

Oconee Nuclear Station 2013-301 Written Examination

Question 1

Given the following Unit 1 conditions:

- The Reactor tripped from 100% power
- The EOP (EP/1/A/1800/001) has been initiated
- An excessive cooldown occurred
- ES-1 and ES-2 has had a valid actuation
- The cooldown has been stopped and RCS temperature is 528 °F and slowly rising
- Indicated pressurizer level is 255 inches and increasing.

Which ONE of the following describes the first procedure to be used to control pressurizer level, and the indicated pressurizer level that will first require entry into Technical Specification 3.4.9 (Pressurizer) ACTION statement?

- A. EOP Rule 6 HPI: 265 inches
- B. Enclosure 5.5 Pzr and LDST Level Control ; 265 inches
- C. EOP Rule 6 HPI; 285 inches
- D. Enclosure 5.5 Pzr and LDST Level Control ; 285 inches

Proposed Answer: A

Explanation (Optional):

1. Per LCO 3.4.9, The pressurizer shall be OPERABLE with pressurizer level less than or equal to 285 inches and a minimum of 400kW of pressurizer heaters OPERABLE and capable of being powered from an emergency power supply in MODES 1 and 2, and MODE 3 with RCS temperature greater than 325 degrees F.
2. Per the PNS-PZR lesson plan and note above step 4.9 of AP/1/A/1700/032 (Loss of Letdown), Tech Spec 3.4.9 applies when indicated pressurizer level is greater than 260 inches.
3. Per OMP 1-18 (Implementation Standard during Abnormal and Emergency Events) Attachment A step 1.17, a manual reactor trip must be initiated if pressurizer level is greater than or equal to 375 inches.
4. Per the PNS-PZR lesson plan, the pressurizer high level statalarm setpoint is 260 inches.
5. Per the PNS-PZR lesson plan, the pressurizer emergency high level statalarm setpoint is 315 inches.

Technical Reference(s): Lesson Plan ADM-TSS Rev.10a
(Attach if not previously provided) Lesson Plan PNS-PZR Rev. 21
(including version/revision number) Tech Spec LCO 3.4.9
OMP 1-18 Rev. 033
AP/1/A1700/032 Rev.006

Proposed references to be provided to applicants during examination: None

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Learning Objective: Lesson Plan ADM-TSS Obj. R4
Lesson Plan PNS-PZR Obj. R25

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

Question History: Last NRC Exam _____
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 X
55.43 _____

Comments:
000007 (BW/E02&E10; CE/E02) Reactor Trip - Stabilization - Recovery / 1

007EG2.2.42 Ability to recognize system parameters that are entry-level conditions for Technical Specifications.

Author: Dan Bacon

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Question # 2

Unit 1 Initial Conditions:

- Core Thermal Power = 100%.

Current Conditions:

- 1RC-1 indicates open.
- 1RC-66 indicates open.
- Core Thermal Power = 95% and decreasing.
- RCS Wide Range pressure = 2190 psig and decreasing.
- 1SA-18 A-1 Pressurizer Relief Valve Flow Statalarm is locked in.
- 6 LED lights are illuminated for 1RC-66 on the Pressurizer Relief Valve Flow Monitor.

Based on the current conditions, which ONE of the following completes the statements listed below?

In accordance with the immediate manual actions of AP/1/A/1700/044, (1) is required to be closed at this time.

And

This transient could have been caused by the failure of a median selected controlling narrow range RCS pressure signal being supplied from RPS Channel (2).

- A. 1) only RC-4
2) E
- B. 1) only RC-4
2) C
- C. 1) RC-1, RC-3 and RC-4
2) B
- D. 1) RC-1, RC-3 and RC-4
2) A

Proposed Answer: A

Explanation (Optional):

1. Per the Immediate Manual Actions steps of AP/1/A/1700/044 (Abnormal Pressurizer Pressure Control):

3.1 **IAAT** all of the following conditions exist:

 PORV open

 RC pressure < 2300 psig (HIGH) or 480 psig (LOW)

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___ PZR level \leq 375"
THEN close 1RC-4.

3.2 ___ **IAAT** all of the following conditions exist:

___ RC pressure $<$ 2155 psig

___ RC pressure decreasing without a corresponding decrease in PZR level

___ PZR heaters unable to maintain RCS pressure.

THEN close the following:

___ 1RC-1

___ 1RC-3

2. OMP 1-18 (Implementation Standard During Abnormal and Emergency Events) Appendix A Step 1.3 lists EOP/AP Immediate Manual Actions and Contingency Actions as a Licensed Operator Memory Item.
3. Narrow Range pressure transmitters (1700-2500 psi) (RPS, ICS feeds)
 - a) Five total Narrow Range RCS Pressure transmitters and are located on Loop A (2) and Loop B (3)
 - 1) Channel A-D RPS Channels (A&B from Loop A, C&D from B)
 - (a) High and Low RC pressure trips (ITS High 2355/Low 1800 – ACTUAL 2345/1810)
 - 2) Median selected signal from RPS channels “A”, “B”, and “E” are used for:
 - (a) RC-66, PORV "HIGH" opening circuit (2450 psig)
 - (b) RC-1, PZR SPRAY AUTO circuit (Open-2205/Shut-2155)
 - (c) Pressurizer Heater AUTO circuit (1-2155 variable, 2-on 2140/off 2150, 3-on 2145/off 2175, 4-on 2130/off 2145)
 - (d) NR Auxiliary Shutdown Panel indication (1700-2500)
 - (e) NR recorder (1700-2500)
 - (f) ICS (high pressure relief to reactor, FDW, and turbine sections of the ICS).
 - 3) Channel E RPS Channel is used for control functions via median select. Channel A and B are used as the other two inputs to median select for control output and RPS inputs.
4. Information also contained in section 4G of AP/1/A/1700/028 (ICS Instrument Failures).

Technical Reference(s):	AP/1/A/1700/044 Rev. 002
(Attach if not previously provided)	AP/1/A/1700/028 Rev. 019
(including version/revision number)	EAP-APG Lesson Plan Rev. 07d
	EAP-APG-AP44 Lesson Plan Rev. 01a
	IC-RCI Lesson Plan Rev. 22c
	OMP 1-18 Rev. 033

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Proposed references to be provided to applicants during examination: None

Learning Objective: EAP-APG-AP44 Lesson Plan Obj. R9
IC-RCI Lesson Plan Obj. R7, 9, 10, 32, 33, 62 & 63

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

Question History: Last NRC Exam _____
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Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X
55.43 _____

Comments:
000008 Pressurizer Vapor Space Accident / 3

008AG2.4.45 Ability to prioritize and interpret the significance of each annunciator or alarm.

Author: Dan Bacon

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Question # 3

Unit 1 Initial Conditions:

- Time = 0800.
- SCM = 0° F.
- Reactor Power is < 1%.
- A Small Break LOCA is in progress.
- EOP Immediate Manual Actions are complete.
- EOP Enclosure 5.1 (ES Actuation) is in progress.

Current Conditions:

- Time = 0804.
- SCM = 0° F.
- RCPs 1A1, 1A2, 1B1 and 1B2 are running.
- EOP Rule 2 (Loss of SCM) is initiated.
- ES Channel 1 failed to go to manual.
- The ES ODD Voter is in OVERRIDE.
- ES Channel 3 was manually initiated.
- RCS WR Pressure is 580 psig and stable.

Based on the current conditions, which ONE of the following completes the statements listed below?

In accordance with EOP Rule 2, RCPs should be (1) by the operator at the current time.

And

In accordance with EOP Enclosure 5.1, ES Channel 3 must (2) in order to stop 1A LPI pump.

- A. (1) stopped
(2) be reset, and then placed in manual
- B. (1) stopped
(2) be placed in manual only
- C. (1) left running
(2) be reset, and then placed in manual
- D. (1) left running
(2) be placed in manual only

Proposed Answer:

C

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Explanation (Optional):

1. Per EOP Rule 2:

- B. **IAAT** all exist:
 Any SCM $\leq 0^{\circ}\text{F}$
 Rx power $\leq 1\%$
 ≤ 2 minutes elapsed since loss of SCM
THEN perform Steps 2 and 3. {9}

1. **IF** all SCMs $> 0^{\circ}\text{F}$,
THEN:
A. Obtain CR SRO concurrence to exit Rule 2.
B. **EXIT** Rule 2.
2. Notify CR SRO of RCP status. {9, 22}
3. **GO TO** Step 4.

2. Stop all RCPs.

1. Place 1TA AUTO/MAN switch in MAN.
2. Place 1TB AUTO/MAN switch in MAN.
3. Open 1TA SU 6.9 KV FDR.
4. Open 1TB SU 6.9 KV FDR.

2. Per EOP Enclosure 5.1:

9. **IAAT** all exist:

- Voter associated with ES channel is in OVERRIDE
 An ES channel is manually actuated
 Components on that channel require manipulation

THEN depress RESET on the required channel.

17. Perform both:

- Place ES CH 3 in MANUAL.
 Place ES CH 4 in MANUAL.

NOTE

- Voter OVERRIDE affects all channels of the affected ODD and/or EVEN channels.
- In OVERRIDE, all components on the affected ODD and/or EVEN channels can be manually operated from the component switch.

1. **IF** ES CH 3 fails to go to MANUAL, **THEN** place ODD voter in OVERRIDE.
2. **IF** ES CH 4 fails to go to MANUAL, **THEN** place EVEN voter in OVERRIDE.

3. Matches K/A due to SCM being associated with both temperature and pressure.

4. 1A LPI pump is started by ES Channel 3.

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Technical Reference(s): EP/1/A/1800/001 Rev. 039 Rule 2 and Enclosure 5.1
(Attach if not previously provided) _____
(including version/revision number) _____

Proposed references to be provided to applicants during examination: None

Learning Objective: EAP-LOSCM Lesson Plan Att. 1 Rev. 3b Obj. R3
IC-ES Lesson Plan Rev.18a Obj. R29

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

Question History: Last NRC Exam _____
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 X
55.43 _____

Comments:
000009 Small Break LOCA / 3

**009EA1.01 Ability to operate and monitor the following as they apply to a small break
LOCA: RCS pressure and temperature**

Author: Dan Bacon

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Question # 4

Given the following Unit 3 conditions:

- Core Thermal Power = 100%
- Core Flood Tank (CFT) sampling is in progress per OP/3/A/1104/001(Core Flooding System)

Unit 3 current conditions:

- 3SA-8 A-11(Core Flood Tank A Pressure High/Low) is in alarm
- 3SA-8 B-11(Core Flood Tank A Level High/Low) is in alarm
- 3SA-8 A-12(Core Flood Tank B Pressure High/Low) is in alarm
- 3SA-8 B-12(Core Flood Tank B Level High/Low) is in alarm
- CFT level/pressure adjustment in progress per OP/3/A/1104/001(Core Flooding System)
- 3A CFT Level = 13.56 feet
- 3A CFT Pressure = 580 psig
- 3B CFT Level = 12.70 feet
- 3B CFT Pressure = 585 psig
- A Reactor trip occurs due to a Large Break LOCA

Based on the current conditions, which ONE of the following statements below describes the effects resulting from 3A and 3B Core Flood Tank conditions?

- A. 3A CFT will discharge an inadequate volume of water into the core due to the CFT level.
- B. 3A CFT will discharge an inadequate volume of water into the core due to the CFT pressure.
- C. 3B CFT will discharge an inadequate volume of water into the core due to the CFT level.
- D. 3B CFT will discharge an inadequate volume of water into the core due to the CFT pressure.

Proposed Answer: A

Explanation (Optional):

Answer Analysis

- A. Correct. Level is above TS maximum allowed level of 13.44 feet.
- B. Incorrect. Plausible because pressure is below low alarm setpoint, but above minimum TS required value of 575 psig.
- C. Incorrect. Plausible because level is at low alarm setpoint, but above minimum TS required value of 12.56 feet.
- D. Incorrect. Plausible because pressure is at low alarm setpoint, but above minimum TS required value of 575 psig.

