the RPM or designee, and
- The RPM or designee, and
- The Plant Manager, and
- The Site Vice President.

NOTE: If there are any periods of exposure during the life of the monitoring individual that have **not** been determined or documented (i.e., Absent/ **No** record), then participation in a PSE is **not** permitted.

4.3.6. All of the individual's previous PSE dose equivalents and previous doses in excess of routine occupational limits must be determined from records for each individual who will participate in the PSE. Doses received in excess of the routine occupational dose limits in effect at the time of exposures during accidents and emergencies must also be determined and subtracted from the limits for PSEs.

4.3.7. **DOCUMENT** each individual's current year and previous years:

- PSE dose equivalents, and
- Dose equivalents in excess of the exposure limits in effect at the time of the exposures (rows (1) and (2) of Table 1, "NRC Exposure Limits," and the former 10 CFR 20.101), and

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3.1. Responsibility for approval of exposures in excess of administrative dose control levels resides with the Radiation Protection Manager (RPM), the Station/Plant Manager, and the Site Vice President.

4. MAIN BODY

4.1. Limitations

4.1.1. Exposures shall **not** exceed the 10CFR20 Exposure Limits as described in Table 1, "NRC Exposure Limits."

TABLE 1 - NRC EXPOSURE LIMITS

Individual and Limit Type	TEDE	LDE	SDE	TODE
(1) Occupational Worker, Minor, Routine Annual	0.5 rem	1.5 rem	5 rem	5 rem
(2) Occupational Worker, Adult, Routine Annual	5 rem	15 rem	50 rem	50 rem
(3) Occupational Worker, Adult, PSE Annual	5 rem	15 rem	50 rem	50 rem
(4) Occupational Worker, Adult, PSE Lifetime	25 rem	75 rem	250 rem	250 rem
(5) Declared Pregnant Woman	Dose equivalent to the embryo/fetus of 500 mrem, or 50 additional mrem if the dose equivalent to the embryo/fetus exceeds 450 mrem at the time the female declares, in writing, her pregnancy. (Note: The dose equivalent to the embryo/fetus equals the DDE to the declared pregnant woman plus the dose equivalent to the embryo/fetus from radionuclides in the embryo/fetus plus the dose equivalent to the embryo/fetus from radionuclides in the declared pregnant woman.)			
(6) Member of The Public	The total effective dose equivalent to individual members of the public shall **not** exceed 100 mrem in a year.			

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		1
	K/A #	G2.1.34	
	Importance Rating		3.5

Knowledge of primary and secondary plant chemistry limits.

Proposed Question: SRO 100

Given the following conditions:

- A reactor shutdown is in progress.
- Power is currently 20%.
- Hydrogen Water Chemistry Injection (HWCI) is out of service.
- Main Steam Line RMS Setpoints are set High.
- 2 Condensate Demineralizers are in service at 3000 gpm each.
- Plant chemistry parameters are as follows:
  - Condensate demin influent conductivity is 0.21 uS/cm
  - Condensate demin effluent conductivity is 0.08 uS/cm
  - Reactor Water Cleanup conductivity is 0.07 uS/cm
  - Reactor coolant sample conductivity is 0.07 uS/cm
  - Reactor coolant specific activity is 1.0 $X10^{-3}$ uci/gm Dose Equivalent Iodine

Based on these conditions, which one of the following would cause these indications and what actions must be taken IAW AB-RPV-0007 "Reactor Coolant Conductivity"?

A. Crud burst due to removing HWCI from service; restore HWCI to service.

B. Main Condenser tube leak; isolate the affected condenser waterbox.

C. Reactor fuel pin cladding leak; continue power reduction at normal rate.

D. Condensate Demineralizer channeling; remove one demineralizer from service.

Proposed Answer: B

Explanation (Optional):

B. Correct. Main Condenser tube leak; isolate the affected condenser waterbox. Conductivity into the Cond Demins is high. This is a symptom of a Condenser tube Leak. Required action would be to remove the waterbox IAW AB-RPV-0008.

A. Incorrect. Crud burst from removing HWCI from service; restore HWCI to service. RWCU and Reactor coolant conductivity levels are normal.

C. Incorrect. Reactor fuel pin cladding leak; continue power reduction at normal rate. Power reduction at normal rate not permitted due to MSL RMS setpoints are set high. Indications are not cause for emergency power reduction.

D. Incorrect. Condensate Demineralizer channeling due to low flow; remove one demineralizer from service. Demineralizer outlet conductivity is normal. Would have low inlet and high outlet conductivity.

Technical Reference(s) HC.OP-AB.RPV-0007 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: HWCI00E006 (As available)

Question Source: Bank # 80628
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam 2003

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 _____
55.43 6

Comments:

SUBSEQUENT OPERATOR ACTIONS

CONDITION	ACTION
A. Rising Condensate Conductivity	☐ A.1 **MONITOR** Reactor Coolant Conductivity.
Time: _____	A.2 **VERIFY** proper operation of the Condensate Demineralizers (CRIDS Page 24).
	☐ • Demin flow 3000-6000 gpm
	☐ • Outlet Conductivity ≤ 0.060 µS/cm
	☐ • Cond. Demineralizer Inlet Temp. <135°F.
	A.3 IF CDI Conductivity is ≤ 0.1 µS/cm THEN **PERFORM** the following:
Condensate cond: _____	☐ • **MONITOR** Main Condenser Drip Tray AND hotwell conductivities for indications of a Main Condenser Tube Leak. (CRIDS Pg. 19)
	☐ • **DIRECT** Chemistry to implement HC.CH-GP.ZZ-0009.
	A.4 IF CDI Conductivity is > 0.1 µS/cm THEN **PERFORM** the following:
	☐ **NOTE 1**
	☐ **NOTE 2**
	☐ a. **VERIFY** via Main Condenser Drip Tray AND Hotwell Conductivities in suspected waterbox.
	☐ b. **ISOLATE** AND **DRAIN** the suspected waterbox per Condition D.
	☐ A.5 IF Main Condenser in-leakage is not indicated, THEN **DIRECT** Chemistry to Sample AND Analyze condensate to determine the source of the rising conductivity.
	☐ A.6 **DIRECT** Radwaste to verify normal Glycol Cooler Conductivity.