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1. From PNS-CF Lesson Plan:

(Obj. R17) Improper Level/Pressure Combinations

The **minimum volume requirement** for the CFTs ensures that both CFTs can provide adequate inventory to re-flood the core (to the hot spot) and downcomer following a LOCA. The downcomer then remains flooded until the HPI and LPI systems start to deliver flow.

The specified limit is 1010 ft³. The corresponding CFT level is 12.56 ft.

The **maximum volume limit** is based upon:

The need to maintain adequate gas volume to ensure proper injection to ensure the ability of the CFTs to fully discharge and therefore cover the core hot spot on re-flood

The specified limit is 1070 ft³. The corresponding CFT level is 13.44 ft.

Limit the maximum amount of boron inventory in the CFTs.

Limiting the total amount of boron by limiting CFT level limits the amount of boron precipitate that occurs following a LOCA.

The **minimum nitrogen cover pressure** requirement ensures that the contained gas volume will generate discharge flow rates during injection that are consistent with those assumed in the safety analysis.

The specified limit is 575 psig.

Liquid volume affects gas volume. The liquid and gas volumes and the gas pressure are calculated for the CFT line size to ensure that the minimum gas pressure, at an adequate volume of gas, will provide flow rates such that there will be enough gas volume at enough pressure to force enough water into the core during the re-flood stage to cover the core hot spot.

The **maximum nitrogen cover pressure** requirement ensures that the amount of CFT liquid inventory that is discharged while the RCS depressurizes, and is therefore lost through the break during the blow down phase, will not be larger than that predicted by the safety analysis.

The specified limit is 625 psig.

If CFT pressure were too high (with proper gas volume), too much water would be discharged out the break during the blow-down phase, leaving an inadequate amount of water left to recover the core hot spot during the reflood phase

Examples:

Since the volume of gas as well as gas pressure is important to ensure enough water is injected to re-cover the core hot spot. It is obvious that if both gas pressure and liquid volume are out of spec high, it may nearly be impossible to determine with any degree of certainty what the outcome of this injection may be. Students are not expected to perform $P_1V_1 = P_2V_2$ calculations.

If liquid volume was high out of spec, gas pressure would have to be extremely high (the same amount of gas molecules as at normal volume and pressure) to off-set the low gas volume due to the high liquid volume. The gas pressure required to produce the required flow rates is not obvious.

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If liquid volume was high out of spec and gas pressure is low out of spec or normal (the same amount of gas molecules as at normal volume and pressure), it is obvious that not enough liquid would be injected to cover the core hot spot even at the lower flow rates. This is essentially just a low gas pressure.

If liquid volume was low out of spec and gas pressure was low out of spec (the same amount of gas molecules as at normal volume and pressure), it is obvious that not enough liquid would be available to cover the core hot spot even at the lower flow rates. This is essentially just a low liquid volume or a low gas pressure.

If liquid volume was low out of spec and gas pressure was high normal or out of spec high (More gas molecules than at normal volume and pressure), it is obvious that not enough liquid would be available to cover the core hot spot during the re-flood stage since more liquid would be blown out the break during the blow-down phase. This is essentially just a low liquid volume or a high gas pressure.

(Obj. R5) Establishing ES Conditions (Enclosure of OP/A/1104/001)

To establish ES conditions, the following must be established and verified:

CFT water levels between 12.7 and 13.3 ft.

CFT boron between 2500 - 3750 ppmb

TS required boron concentration varies depending on the unit (defined in the COLR).

CFT pressure between 585 -615 psig

When RCS pressure 730-750 psig:

Open CF-1 & 2 (CFT Outlets)

Visually verify open CF-1 and CF-2 (locally).

Close CF-5 & 6 (CFT Vents)

Lock OPEN and white tag the power supplies to CF-1 and CF-2.

Lock OPEN and white tag the breakers for CF-5 & 6.

- Procedural step only not required per TS.

Instrumentation

1. CFT Pressure: Each tank is provided with two pressure detectors. Both of these detectors indicate in the Control Room. The range of each indication is 0 to 700 psig.
2. CFT Level: Each tank is provided with two differential pressure detectors. Both of these detectors indicate in the Control Room. The range of each indicator is 1.5 - 14 feet.

Alarm Setpoints:

CFT Pressure 585 - 615 psig

CFT Level 13 ± .3 feet

Alarms:

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1. SA-8/A-11 and 12, CFT Pressure High/Low (615 psig increasing / 585 psig decreasing)
2. SA-8/B-11 and 12, CFT Level High/Low Alarm - (13.3 feet increasing, 12.7 feet decreasing)
3. SA-8/C-11, CFT "A" Outlet Valve - "Not Open"
4. SA-8/C-12, CFT "B" Outlet Valve - "Not Open"

Technical Reference(s): TS 3.5.1
(Attach if not previously provided) PNS-CF Lesson Plan Rev. 15e
(including version/revision number) OP/3/A/1104/001 Rev. 075

Proposed references to be provided to applicants during examination: None

Learning Objective: PNS-CF Lesson Plan R3, R5 and R17

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

Question History: Last NRC Exam 2012-301 (ILT 41) Question 31
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 X
55.43 _____

Comments:
000011 Large Break LOCA / 3

011EG2.2.44 Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.

Author: Dan Bacon

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ILT41 ONS SRO NRC Examination QUESTION 31 D 31

Given the following Unit 3 conditions:

Reactor power = 100%

"3A" Core Flood Tank

Pressure = 587 psig

Level = 12.87 ft

"3B" Core Flood Tank

Pressure = 629 psig

Level = 13.36 ft

Which ONE of the following describes the potential adverse effects if a large break

LOCA were to occur?

- A. 3A CFT will discharge an inadequate volume of water into the core due to the CFT level.
- B. 3A CFT will discharge an inadequate volume of water into the core due to the CFT pressure.
- C. 3B CFT will discharge too much inventory during the blow down phase and not cover the hotspot during re-flood due to CFT level.
- D. 3B CFT will discharge too much inventory during the blow down phase and not cover the hotspot during re-flood due to CFT pressure.

SYS006 K6.02 - Emergency Core Cooling System (ECCS)

Knowledge of the effect of a loss or malfunction on the following will have on the ECCS: (CFR: 41.7 / 45.7)

Core flood tanks (accumulators)

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ILT41 ONS SRO NRC Examination QUESTION 31 D 31

General Discussion

Answer A Discussion

Incorrect.. Plausible because level is low but within the TS range.

Answer B Discussion

Incorrect.. Plausible because pressure is low but within the TS range.

Answer C Discussion

Incorrect. Plausible because level is high but within the TS range.

Answer D Discussion

Correct. The CFT pressure is high > 625 psig. This can cause the CFT to discharge too soon and not cover the hot spot during re-flood.

Cognitive Level

Comprehension

Job Level

RO

QuestionType

NEW

Question Source

Development References Student References Provided

PNS-CF R17

401-9 Comments: Remarks/Status

Basis for meeting the KA

Question requires knowledge of how CFT conditions (pressure and level) can affect ECCS.

Basis for Hi Cog

Basis for SRO only

SYS006 K6.02 - Emergency Core Cooling System (ECCS)

Knowledge of the effect of a loss or malfunction on the following will have on the ECCS: (CFR: 41.7 / 45.7)

Core flood tanks (accumulators)

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Question # 5

Given the following Unit 1 conditions:

- Core Thermal Power = 100%.

Current conditions:

- A Station Blackout occurs at 0600.
- AP/0/A/1700/025 (Standby Shutdown Facility Emergency Operating Procedure) has been initiated.
- 1XSF is being powered from 0XSF.

Based on the current conditions, which ONE of the following completes the statements listed below?

In accordance with station Time Critical Actions, SSF RCMU flow must be established to Unit 1 RCP seals no later than (1).

And

1HP-20 (RCP Seal Return) (2) be operated from Unit 1 Control Room at this time.

- A. (1) 0614
(2) can
- B. (1) 0620
(2) can
- C. (1) 0614
(2) cannot
- D. (1) 0620
(2) cannot

Proposed Answer: D

Explanation (Optional):

1. It is a Time Critical Action to commence RCMU flow to the affected units' RCP seals within 20 minutes of an event that causes a loss of HPI seal injection and a loss of CC to the RCPs.
2. It is a Time Critical Action to commence SSF-ASW flow to the affected units' steam generators within 14 minutes of a LOHT event.
3. RC-5, RC-6, HP-3, HP-4, HP-20 and RC-4 will not receive an ES signal, will lose indicating lights in the Unit control room and cannot be operated from the Unit control room after 1,2,3XSF MCCs have been swapped to their alternate supply (0XSF).

Technical Reference(s): Lesson Plan EAP-SSF Rev. 24f

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(Attach if not previously provided) Lesson Plan EAP-TCA Rev. 05b
(including version/revision number) AP/0/A/1700/025 Rev. 055

Proposed references to be provided to applicants during examination: None

Learning Objective: Lesson Plan EAP-TCA Obj. R14
Lesson Plan EAP-TCA Obj. R30

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

Question History: Last NRC Exam _____
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Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X
55.43 _____

Comments:
000015/17 RCP Malfunctions / 4

015AK2.07 Knowledge of the interrelations between the Reactor Coolant Pump Malfunctions (Loss of RC flow) and the following: RCP seals

Author: Dan Bacon

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Question # 6

Given the following Unit 1 Initial Conditions:

- Core Thermal Power = 100%.
- SASS is in manual.
- ICCM Train B is off-line for maintenance.
- PZR Level Select Pushbutton #2 is selected on 1UB1.
- 1SA-02 C-3 RC PZR Level High/Low Statalarm is in alarm.
- 1SA-02 C-4 RC PZR Level Emergency High/Low Statalarm is in alarm.
- PZR level #2 Dixon meter on 1UB1 is failed high.
- PZR level indicated by properly indicating PZR level Dixon meter on 1UB1 is 215 inches and decreasing.
- 1HP-120 (RC Volume Control) is in automatic and fully closed.

Unit 1 Current Conditions:

- AP/1/A/1700/014 (Loss of Normal HPI Makeup AND/OR RCP Seal Injection) has been initiated.
- PZR Level is being controlled at 220 inches with 1HP-120 in HAND.

Based on the current conditions, which ONE of the following completes the statements listed below?

A condition that would allow 1HP-120 to be placed back in AUTO would be selecting PZR Level Select Pushbutton # (1) .

And

After the appropriate PZR Level Select Pushbutton is selected, the PZR Emergency High/Low Statalarm will (2).

- A. (1) 1
(2) clear
- B. (1) 3
(2) clear
- C. (1) 1
(2) remain in alarm
- D. (1) 3
(2) remain in alarm

Proposed Answer: A

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Explanation (Optional):

1. From AP/1/A/1700/014:

1. Entry Conditions

1.1 **NO** HPI pumps operating

1.2 Complete loss of seal injection to all RCPs

1.3 1HP-120 failure to respond to Pzr level decreasing outside of control band

1.4 Directed entry from another procedure

4.12 ___ Verify 1HP-120 operable in AUTO.

1. ___ Attempt to operate 1HP-120 in HAND.

2. ___ **IF** 1HP-120 fails to operate,
THEN GO TO Step 4.176.

3. ___ Maintain desired Pzr level with 1HP-120 in HAND.

4.13 ___ **WHEN** conditions permit,
THEN EXIT this procedure.

2. From PNS-PZR lesson plan:

A. PZR Level Control

1. PZR level is controlled during normal operation by an air operated control **valve HP-120**, RC Volume Control, located in the east penetration room. A Bailey Hand/Auto station on UB1 is used to maintain level automatically or manually. The level taps are not at the true top and bottom of the PZR, so the level can indicate off scale high or low and there will still be some volume left.

2. HP-120 is provided with a **manual bypass, HP-122**, which can be used to maintain level in the event of a failure of HP-120.

3. Level **statalarms**:

a) Emergency High Level Alarm = 315 inches

b) High Level Alarm = 260 inches

c) Low Level Alarm = 200 inches

d) Emergency Low Level Alarm = **80 inches**

4. TS 3.4.9, Pressurizer, maximum allowed level is **285"**. Operations guidance is to require entry into TS 3.4.9 when 260" is reached. This is done to allow (based on instrument error) the use of any available C/R indication for PZR Level.

5. The PZR Heaters are interlocked to automatically de-energize at **80 inches (85" SSF)**.

B. PZR Level and Temperature Instrumentation

1. Level and temperature instrumentation consists of four QA-1 level transmitters and one QA-1 three element RTD.

2. Uncompensated level channels A and B are fed through ICCM Train A and uncompensated level channel C is fed through ICCM Train B.

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- a) RCLT0004P1 (level channel 1) and RCLT0004P2 (level channel 2) feeds ICCM cabinet A.
 - b) RCRD0043A (temperature channel A) feeds ICCM cabinet A.
 - c) RCLT0004P3 (level channel 3) feeds ICCM cabinet B.
 - d) RCRD0043B (temperature channel B) feeds ICCM cabinet B.
 - e) RCRD0043C (temperature channel C) feeds the OAC through ICS and provides input for the calculation of PZR Saturation Pressure for the Saturation Recovery Unit associated with Bank 2 (Grps B & D) heaters.
3. The fourth level transmitter (RC-LT-0072), shares a reference leg with RC-LT-004P2 (PZR Level 2 input into ICCM Train A). The level transmitters have separate variable legs.
- a) RC-LT-0072 feeds the level signal (0-600 inches) to the SSF.
 - b) A pressure transmitter (RC-PT-224) taps off of the reference leg for the two level transmitters and feeds the PZR Pressure signal to the SSF.
 - c) If the SSF Level Transmitter is removed from service, the Level 2 Transmitter is not affected unless the action is taken with the reference leg. This will affect the indication seen from the Level 2 Transmitter.
 - 1) Example: If the equalization valve for the SSF Level Transmitter is leaking, or left open, the reference leg will drain to the PZR through the variable leg and a high PZR Level will be indicated.
4. ICCM cabinets provide outputs to both safety and non-safety related components.
- a) Safety related outputs:
 - 1) ICCM cabinet A feeds:
 - (a) Compensated Level #1 to a Dixon gauge on UB1 (PAM, Scale 0 to 400 inches).
 - (b) Compensated Level #2 to a Dixon gauge on UB1 (PAM, Scale 0 to 400 inches).
 - (c) Compensated Level #1 and #2 to a recorder on VB2
 - (d) PZR Temperature A to a Dixon gauge on UB1 (PAM, Scale 50 to 700°F)
 - (e) PZR Temperature A to a recorder on VB2. PZR Temp A (PAM)
 - 2) ICCM cabinet B feeds:

- (a) Compensated Level #3 to a Dixon gauge on UB1 (PAM, scale = 0 to 400 inches).
- (b) PZR Temperature B to a Dixon gauge on UB1 (PAM, scale = 50 to 700°F).
- (c) Non-safety related outputs: (Computer Points)
 - 3) RC Pressurizer Water Temperature A (ICCM Plasma Display)
 - 4) RC Pressurizer Water Temperature B ((ICCM Plasma Display)
 - 5) RC Pressurizer Level 1 (Pressure Compensated)
 - 6) RC Pressurizer Level 1 (Temperature Compensated)
 - 7) RC Pressurizer Level 1 (Uncompensated)
 - 8) RC Pressurizer Level 2 (Pressure Compensated)
 - 9) RC Pressurizer Level 2 (Temperature Compensated)
 - 10) RC Pressurizer Level 2 (Uncompensated)
 - 11) RC Pressurizer Level 3 (Pressure Compensated)
 - 12) RC Pressurizer Level 3 (Temperature Compensated)
 - 13) RC Pressurizer Level 3 (Uncompensated)
 - 14) RC Pressurizer Water Temperature 3

C. PZR Temperature/Pressure Compensation

- 1. The temperature of the water in the PZR varies from near ambient (at cold S/D) to 650° F (at normal operating pressure). This large change in temperature results in a significant density change in the PZR water.
- 2. The temperature, and therefore the density, of the water in the reference legs remains essentially constant.
- 3. As the temperature of the PZR increases during a unit startup, the volume of the water expands and actual level increases. The leg of the level transmitter that is attached to the PZR (the variable leg) is exerting approximately the same pressure (the mass hasn't changed) against the transmitter in relation to the water in the reference leg. **Without compensation PZR level will read lower than actual.**
- 4. To compensate for changes in water density, PZR temperature and RCS pressure are used to provide temperature and pressure compensated PZR level.
- 5. Temperature can be used for accurate level compensation in a saturated system such as the PZR because changes in liquid density can be determined by use of the steam tables.
- 6. RCS pressure can be used similarly. Here, the pressure signal is converted into its corresponding saturation temperature signal, and this temperature signal is then applied for level compensation.
- 7. Level compensation takes place inside the ICCM trains.

D. PZR Level Selection

1. The **PZR Level select pushbutton** located on UB1 is normally used to select which level signal will be used to provide level control functions and provides input for:
 - a) PZR Level recorder on UB1
 - b) Input for HP-120 control
 - c) Input of which temperature compensated PZR level is displayed at the Auxiliary Shutdown Panel.
 - d) **PZR Emergency High/Low Statalarm** – Note: the High/Low level statalarm will alarm if any of the three level indications reach a high or low condition, regardless of which one is selected.
 - e) PZR heaters Low Level Cut-Off
2. **(OBJ.R31)** If the pb selected level fails and the Smart Automatic Signal Selector (SASS) is in automatic, a transfer of control to an unaffected level channel in the other ICCM train will occur.
 - a) SASS provides protection for selected plant parameters against instrument failures by detecting the failure and then selecting an operable alternate instrument.
 - b) SASS will automatically transfer the level signal if a failure is detected in the PZR signal output from ICCM train A (either level signal #1 or #2) or the PZR signal output from ICCM train B (level #3).
 - c) If the operator has PZR level #1 or 2 selected, then SASS input will be from that selected level channel and the second SASS input will be from level #3. Level #3 is always the second SASS input.
 - d) If the operator has level #3 selected, then the second SASS input **defaults** to level #1. Level #2 is never the second SASS input.
 - e) If SASS is de-energized or in Manual, then no automatic signal swapping will occur.
3. **(OBJ.R35)** PZR levels feed through ICCM and the **failure of an ICCM train** can occur with **SASS out of service**. The operator must know how the different failure modes of an ICCM train will affect level control.
 - a) **Loss of ICCM main cabinet incoming power** with the selected level feeding from the failed ICCM train and SASS unavailable
 - 1) Plasma display indicates a **BLANK** screen.
 - 2) Indicated PZR level “**fails LOW**”
 - 3) PZR Heaters cut off on low level.
 - 4) HP-120 opens.
 - 5) PZR “Emergency High/Low” statalarm actuates (Low).
 - 6) PZR “High/Low” statalarm actuates (Low).
 - 7) Aux. Shutdown Panel PZR level indicates low.

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- b) **Loss of ICCM INTERNAL cabinet power** with the selected level feeding from the affected ICCM train and SASS unavailable
- 1) Plasma display indicates **DATA LINK FAILURE**.
 - 2) Indicated PZR level **“fails LOW”**
 - 3) Exactly the same as 3) - 7) above.

Aside: PZR Heater Bank 2 -Group B & C when powered from the SSF, are unaffected by the two ICCM failures described above. PZR Level #2 “uncompensated” signal reads out at the SSF and does not feed ICCM. At **85” decreasing**, Groups B & C will cutoff when being controlled from the SSF.

- c) **If an ICCM train is taken off-line for test, maintenance, or calibration then the following will occur:**
- 1) The plasma display will indicate **DATA LINK FAILURE**.
 - 2) All other indications (gauges, recorders, computer points) will **freeze as is** unless that particular indication is being calibrated. An indication that is being calibrated will **NOT** have a constant value.

Technical Reference(s): AP/1/A/1700/014 Rev. 018
(Attach if not previously provided) PNS-PZR Lesson Plan Rev. 21
(including version/revision number) IC-RCI Lesson Plan Rev. 22c

Proposed references to be provided to applicants during examination: None

Learning Objective: PNS-PZR Lesson Obj. R31 &R35
IC-ICI Lesson Plan Obj. 13,15,16,17,18,20,22-31

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

Question History: Last NRC Exam _____
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41
55.43 _____

Comments:
000022 Loss of Rx Coolant Makeup / 2

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022AK1.04 Knowledge of the operational implications of the following concepts as they apply to Loss of Reactor Coolant Makeup: Reason for changing from manual to automatic control of charging flow valve controller

Author: Dan Bacon

Oconee Nuclear Station 2013-301 Written Examination

Question # 7

Given the following Unit 1 current conditions:

- Unit shutdown is in progress.
- LPI is aligned in the switchover mode.
- RCS Pressure = 150 psig.

Based on the current conditions, which ONE of the following completes the statements listed below?

If (1) fails closed a loss of decay heat removal will occur.

And

In accordance with AP/26 (Loss of Decay Heat Removal), heat removal will be restored by (2) to control the cooldown.

- A. (1) 1LPSW-4
(2) aligning the LPI system in high pressure mode
- B. (1) 1LPSW-4
(2) aligning the LPI system to the other cooler in normal mode
- C. (1) 1LPSW-5
(2) aligning the LPI system in high pressure mode
- D. (1) 1LPSW-5
(2) aligning the LPI system to the other cooler in normal mode

Proposed Answer: C

Explanation (Optional):

1. LPSW-5 isolates all LPSW flow to the B LPI Cooler. LPSW-252 is the control valve downstream of LPSW-5.
2. LPSW-4 isolates all LPSW flow to the A LPI Cooler. LPSW-251 is the control valve downstream of LPSW-4.
3. LPI cooler = RHR heat exchanger = DHR heat exchanger
4. The B LPI cooler is the only one with RCS flow through it in the switchover mode. It is placed on the suction side of the LPI pump.
5. The A LPI cooler is the only one with RCS flow through it in the high pressure mode.

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6. Per AP-26 if there is not adequate LPSW flow to the B LPI cooler, then A LPI cooler will be placed in service. If <125 psig, then the normal mode is used. If ≥125 psig, then high pressure mode is used.
7. This is close to question #6 from ILT 42 exam. Significantly modified. See attached question.

Answer Analysis:

- A. Incorrect. First part is plausible because it would be correct in high pressure mode. Second part is correct.
- B. Incorrect. First part is plausible because it would be correct in high pressure mode. Second part is plausible because it would be correct if RCS pressure was <125 psig.
- C. Correct.
- D. Incorrect. First part is correct. Second part is plausible because it would be correct if RCS pressure was <125 psig.

Technical Reference(s): AP/1/A/1700/026 Rev. 22
(Attach if not previously provided) EAP-APG Lesson Plan Rev. 7d
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: EAP-APG Lesson Plan Obj. R8 & R9

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

Question History: Last NRC Exam 2012-302
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 X
55.43 _____

Comments:
000025 Loss of RHR System / 4

025AK2.01 Knowledge of the interrelations between the Loss of Residual Heat Removal System and the following: RHR heat exchangers

FOR REVIEW ONLY - DO NOT DISTRIBUTE
ILT42 ONS SRO NRC Examination QUESTION 6 A 6

Given the following Unit 1 conditions:
Unit shutdown in progress
LPI aligned in High Pressure Mode

- 1) If ___ (1) ___ fails to control properly a loss of decay removal can occur.
- 2) In accordance with AP/26 (Loss of Decay Heat Removal), heat removal will be restored by ___ (2) ___ to control the cooldown.

Which ONE of the following completes the statements above?

- A. 1. 1LPSW-251
2. placing 1LPSW-251's FAIL SWITCH in FAIL OPEN and throttling 1LPSW-4
- B. 1. 1LPSW-251
2. aligning the LPI system in series mode and using 1LPSW-252
- C. 1. 1LPSW-252
2. placing 1LPSW-252's FAIL SWITCH in FAIL OPEN and throttling 1LPSW-5
- D. 1. 1LPSW-252
2. aligning the LPI system in switchover mode and using 1LPSW-251

APE025 AK2.01 - Loss of Residual Heat Removal System (RHRS)
Knowledge of the interrelations between the Loss of Residual Heat Removal System and the following: (CFR 41.7 / 45.7)
RHR heat exchangers

ILT42 ONS SRO NRC Examination QUESTION 6 A 6

General Discussion

Answer A Discussion

Correct. In the High Pressure Mode flow is through the 1A LPI cooler only. 1LPSW-251 failing closed will result in having to enter AP/26 (Loss of Decay Heat Removal). AP/26 will direct placing 1LPSW-251's FAIL SWITCH in FAIL OPEN and using 1LPSW-4 to control heat removal.

Answer B Discussion

Incorrect. First part is correct. Second part is plausible because it is correct for switchover mode.

Answer C Discussion

Incorrect. First part is plausible because the candidate could have the misconception that in the High Pressure Mode LPI is aligned through the "B" cooler. Second part is plausible because it would be correct if the first part of the question were correct.

Answer D Discussion

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Incorrect. First part is plausible because the candidate could have the misconception that in the High Pressure Mode LPI is aligned through the "B" cooler. Second part is plausible because it would restore heat transfer.

Cognitive Level

Comprehension

Job Level

RO

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Question # 8

Given the following Unit 1 conditions:

- Core Thermal Power = 100%.
- The Primary Instrument Air Compressor is isolated for maintenance.

Current Conditions:

- 1SA-9 A-2 (RCW Header Pressure Low) Statalarm is received.

Based on the current conditions, which ONE of the following completes the statements listed below?

In accordance with OP/0/A/1106/27 (Instrument Air System), (1) Pressure Service Water may be lined up to provide an alternate source of cooling water to the Backup (Worthington) Instrument Air Compressors.

And

1SA-9 A-2 may indicate a reduction in cooling provided to the RCP (2) Coolers.

- A. 1. Low
2. Seal
- B. 1. Low
2. Seal Return
- C. 1. High
2. Seal
- D. 1. High
2. Seal Return

Proposed Answer: B

Explanation (Optional):

1. Components cooled by CCW (Component Cooling Water) on Westinghouse plants are cooled by CC (Component Cooling), RCW (Recirculating Cooling Water) or LPSW (Low Pressure Service Water) at Oconee. For example Spent Fuel coolers and PALS sample coolers are cooled by CCW on a Westinghouse plant and are cooled by RCW at Oconee.
2. There are no components cooled by CC at Oconee that can receive backup cooling from a Service Water System.

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Question # 9

Unit 1 Initial Conditions:

- Core Thermal Power = 60%.
- 1A1 RCP was previously secured due to high vibrations.

Current Conditions:

- LDST level = 39”.
- The Reactor is tripped.
- EP/1/A/1800/001 (EOP SGTR Tab) has been initiated due to a SGTR on 1B S/G.
- RCS depressurization methods are inadequate in minimizing core SCM.

Based on the current conditions, which one of the following completes the statements listed below?

In accordance with the EOP Steam Generator Tube Rupture Tab, (1) temperature should be used to determine pressurizer spray nozzle ΔT .

And

In accordance with the EOP Steam Generator Tube Rupture Tab, Auxiliary Pressurizer Spray may be initiated when pressurizer spray nozzle ΔT is less than (2) degrees F.

- A. (1) LDST
(2) 410
- B. (1) LDST
(2) 470
- C. (1) BWST
(2) 410
- D. (1) BWST
(2) 470

Proposed Answer: C

Explanation (Optional):

1. Per EOP Enclosure 5.20 (Aux Pzr Spray), LDST temperature should be used for calculating ΔT if LDST is supplying HPI pump suction. BWST temperature should be used for calculating ΔT if BWST is supplying HPI pump suction. Computer point O1P3367 provides Pzr spray nozzle ΔT information based on which source is supplying HPI pump suction.

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2. Per the note prior to step 38 in the SGTR Tab, BWST temperature should be used in determining Pzr spray nozzle ΔT . Computer point O1P3367 provides Pzr spray nozzle ΔT information. This is because 1HP-24 and 1HP-25 were opened in step 29.

3. Step 38 RNO of the SGTR Tab initiates EOP Enclosure 5.20 (Aux Pzr Spray) if Pzr spray nozzle ΔT is less than 410 degrees F.

4. 470 degrees F is plausible because step 4 of the FCD tab states, **IAAT** T_{cold} approaches 470°F, **AND** all RCPs are operating, **THEN** ensure < four RCPs are operating.

5. Per OP/1/A/1103/005 (Pzr Operation) Limits and Precautions, Maintain ΔT between spray water and Pzr < 410°F.

Technical Reference(s): EP/1/A/1800/001 (SGTR) Rev. 039
(Attach if not previously provided) EAP-SGTR Lesson Plan Rev. 21a
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: EAP-SGTR Lesson Plan Obj. R17 & R22

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

Question History: Last NRC Exam _____
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 X
55.43 _____

Comments:
000027 Pressurizer Pressure Control System Malfunction / 3

027AK3.03 Knowledge of the reasons for the following responses as they apply to the Pressurizer Pressure Control Malfunctions: Actions contained in EOP for PZR PCS malfunction

Author: Dan Bacon

Oconee Nuclear Station 2013-301 Written Examination

Question # 10

Given the following Unit 3 conditions:

- Time = 0600.
- Rule 1 (ATWS / Unanticipated Nuclear Power Production) has been initiated.

Current conditions:

- Time = 0603.
- RCS pressure is 2455 psig and increasing.

Which ONE of the following completes the statements listed below?

- 1) In accordance with Rule 1, an operator will be dispatched to open the Unit 3 CRD 600V normal power supply breaker at 3X9 and alternate 600V power supply breaker at (1).
- 2) Based on the current conditions, DSS (2) open all four CRD breakers to de-energize Control Rod Groups 1-7.

Which ONE of the following completes the statements above?

- A. 1. 1X1
2. will
- B. 1. 2X2
2. will
- C. 1. 1X1
2. will NOT
- D. 1. 2X2
2. will NOT

Proposed Answer: D

Explanation (Optional):

1. EP/3/A/1800/001 ATWS Rule 1 Step 8 states "Dispatch one operator without wearing Arc Flash PPE to open 600V CRD breakers:" Rule 1 for Units 1 and 2 makes the same statement.
2. Unit 3 normal 600V CRD power supply is from 3X9.
3. Unit 3 alternate 600V CRD power supply is from 2X2.
4. Unit 1 and 2 normal 600V CRD power supply is from 1X9 and 2X9 respectively.

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5. Unit 1 and 2 alternate 600V power supply is from 2X1 and 1X1 respectively.
6. When RCS pressure reaches 2450 psig, a DSS signal will be generated. DSS does not trip the CRD breakers. DSS causes the safety and regulating rods to drop by interrupting the gating current to the motors.
7. AP/8 step 18 RNO for loss of control room dispatches an operator to the cable room to trip all 4 CRD breakers in addition to opening the 600V CRD supply breakers. This information is provided for other possible questions.
8. For vital DC power, Unit 3 backs up Unit 2, Unit 2 backs up Unit 1, and Unit 1 backs up Unit 3. This information is provided for plausibility of alternate power supplies.

Technical Reference(s): IC-CRI Lesson Plan Rev. 13c
(Attach if not previously provided) EAP-UNPP Att. 1 Rule 1 Lesson Plan Rev. 04a
(including version/revision number) EP/3/A/1800/001 Rules and Appendix Rev. 038

Proposed references to be provided to applicants during examination: None

Learning Objective: IC-CRI Lesson Plan Obj. R25
EAP-UNPP Att. 1 Rule 1 Lesson Plan Obj. R3

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

Question History: Last NRC Exam _____
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 X
55.43 _____

Comments:

EK2 Knowledge of the interrelations between the following during an ATWS:

(CFR 41.7 / 45.7)

EK2.06 Breakers, relays, and disconnects 2.9* 3.1*

Author: Dan Bacon

Oconee Nuclear Station 2013-301 Written Examination

Question 11

Given the following Unit 1 conditions:

Time = 0600

- Reactor power = 35%
- TDEFDW pump is OOS

Time = 0601

- Both Main FDW pumps trip

Time = 0610

- 1A SG level = 30 inches XSUR stable
- 1B SG level = 36 inches XSUR increasing

At 0600, 1RIA-59 & 1RIA-60 (1) be used to determine the SG tube leak rate.

At 0610, SG level response is due to (2).

Which ONE of the following completes the statements above?

- A 1. can
 2. a loss of Instrument Air to 1FDW-316
- B 1. can
 2. a primary to secondary rupture in the 1B SG
- C 1. cannot
 2. a loss of Instrument Air to 1FDW-316
- D 1. cannot
 2. a primary to secondary rupture in the 1B SG

Proposed Answer: D

K/A Match Analysis

Requires the ability to interpret control board indication and plant response to determine if a SG is ruptured.

Answer Choice Analysis

- A. INCORRECT. First part is plausible because 1RIA-59 & 60 (MS Line N-16 gamma detectors) are accurate above 40% power. Below 40% they provide a trend but by procedure cannot be used to determine leak rate. Second part is plausible because a

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loss of all gas to the valve positioner would cause the valve to fail open, however 1FDW-316 is backed upped with Nitrogen.

- B. INCORRECT. First part is plausible because 1RIA-59 & 60 (MS Line N-16 gamma detectors) are accurate above 40% power. Below 40% they provide a trend but by procedure cannot be used to determine leak rate. Second part is correct.
- C. INCORRECT. First part is correct. Second part is plausible because a loss of all gas to the valve positioner would cause the valve to fail open, however 1FDW-316 is backed upped with Nitrogen.
- D. CORRECT. First part correct. Second part correct, the SG levels should be controlling at 30 inches XSUR. 1B SG level is increasing this is cause by the tube leakage.

Technical Reference(s):

Proposed references to be provided to applicants during examination: None

Learning Objective:

EAP-SGTR, Obj. R2, Given a set of conditions, be able to identify and quantify OTSG tube leakage.

RAD-RIA, Obj. R2, Describe the basic function of each applicable monitor.

Question Source: Bank # _____
Modified Bank # EAP090404 and 2009 exam
(Combined part of each question and changed initial power level to change correct answer)
New _____

Question History: Last NRC Exam: ILT42 Q10 (First half of question is from ILT 42 Q10, second half is from 2009 Q10)

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

10 CFR Part 55 Content: 55.41 X
55.43 _

Comments:

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038EA2.03 Ability to determine or interpret the following as they apply to a SGTR: Which S/G is ruptured.

Supporting References (excerpted from lesson plans)

3. In general, Operations procedures use RIA-59/60 (Main Steam Line N-16 detectors) to provide diverse/redundant indication of a SG tube leak/rupture and (where allowed) as an alternate for RIA-40.

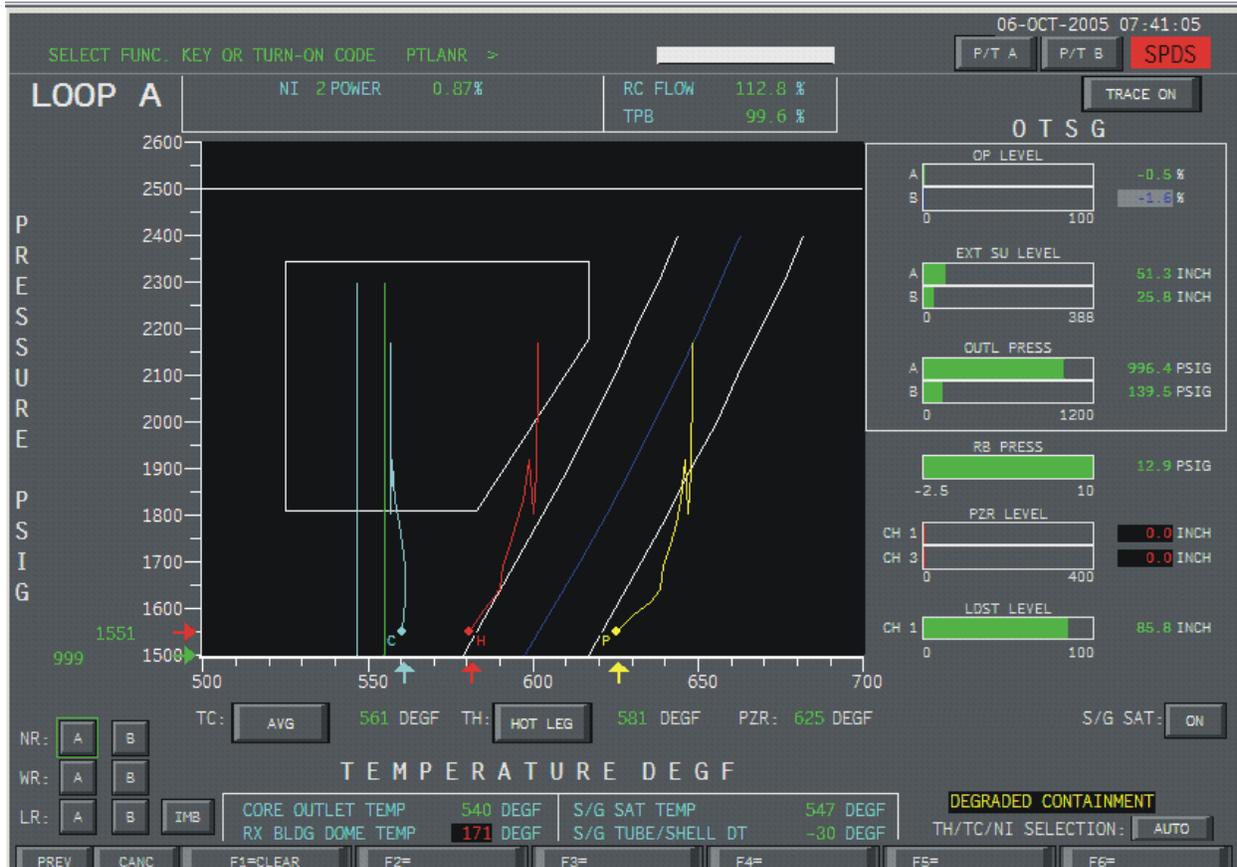
NOTE: RIA-59 or 60 are NOT used by Operation's procedures below 40% power.

References Provided to Applicant

- None

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Question 12



Unit 1 plant conditions:

- Main Steam Line Break occurred 20 seconds ago
- Plant response as indicated above

Based on the above conditions, which ONE of the following describes:

Digital ES channels (1) should have automatically actuated.

and

Diverse HPI (2) have automatically actuated.

- A 1) Channels 1 through 8
2) should
- B 1) Channels 1 and 2 ONLY
2) should
- C 1) Channels 1 through 8
2) should NOT
- D 1) Channels 1 and 2 ONLY
2) should NOT

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Proposed Answer: C

K/A Match Analysis

Question requires knowledge of when various ES channels require initiation during a MSLB.

Answer Choice Analysis

- E. INCORRECT. First part correct. Second part plausible, the setpoint for Diverse HPI is <1550 psig.
- F. INCORRECT. First part Plausible, would be correct if only RCS pressure were considered and containment pressure were disregarded. Second part plausible, the setpoint for Diverse HPI is <1550 psig.
- G. CORRECT. First part Plausible, would be correct if only RCS pressure were considered and containment pressure were disregarded. Second part is correct.
- H. INCORRECT. Plausible, would be correct if only RCS pressure were considered and containment pressure were disregarded.

Technical Reference(s):

EOP Enclosure 5.1, ES Actuation, Rev 39

(Attach if not previously provided)

(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective:

OP-OC-IC-ES, Rev 18a

Obj. R26 - Given a set of conditions determine Instrument Channel response.

Obj. R30 - Describe the following for the Diverse Pressure Injection System (DPIAS):
DHPIAS/DLPIAS Setpoints for Actuation and Reset

<u>Question Source:</u>	Bank #	_____
	Modified Bank #	_____
	New	<u> X </u>

Question History: Last NRC Exam: NEW

<u>Question Cognitive Level:</u>	Memory or Fundamental Knowledge	_
	Comprehension or Analysis	<u> X </u>

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10 CFR Part 55 Content: 55.41 X
 55.43 _

Comments:

040AA2.04 Ability to determine and interpret the following as they apply to the Steam Line Rupture: Conditions requiring ESFAS initiation.

Supporting References (excerpted from lesson plans)

Lesson Plan OP-OC-IC-ES, Rev 18a

	<u>TS Setpoint</u>	<u>Actual Setpoint</u>
High Pressure Injection, KeoweeEmerg. Start and RB Non-Essential Isolation Channels 1&2	RCS ≥ 1590 psig OR RB ≤ 4 psig	RCS 1600 psig OR RB 3.0 psig
Low Pressure Injection and LPSW Channels 3 &4	RCS ≥ 500 psig OR RB ≤ 4 psig	RCS 550 psig OR RB 3.0 psig
Reactor Building Cooling, Penetration Room Ventilation and RB Essential Isolation Channels 5 &6	RB ≤ 4 psig	RB 3.0 psig
Reactor Building Spray Channels 7 &8	RB ≤ 15 psig	RB 10 psig

FUNCTION	Bistable SETPOINT	RESET
<u>DHPIAS</u>	<u>1550 psig</u>	<u>1600 psig</u>
<u>DLPIAS</u>	<u>462 psig</u>	<u>512 psig</u>

References Provided to Applicant - None

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Question 13

Given the following Unit 1 conditions:

- Reactor power = 100%

Which ONE of the following will result in an AUTOMATIC trip of the Main Turbine?

- A EHC header pressure 1150 psig
- B High Moisture Separator Reheater level on one level switch on two of four MSRHS.
- C 780 psig discharge pressure on BOTH Main Feedwater pumps
- D 72 psig hydraulic oil pressure on BOTH Main Feedwater pumps

Proposed Answer: D

K/A Match Analysis

Requires the ability to determine the occurrence of a turbine trip as it applies to the Loss of Main Feedwater.

Answer Choice Analysis

- A. INCORRECT. Plausible since there is a low EHC header pressure trip of the main turbine at 1100 psig.
- B. INCORRECT. Plausible since a high level on 2 out of 3 level switches on any one of the four MSRHS is a main turbine trip.
- C. INCORRECT. Plausible since AMSAC will trip the Main Turbine and start all operable EFWPs, need both channels of AMSAC/DSS to be enabled (2/2 logic) AND: either Both MFPs have low hydraulic oil pressure (<75 psig) Or Both MFPs have low discharge pressure (<770 psig).
- D. CORRECT. Since there is an AMSAC trip of the Main Turbine at < 75 psig hydraulic oil pressure of both MFPs.

Technical Reference(s):

(Attach if not previously provided)

(including version/revision number)

Proposed references to be provided to applicants during examination: None

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Learning Objective:

STG-EHC Rev 16a, Obj. R10 and 23

Question Source:

Bank # _____

Modified Bank # 2011B Q25

(Entirely changed distractors A and B. Changed values in C and D to make C incorrect and change correct answer to D.)

New _____

Question History:

Last NRC Exam: modified from 2011B Q25

Question Cognitive Level:

Memory or Fundamental Knowledge X

Comprehension or Analysis _____

10 CFR Part 55 Content:

55.41 X

55.43 _____

Comments:

054AA2.01 Ability to determine and interpret the following as they apply to the Loss of Main Feedwater: Occurrence of reactor and/or turbine trip

Supporting References (excerpted from lesson plans)

STG-EHC Rev 16a, Obj.R10 and 23

Identify the automatic turbine trips including setpoint and type of protection provided.

5.

AMSAC (ATWS Mitigation Safety Actuation Circuit

Provided to mitigate the consequences of an anticipated transient without SCRAM.

AMSAC will trip the Main Turbine and start all operable EFWPs.

Setpoint: need both channels of AMSAC to be enabled (2/2 logic) **AND:**
either

Both MFPs have low hydraulic oil pressure (<75 psig)

OR

Both MFPs have low discharge pressure (<770 psig)

(Obj. R23, R10) Summary of Turbine Trips

Mechanical Overspeed - 1980 RPM / 110% of rated speed.

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Backup Overspeed - \square 2003 RPM / 111.25% of rated speed.

Loss of Speed signals to the turbine speed control circuitry.

Low Condenser Vacuum - \square 21.75 inches Hg.

Loss of Stator Coolant - If runback does not reduce phase amps below \approx 22,000 within 2 minutes and below \approx 7000 within an additional 1½ minutes if condition has not cleared when first plateau is reached.

Low Bearing Oil Pressure - incorporates 3 pressure switches in a 2 out of 3 trip logic at \leq 8 psig.

High Steam Generator Level Trips

96% Level on Operating Range on either SG

SG Overfill Protection - There is an additional auxiliary relay for the OTSG level control system circuitry. The existing high level contacts feed this relay. When both level signal monitors for either A or B SGs sense a high level, a redundant trip signal is sent to both MFDWPS and the Main Turbine.

High Moisture Separator Level Trip – level at bottom of MSRH

This trip incorporates 3 level switches on each MSRH to provide 2 out of 3 Turbine Trip logic.

Reactor Trip

AMSAC actuation

Low EHC Discharge Header Pressure - 1100 psig decreasing

Emergency Low Hydraulic Oil Pressure - <800 psig.

Generator Lockout - 86 GA

Loss of 24V DC

Manually initiated from:

Turbine Oil Fire Trip

Manual Trip Handle on Front Standard

Master Trip Button in Control Room

Low Frequency

Generator is On-line and selected frequency drops below 59.5 Hz for 48 minutes cumulative.

Generator is On-line and selected frequency drops below 58.6 Hz for 8 minutes cumulative.

Generator is On-line and selected frequency drops below 58.1 Hz for 48 seconds cumulative.

Generator is On-line and selected frequency drops below 57.6 Hz instantaneously.

References Provided to Applicant

- None

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Question 14

Given the following Unit 1 conditions:

Initial conditions:

- Reactor power = 100%

Current conditions:

- Station Blackout (power has NOT been restored)
- RCS Temperatures 2 minutes after trip
 - T_c = 548°F
 - T_h = 556°F
 - CETCs = 558°F
- SG Pressures = 1010 psig stable

Which ONE of the following describes the response of RCS temperature indications over the next five minutes?

	<u>RCS Tcold</u>	<u>CETC</u>
A	Approximately Stable	Approximately Stable
B	Approximately Stable	Increasing
C	Decreasing	Approximately Stable
D	Decreasing	Increasing

Proposed Answer: B

K/A Match Analysis

Requires the knowledge of the operational implications natural circulation cooling as it applies to the relationship between RCS Loop Temperatures and CETC temperature to determine Natural Circulation cooling during a station blackout.

Answer Choice Analysis

- A. INCORRECT. Plausible in that T_c would be stable and CETC response could be correct if at low decay heat levels however this is a Rx trip from 100% therefore Decay heat would be high. Additionally, CETC's would be stable once Natural Circ had been established and stable.
- B. CORRECT. In the time period following the trip natural circulation conditions will be developing. T_{hot} and CETC will be increasing with T_{cold} being held constant by SG pressures in order to build in an adequate thermal driving head to establish flow. It takes 10 - 15 minutes to fully establish Natural Circ flow. Since T_c is already at T_{sat} for SG pressure, T_c will remain stable while CETC's increase. After flow is established CETC & T_{hot} will stabilize and eventually decrease as either decay heat level drops off or SG pressures are reduced

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- C. INCORRECT. Tcold is plausible based on assuming CETC's remain constant and Tc decreases to establish the required delta T to ensure Natural Circulation flow occurs. CETC response could be correct if at low decay heat levels however this is a Rx trip from 100% therefore Decay heat would be high. Additionally, CETC's would be stable once Natural Circ had been established and stable.

- D. INCORRECT. Prior to stable Natural Circulation being established it is plausible to believe that Tc decreases since there is no flow in the loops and CETC's increase as part of the process of establishing the Delta T required for Natural Circ flow.

Technical Reference(s):

Proposed references to be provided to applicants during examination: None

Learning Objective: Obj. TA-AM1 R3

<u>Question Source:</u>	Bank #	<u>2011B Q 11</u>
	Modified Bank #	_____ (Note changes or attach parent)
	New	_____

Question History: Last NRC Exam: 2011B

<u>Question Cognitive Level:</u>	Memory or Fundamental Knowledge	—
	Comprehension or Analysis	<u>X</u>

<u>10 CFR Part 55 Content:</u>	55.41 <u>X</u>
	55.43 _

Comments:

055EK1.02 Knowledge of the operational implications of the following concepts as they apply to the Station Blackout: Natural circulation cooling.

Supporting References (excerpted from lesson plans)

TA-AM1

References Provided to Applicant

- None

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Question 15

Automatic control circuits will close the associated feeder breakers of 1X7, 2X4 & 3X4 after a load shed has occurred and a _____ second timer has timed out.

The reason for the time delay to re-energize 1X7, 2X4 & 3X4 after a load shed is to _____.

- A
 - 1. 30
 - 2. insure the integrity of the RCP seals
- B
 - 1. 30
 - 2. prevent overloading the standby transformers
- C
 - 1. 60
 - 2. insure the integrity of the RCP seals
- D
 - 1. 60
 - 2. prevent overloading the standby transformers

Proposed Answer: D

K/A Match Analysis

056AK3.01 Knowledge of the reasons for the following responses as they apply to the Loss of Offsite Power: Order and time to initiation of power for the load sequencer.

Requires knowledge of the reasons for the order and time to initiation of portions of the power switching logic as it applies to a Loss of Offsite Power.

Answer Choice Analysis

- I. INCORRECT. First part is plausible because X5 & X6 load centers are load shed under certain conditions and the auto reclose time delay for these is 30 seconds. Second part is plausible because this is one of the purposes of the Main Feeder Bus Monitor Panels. MFBMP's assure the integrity of the RCP seals by insuring that seal injection and component cooling flows are regained following a loss of power.
- J. INCORRECT. First part plausible, see A. Second part is correct, the purpose of the load shed system is the shedding non-essential loads reducing the load on the Main Feeder Bus to within the capacity of the standby transformers (CT4 and CT5).
- K. INCORRECT. First part is correct; the time delay for 1X7, 2X4 & 3X4 after a load shed is 60 seconds to reenergize. Second part is plausible because this is one of the purposes of the Main Feeder Bus Monitor Panels. MFBMP's assure the integrity of the RCP seals by insuring that seal injection and component cooling flows are regained following a loss of power.
- L. CORRECT. The time delay for 1X7, 2X4 & 3X4 after a load shed is 60 seconds to

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reenergize. The purpose of the load shed system is the shedding non-essential loads reducing the load on the Main Feeder Bus to within the capacity of the standby transformers (CT4 and CT5).

Technical Reference(s):

Proposed references to be provided to applicants during examination: None

Learning Objective:

OP-OC-EL-PSL Rev. 13e Objectives R3, R6 and R11

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

Question History: Last NRC Exam: NEW

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

10 CFR Part 55 Content: 55.41 X
55.43 _

Supporting References (excerpted from lesson plans)

OP-OC-EL-PSL Rev. 13e, page 12 of 54

2.3 Load Shed (**PSL-03**)

E. (**Obj. R3**) Purpose

1. Trip the feeder breakers to non-essential loads on the 4KV switchgear. Shedding the non-essential loads reduces the load on the Main Feeder Bus to a level within the capacity of the standby transformers (CT4 and CT5) in the event a standby source is required.

OP-OC-EL-PSL Rev. 13e, page 18 of 54

2.4 Main Feeder Bus Monitor Panel (MFBMP) (**PSL-07**)

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- F. (Obj. R11) Purpose
1. The MFBMP's are designed to assure a reliable source of power to the MFB's during non-LOCA loss of power events.
 2. In addition, the MFBMP's assure the integrity of the RCP seals by insuring that seal injection and component cooling flows are regained.

OP-OC-EL-PSL Rev. 13e, page 14 and 15 of 54

- E. (Obj. R6) Upon actuation, the load shed circuits will:
1. Open the feeder breakers to non-essential loads on the 4160V switchgear. These loads are:
 - l) **X5 and X6** feeder breakers will only open automatically (X5 & X6 will be load shed) if ALL of the following conditions are met:
 - (Either channel) PSL "Standby Close Initiate" logic signal initiated
 - No ES signal present
 - Either SL1 or SL2 breaker closed into CT-5
 1. **X5 & X6** control circuitry provides for an **automatic reclosure** of their associated feeder breaker after a **SBCI** (on either PSL channel) has opened their breakers.
 - (a) The auto reclosure circuit will only work if **both PSL channels of SBCI** on the affected unit activate.
 - (b) The **auto reclosure circuit starts a 30 second timer** upon receipt of PSL channel A and B SBCI signals (either SBCI signal WILL trip open the feeder breaker), when the timer times out, the auto reclosure circuit will then close back in the feeder breaker to X5 & X6.
 - (c) The reclosure circuit will reclose the feeder breaker even if power has not been restored to the affected unit.
 - m) **1X7, 2X4 & 3X4** control circuitry provide for an **automatic reclosure** of their associated feeder breaker after a **load shed** has opened their breaker.
 - 1) The auto reclosure circuit will only work if **both channels of Load Shed** on the affected unit activate.
 - 2) The **auto reclosure circuit starts a 60 second timer** upon receipt of channel A and B Load Shed signals (either Load Shed signal WILL trip open the feeder breaker), when the timer times out, the auto reclosure circuit will then close back in the feeder breaker to 1X7, 2X4, OR 3X4.

References Provided to Applicant

- None

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Question 16

Unit 2 plant conditions:

INITIAL CONDITIONS:

- Reactor power = 100%
- Pressurizer (PZR) Level 1 selected
- 2HP-120 (RC Volume Control) in AUTOMATIC
- SASS in MANUAL

CURRENT CONDITIONS:

- ICCM Train "A" experiences a total loss of power

2HP-120 will _____ and will _____ to control PZR level.

Which ONE of the following completes the statement above?

- A 1) automatically swap to manual
 2) need to be throttled closed
- B 1) remain in automatic
 2) need to be throttled closed
- C 1) automatically swap to manual
 2) need to be throttled open
- D 1) remain in automatic
 2) need to be throttled open

Proposed Answer: B

K/A Match Analysis

Requires the ability to monitor and operate the manual control of PZR level following the Loss of Vital AC Instrument Bus power.

Answer Choice Analysis

- A. INCORRECT – First part plausible, controller will swap to manual on loss of automatic power (from essential power system), but not on loss of ICCM train (vital power system). Second part is correct, PZR level fails low and 3HP-120 will remain in automatic therefore 2HP-120 will fully open and need to be throttled closed to control PZR level.
- B. CORRECT—Controller will remain in automatic, would only automatically swap to manual upon loss of essential power. For total loss of power failure to ICCM train w/ SASS in manual, PZR level fails low and 2HP-120 will remain in automatic therefore 2HP-120 will fully open and need to be throttled closed to control PZR level.
- C. INCORRECT -First part plausible, controller will swap to manual on loss of automatic power (from essential power system), but not on loss of ICCM train (vital power system).

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Second part plausible if applicant mistakenly believes indicated PZR level will fail high or that 2HP-120 will go closed or fail closed.

- D. INCORRECT –First part correct. Second part plausible if applicant mistakenly believes indicated PZR level will fail high or that 2HP-120 will go closed or fail closed.

Technical Reference(s):

Lesson Plans PNS-PZR Rev. 21 and EL-VPC Rev. 16b

Proposed references to be provided to applicants during examination: None

Learning Objective:

PNS-PZR, Obj. 35: Given a set of conditions, determine how PZR level control/indication is affected by a loss of SASS and/or ICCM.

EL-VPC, Obj. 7: List the major loads supplied by the KI, KU, and KX inverters.

Question Source: Bank # PNS143503

<u>Question Source:</u>	Bank #	_____
	Modified Bank #	<u>143503</u> (parent attached)
	New	_____

Question History: Last NRC Exam: 2004

<u>Question Cognitive Level:</u>	Memory or Fundamental Knowledge	—
	Comprehension or Analysis	<u>X</u>

<u>10 CFR Part 55 Content:</u>	55.41	<u>X</u>
	55.43	—

Comments:

057AA1.02 Ability to operate and / or monitor the following as they apply to the Loss of Vital AC Instrument Bus: Manual control of PZR level.

Supporting References (excerpted from lesson plans)

References Provided to Applicant

- None

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Question 17

Given the following plant conditions:

Initial conditions:

- ALL three Units Reactor power = 100%
- 3CB battery is inoperable due to low cell voltages

Current conditions:

- 2CA battery is inoperable due to the failure of the 2CA Battery Charger

Unit(s) _____ DO/DOES NOT meet(s) the minimum requirements of LCO 3.8.3 (DC Sources –Operating).

Which ONE of the following completes the statement above?

- A 2 ONLY
- B 2 and 3 ONLY
- C 3 ONLY
- D 1, 2, and 3

Proposed Answer: D

K/A Match Analysis

Requires knowledge of the operational implications of the loss of one control battery charger on all three units DC systems.

Answer Choice Analysis

- A. INCORRECT. Plausible because the candidate could have the misconception that since Unit 2 is backed up by unit 3 that only Unit 2 would be affected.
- B. INCORRECT. Plausible because the applicant could have the misconception that Unit 2 and Unit 3 are the only ones that would be affected.
- C. INCORRECT. Plausible because the candidate could have the misconception that Unit 3 is backed up by Unit 2 therefore only Unit 3 is affected.
- D. CORRECT. With ALL three units at 100% power 5 of 6 batteries are required. ALL units **do not** meet the requirements of LCO 3.8.3.

Technical Reference(s): TS 3.8.3

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Proposed references to be provided to applicants during examination: None

Learning Objective:

OP-OC-EL-DCD, Rev 13b, Obj. 15, 16 & 17 designated for RO and SRO.

Question Source: Bank # _____
Modified Bank # ILT-41 Q 13 (parent attached)
New _____

Question History: Last NRC Exam: ILT-41

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

10 CFR Part 55 Content: 55.41 X
55.43 _

Comments:

058AK1.01 Knowledge of the operational implications of the following concepts as they apply to Loss of DC Power: Battery charger equipment and instrumentation.
2.8/3.1

Supporting References (excerpted from lesson plans)

OP-OC-EL-DCD, Rev 13b, Obj 15,16 & 17 designated for RO and SRO.

H. **(Obj. R15, 16, 17)** Technical Specifications

1. Applicable Specs
 - a) Tech Spec 3.8.3, DC Sources - Operating
 - b) Tech Spec 3.8.4, DC Sources - Shutdown
 - c) Tech Spec 3.8.8, Distribution Systems - Operating
 - d) Tech Spec 3.8.9, Distribution Systems – Shutdown
2. TS 3.8.3 Major LCOs and Actions
 - a) TS 3.8.3.a lists the I&C power sources required to be operable (3 of 4) per unit; if the unit is in Mode 1, 2, 3, or 4.
 - 1) Each power source shall be aligned to at least one panelboard provided that a power source is not the only source for two or more of the Unit's panelboards
 - b) TS 3.8.3.b requires two additional power sources when any other unit is in Mode 1, 2, 3, or 4.

- 1) This will require 5 of the 6 sources (1CA, 1CB, 2CA, 2CB, 3CA, 3CB)

References Provided to Applicant

- None

ILT41 ONS SRO NRC Examination QUESTION 13 **D** 13

Given the following plant conditions:

Initial conditions:

- ALL three Units Reactor power = 100%

Current conditions:

- 1CA battery is inoperable due to the failure of the 1CA Battery Charger

Unit(s) ___ MEET(s) the minimum requirements of LCO 3.8.3 (DC Sources – Operating).

Which ONE of the following completes the statement above?

- A. 2 ONLY
- B. 3 ONLY
- C. 2 and 3 ONLY
- D. 1, 2, and 3

APE058 AK1.01 - Loss of DC Power

Knowledge of the operational implications of the following concepts as they apply to Loss of DC Power: (CFR 41.8 / 41.10 / 45.3) Battery charger equipment and instrumentation

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Question 18

Given the following Unit 2 conditions:

- 2SA-18/B-11 (TURBINE BSMT WATER LEVEL ALERT) is in alarm
- 2SA-18/A-11 (TURBINE BSMT WATER EMERGENCY HIGH LEVEL) is in alarm
- Severe flooding exists in the Turbine Building
- Reactor is tripped
- AP/1/A/1700/010 Turbine Building Flood is in progress
- EOP Turbine Building Flood Tab is in progress

In 30 minutes, decay heat will be removed by (1) .

When LPSW is lost, All RCPs are stopped (2) .

Which ONE of the following completes the statement above?

- A 1. steaming with Atmospheric Dump Valves
 2. to minimize heat input to the RCS
- B 1. steaming with Atmospheric Dump Valves
 2. due to loss of cooling water
- C 1. steaming with Turbine Bypass Valves
 2. to minimize heat input to the RCS
- D 1. steaming with Turbine Bypass Valves
 2. due to loss of cooling water

Proposed Answer: B

K/A Match Analysis

Requires knowledge of the reasons for actions in the EOP for a loss of LPSW (nuclear service water).

Answer Choice Analysis

- M. INCORRECT. First part correct. Second part plausible because early in the TBF tab of the EOP three RCPs are stopped to minimize heat input into the RCS.
- N. CORRECT. First part, TBVs are used until vacuum is lost, early actions are to stop CCW pumps which will cause vacuum to be lost over a short period of time. Second part, upon loss of LPSW, all cooling is lost to RCPs and the final RCP will be stopped.
- O. INCORRECT. First part plausible because before vacuum is lost, decay heat is removed with TBVs
- P. INCORRECT. First part incorrect, Second part correct.

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Technical Reference(s):
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective:

OP-OC-EAP-TBF:

Obj. R3: Explain the basis for cautions, notes and major steps in TBF tab.

Obj. R9: Without the use of reference, when AP/10 (Turbine Building Flood) is required to be utilized by the operator be able to demonstrate the following:

- 9.1 State the Entry Conditions and Immediate Manual Actions in the AP.
- 9.2 Explain the basis for limits, cautions, notes and major steps in the AP.
- 9.3 Based on plant data received, summarize proper operator actions and strategies required in the AP to mitigate the abnormal plant condition.
- 9.4 Provide proper directions to operators and supporting groups performing actions of the AP outside the control room.

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

Question History: Last NRC Exam: NEW

<u>Question Cognitive Level:</u>	Memory or Fundamental Knowledge	_
	Comprehension or Analysis	<u>X</u>

<u>10 CFR Part 55 Content:</u>	55.41	<u> X </u>
	55.43	_

Comments:

062AK3.03 Knowledge of the reasons for the following responses as they apply to the Loss of Nuclear Service Water: Guidance actions contained in EOP for Loss of nuclear service water

Supporting References (excerpted from lesson plans)

OP-OC-EAP-TBF, Rev. 10a, pages 8 & 9 of 22.

2.3 Reduce operating RCPs to one.
Reduces heat input into the RCS.

NOTE

TurbineBuilding flooding will result in loss of condenser vacuum, rendering TBVs unavailable.

2.4 Stabilize temperature determined in step 1 by performing the following:

- Dispatch two operators to perform Encl 5.24 (Operation of the ADVs) (PS).
- Adjust SG pressure as necessary using:

_TBVs while available

_ADV after TBVs are lost

ADV are used to control SG pressure when the TBVs are not available. When the EOP instructs the operator to manually control SG pressure, this means to use TBVs if available or ADVs if not. After condenser vacuum is lost ADVs are used to control SG pressure. An NLO can be dispatched to the ADVs without a procedure because the required enclosure is pre-staged at the ADVs.

Secondary inventory is lost while using ADVs. Also if a primary to secondary leak exists then ADVs will provide a direct leak path to the environment.

2.5 IAAT LPSW is lost,
THEN stop all RCPs.

RCPs are stopped due to loss of cooling.

References Provided to Applicant

- None

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Question 19

Unit 1 initial conditions:

- Reactor power = 68% stable
- 1B2 RCP secured
- Control Rod Group 7 position = 65% withdrawn

Current conditions:

- Control Rod Group 7 Rod 1 drops to 40% withdrawn

Based on the above conditions, which ONE of the following correctly completes the statement:

The CRD system (1) generate a runback fault and the MAXIMUM final power level (Core Thermal Power) directed by AP/1 (Unit Runback) is (2).

- A 1. will
 2. < 60%
- B 1. will
 2. < 45%
- C 1. will **NOT**
 2. < 60%
- D 1. will **NOT**
 2. < 45%

Proposed Answer: D

K/A Match Analysis

Requires the ability to operate and / or monitor the reactor and turbine power as it applies to an inoperable or stuck control rod.

Answer Choice Analysis

- A. Incorrect: First part is plausible because an asymmetric fault does exist. However the runback fault is not met because the affected rod does not have a 0% or in limit. Second part is plausible since it would be correct if all 4 RCP's were operating.
- B. Incorrect: First part is plausible because an asymmetric fault does exist. However the runback fault is not met because the affected rod does not have a 0% or in limit. Second part is correct.
- C. Incorrect: First part is correct. Second part is plausible since it would be correct if all 4 RCP's were operating.

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- D. Correct: If any individual rod in groups 1-7 has a 0% or In Limit indication along with an individual rod asymmetric rod fault a runback fault will be generated. These conditions are not met. With 3 RCP's operating, the maximum allowable thermal power is 75%. With a dropped or misaligned rod, power must be reduced to 60% of allowable thermal power. AP/1 directs decreasing power to less than or equal to 45%.

Technical Reference(s):

AP/1/A/1700/001

Proposed references to be provided to applicants during examination: None

Learning Objective: Obj. IC-CRI R33

<u>Question Source:</u>	Bank #	<u>2009B Q 19</u> (Some formatting changes)
	Modified Bank #	_____
	New	_____

Question History: Last NRC Exam: 2009B

<u>Question Cognitive Level:</u>	Memory or Fundamental Knowledge	_
	Comprehension or Analysis	<u>X</u>

<u>10 CFR Part 55 Content:</u>	55.41	<u>X</u>
	55.43	_

Comments:

005AA1.03 Ability to operate and / or monitor the following as they apply to the Inoperable / Stuck Control Rod: Reactor and turbine power

Supporting References (excerpted from lesson plans)

IC-CRI R33

References Provided to Applicant

- None

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Question 20

Given the following plant conditions on Unit 1:

Time = 22:00

- Reactor power = 100%
- Turbine trip

Time = 22:05

- Reactor power = 11% slowly decreasing

In accordance with Rule 6 (HPI), HPI may be throttled when (1) NIs are $\leq 1\%$.

Power level is used to determine if throttling HPI is appropriate because it ensures that reactor power is (2).

Which ONE of the following completes the statements above?

- A 1. Wide Range
 2. below the point of adding heat
- B 1. Wide Range
 2. within the capacity of the EFDW system
- C 1. Power Range
 2. below the point of adding heat
- D 1. Power Range
 2. within the capacity of the EFDW system

Proposed Answer: B

K/A Match Analysis

024AK3.02 Knowledge of the reasons for the following responses as they apply to the Emergency Boration: Actions contained in EOP for emergency boration

Requires knowledge of the reason for emergency boration actions contained in the EOP.

Answer Choice Analysis

- A. INCORRECT. First part is correct. Per Rule 6 (HPI) with HPI Forced Cooling NOT in progress ALL WR NIs must be $\leq 1\%$. Second part is plausible because of a misconception that power is reduced so that the no nuclear heat is being added.

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- B. CORRECT. First part, Per Rule 6 (HPI) with HPI Forced Cooling NOT in progress ALL WR NIs must be \leq to 1%. Second part, the basis for the power at which HPI can be throttled during emergency boration is to reduce reactor power to within the heatremoval capacity of the EFDW system.
- C. INCORRECT. First part is plausible because Power Range NIs are used in IMAs to determine if entry into Rule 1 is required. Second part is plausible because of a misconception that power is reduced so that the no nuclear heat is being added.
- D. INCORRECT. First part is plausible, Power Range NIs are used in IMAs to determine if entry into Rule 1 is required. Second part is correct.

Technical Reference(s):

EOP Rule 6, Rev. 39

Proposed references to be provided to applicants during examination: None

Learning Objective

OP-OC-EAP-UNPP, Rev 18, Obj. R12

C. Given plant conditions, determine if HPI throttling requirements are met per Rule 6 (Throttling HPI).

C.2 Understand the termination criteria for emergency boration per the UNPPtab and Rule 6.

Question Source: Bank # _____
 Modified Bank # EAP111205(parent question attached)
 New _____

Question History: Last NRC Exam: ILT 39

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

10 CFR Part 55 Content: 55.41 X
 55.43 _

Comments:

Supporting References (excerpted from lesson plans)

References Provided to Applicant

- None

ILT39 ONS SRO NRC Examination QUESTION 2

Given the following Unit 1 conditions:

Time = 1200:

Reactor power = 100%

BOTH Main FDW Pumps trip

Time = 1205:

Reactor power = 26% slowly decreasing

PORV has failed open

1) In accordance with Rule 6 (HPI), the MAXIMUM power level at which HPI can be throttled is ___ (1) ___.

2) The reason power level is used to determine if throttling HPI is appropriate is that it ensures ___ (2) ___.

Which ONE of the following completes the statements above?

- A. 1. 1%
 2. Boron addition continues until power is less than 1%

- B. 1. 5%
 2. Boron addition continues until power is less than 5%

- C. 1. 1%
 2. sufficient core cooling exists until power level is low enough that HPI Forced cooling can remove the heat

- D. 1. 5%
 2. sufficient core cooling exists until power level is low enough that HPI Forced cooling can remove the heat

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Question 21

Unit 1 plant conditions:

- Reactor power = 80%
- The ICS SG Master controller is in MANUAL
- A condenser air leak has caused vacuum to slowly decrease with a corresponding increase in feedwater temperature

As the feedwater temperature increases, ICS ____ (1) ____ feedwater flow and will ____ (2) ____.

- A 1) will adjust
 2) start building a signal to insert control rods
- B 1) will adjust
 2) maintain control rods at current position
- C 1) will not adjust
 2) start building a signal to insert control rods
- D 1) will not adjust
 2) maintain control rods at current position

Proposed Answer: B

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Explanation (Optional):

As vacuum decreases:

Condensate temperature increases. This temperature increase will carry through the system to final feedwater temperature. Since the feedwater temperature input of ICS is downstream of the MFW Master controller, it's input will still modify the feedwater signal. As final feedwater temperature increases, ICS will increase feedwater flow which will heat transfer and therefore CTP to remain approximately the same. ICS should not sense enough of an RCS temperature change to adjust control rods.

- A. 1st part is correct. 2nd part is Incorrect but plausible because as feedwater temperature increases you would expect Tave to increase requiring a control rod insertion.
- B. Correct. (see above).
- C. 1st part is incorrect but plausible because the MFW Master controller is in MANUAL . 2nd part is Incorrect but plausible because as feedwater temperature increases you would expect Tave to increase requiring a control rod insertion.
- D. 1st part is incorrect but plausible because the MFW Master controller is in MANUAL . 2nd part is correct.

Technical Reference(s): STG-ICS Chap 4 Feedwater Subsystem
(Attach if not previously provided) STG-ICS Ch 4. Rev. 09b

Proposed references to be provided to applicants during examination: None

Learning Objective: STG-ICS Chap 4 Feedwater Subsystem Obj. R4b

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

Question History: Last NRC Exam _____
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 X
55.43 _____

Comments:

AA1 Ability to operate and/or monitor the following as they apply to the loss of condenser vacuum.:

(CFR 41.7 / 45.5 / 45.6)

AA1.04 Rod Position 2.5* 2.5*

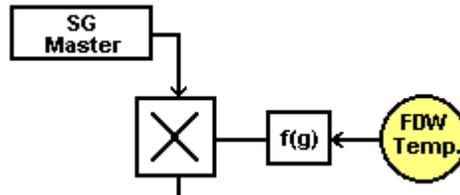
Author: Ken Schaaf

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OP-OC-STG-ICS FDW Subsection

(Obj. R12) Following the SG Master Controller the total FDW Demand is modified by the following signals prior to being divided into the individual FDW Loops demands:

(Obj. R14) Feedwater Temperature



At any given load, there is a balance of energy (BTU) exchange between primary and secondary during steady state conditions, and the BTU's transferred out of the primary should be equal to the BTU's transferred into the secondary side of the SG.

To maintain this balance, when feedwater temperature changes, the FDW demand is modified by a function of FDW temperature.

For any feedwater flow rate there is an expected FDW temperature.

If this normal relationship between flow and temperature is upset (isolation of a FDW heater) the parameter relationships used to program the ICS will be incorrect.

If feedwater temperature decreases with no change in flow rate, the colder fluid would draw more heat from the primary, causing T_{ave} to drop and reactor power to increase. More heat would be required to raise the feedwater to the proper SG outlet conditions.

This can be best explained using the heat transfer formula

$$Q = m\Delta h$$

With a reduction of FDW temperature, the enthalpy (h) of the secondary would decrease. This creates a larger enthalpy difference between the primary and secondary causing the heat transfer (Q) to increase across the steam generator.

To maintain a constant "Q", the mass flowrate of the FDW (M) is reduced by the ICS. This is performed in the FDW temperature modification circuit.

As feedwater flow decreases, so does steam flow out of the S/G and therefore the rest of the ICS will respond in an effort to maintain CTP at setpoint. The result will be a lower

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unit electrical megawatt output with CTP remaining at the set value.

The inverse would be true for **increasing feedwater temperatures.**

The controlling final FDW Temperature (FFT) fails to 447°F (the only credible failure). This is enough (~ -7°F at full power) to see a **small decrease in FDW flow and reactor power** due to the signal failure.

At a low power of ~ 25%, FFT is ~ 345°F. A failure of the controlling FFT to 447°F (~ +100°F change) will cause FDW to increase, lowering Tave, and cause reactor power to increase approximately 3%.

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Question 22

Given the following Unit 3 conditions:

- Reactor in MODE 6
- Core offload in progress
- Main Fuel Bridge is withdrawing a fuel assembly that appears to be binding

The __ (1) __ interlock will stop the withdrawal of the fuel assembly to prevent fuel Damage. The load setpoint for this interlock is __ (2) __.

- A. (1) Fuel Overload (TS-1)
(2) 2500 lb
- B. (1) Fuel Overload (TS-1)
(2) 2000 lb
- C. (1) Fuel Hoist Up/Down (TS-2)
(2) 2500 lb
- D. Fuel Hoist Up/Down (TS-2)
(2) 2000 lb

Proposed Answer: A

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Explanation (Optional):

- A. Correct
- B. 1st part is correct. 2nd part is incorrect but plausible because this is the normal reading during assembly withdrawal.
- C. 1st part is incorrect but plausible because this interlock will also stop upward travel of the hoist however it is only in effect when the grapple is disengaged. 2nd part is correct.
- D. 1st part is incorrect but plausible because this interlock will also stop upward travel of the hoist however it is only in effect when the grapple is disengaged. 2nd part is incorrect but plausible because this is the normal reading during assembly withdrawal.

Technical Reference(s): OP-OC-FH-FHS
(Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: OP-OC-FH-FHS Obj R36

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

Question History: Last NRC Exam 2011B
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 X
55.43 _____

Comments:

036AK3.02 Knowledge of the reasons for the following responses as they apply to the Fuel Handling Incidents:

(CFR 41.5 / 41.10 / 45.6 / 45.13)

AK3.02: Interlocks associated with fuel handling equipment. 2.9 3.6

Author: Ken Schaaf

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Question 23

Unit 1 initial conditions:

- Mode = 5
- Equipment hatch is open
- RB Purge is operation
- Fuel movement is in progress

Current conditions:

- 1SA-08 B-10 PROCESS MONITOR FAULT in alarm
- The Reactor Building Purge system has isolated

Based on the above conditions, UNIT VENT MONITOR 1-RIA _____ in HIGH alarm would have caused the isolation.

If the equipment hatch is closed 33 minutes later, it _____ meet the requirement for establishing containment closure in accordance with OP/1/A/1502/009, CONTIANEMNT CLOSURE CONTROL.

- A. 1) 44
2) does
- B. 1) 44
2) does not
- C. 1) 45
2) does
- D. 1) 45
2) does not

Proposed Answer: D

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- A. 1st part incorrect but plausible because RIA 44 monitors vent gas also. 2nd part is incorrect but plausible because if fuel movement was not in progress, it would be correct.
- B. 1st part incorrect but plausible because RIA 44 monitors vent gas also. 2nd part is correct.
- C. 1st part is correct. 2nd part is incorrect but plausible because if fuel movement was not in progress, it would be correct.
- D. Correct.

Explanation (Optional):

Technical Reference(s): OP-OC-RAD-RIA
(Attach if not previously provided) STG-ICS Ch 4. Rev. 09b

Proposed references to be provided to applicants during examination: None

Learning Objective: OP-OC-RAD-RIA Obj. R8

Question Source: Bank # _____

 Modified Bank # _____ (Note changes or attach parent)

 New X

Question History: Last NRC Exam _____

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____

 Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

 55.43 _____

Comments:

069 Loss of CTMT Integrity
G2.2.44 Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions:
(CFR 41.5 / 43.5 / 45.12)
4.2 4.4

Author: Ken Schaaf

RM
PROCESS MONITOR FAULT

1. Alarm Setpoint

1.1 Actuation of failure circuit on any of the process monitors.

2. Automatic Action

NOTE: 1. This Statalarm has reflash capability.

2. This Statalarm will **NOT** reflash, if another RIA on the same skid as the RIA which caused the initial actuation of the Statalarm actuates its failure circuit.

3. A Loss of Power to 1RIA-47, 48, 49, & 49A skid will result in 1LWD-2 interlock on high radiation. (Breaker 1KM-9)

4. Loss of Power to RIA skid will cause the High Alarm and associated interlocks will actuate.

1, 2, 3 RIA-43, 44, 45, 46 - Unit Vent Monitors (CPM)

Particulate (RIA-43), Iodine (RIA-44), Normal gas (RIA-45), High Gas (RIA-46) "PIGG"

RIAs-43 & 45 are plastic beta scintillation detectors.

RIA-44 is a NaI detector.

RIA-46 is a Cadmium Telluride solid state detector.

Located on 6th floor Auxiliary Building in the Purge Equipment room close to the Unit Vent Stack

On HIGH alarm, RIA-45 will do the following:

Close PR-2 through PR-5

Trip the main and mini purges

Actuates statalarm "RM Reactor BLDG Purge Disch RAD Inhibit"

When RIA-46 reaches the "switchover acceptance range setpoint", the following occurs:

RIA-45 will read zero

RIA-46 will now perform the same interlock functions that RIA-45 performed

This provides a backup function so that in case of a failure of RIA-45 HIGH alarm, then RIA-46 HIGH alarm will actuate the required interlock functions. Normally RIA-45 HIGH alarm setpoint will be reached prior to RIA-46 reaching the "switchover acceptance range setpoint".

RIA Swapovers

Under normal operating circumstances, when RIA 45/46 (and RIA 49/49A) are both in service, the RIA 45 (49) readings would increase to the high alarm setpoint and actuate the interlock. RIA 46 (49A) would continue to read zero on the RIA view screens while all this occurs. At this point, the interlock is NOT actuated by RIA 46 (49A). RIA 46 (49A) could actually be seeing some value (less than the 'switchover acceptance range setpoint'). Only when the 'switchover acceptance range setpoint' is reached will the RIA indicate a value.

If RIA 45 (49) is out of service, but RIA 46 (49A) is in service and the accepted range setpoint is left where it currently is, the activity in the Vent (Reactor Building) would increase with RIA 46 (49A) reading zero until the 'switchover accepted range setpoint' is met. At this point the interlock would be actuated because the 'switchover acceptance range setpoint' is currently set above the high setpoint. Based on current setpoints, the High Gas RIAs (46 & 49A) could be in alarm before the 'switchover acceptance range setpoint' is reached and the RIA begins to indicate something other than

zero. Therefore, the possibility exists such that the alarm setpoint is exceeded before the High Gas RIA alarms.

Enclosure 4.8

OP/1/A/1502/009

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**Completing
Penetration Contingency Actions
Paperwork**

4.7 Perform **one** of the following to determine "Time Available To Isolate Penetration":

NOTE: For RB Equipment Hatch removal Step 4.7.1 always applies regardless of RCS or fuel movement status.

4.7.1 **IF** RB Equipment Hatch removal is required, place N/A in blanks next to statement "Penetration shall be isolated by 'Time Available To Isolate Penetration'" on Enclosure 4.9 (Penetration Contingency Actions).

4.7.2 **IF** RCS loops filled and vented per OP/1/A/1103/002 (Filling And Venting RCS) **AND** SGs are available, place "2" in the hour blank next to statement "Penetration shall be isolated by 'Time Available To Isolate Penetration'" on Enclosure 4.9 (Penetration Contingency Actions).

4.7.3 **IF** fuel movement inside containment in progress **OR** FTC fill complete in prep for fuel movement inside containment, record 30 minutes next to statement "Penetration shall be isolated by 'Time Available To Isolate Penetration'" on Enclosure 4.9 (Penetration Contingency Actions).

4.7.4 **IF** containment closure is being maintained during defueled maintenance **AND** directed by OWPG, place N/A in blanks next to statement "Penetration shall be isolated by 'Time Available To Isolate Penetration'" on Enclosure 4.9 (Penetration Contingency Actions).

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Question 24

Unit 1 Initial conditions:

- Reactor power = 80% for the last week
- Chemistry reports the following:
 - RCS Gross Activity = 1.1 $\mu\text{Ci/gm}$
 - DEI activity = 0.7 $\mu\text{Ci/gm}$
- AP/21 (High Activity in RCS) entry has been made

Based on the above conditions, AP/21 _____ require power to be reduced and entry into TS 3.4.11 RCS Specific Activity _____ required.

Which ONE of the following completes the statements above?

- A. (1) does
(2) is
- B. (1) does
(2) is not
- C. (1) does not
(2) is
- D. (1) does not
(2) is not

Proposed Answer: B

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Explanation (Optional):

TS 3.4.11 requires entry if DEI activity is $>1.0 \mu\text{ci/gm}$. Per AP/21, step 4.10 & 11. If DEI is $\geq 0.5 \mu\text{ci/gm}$, it directs initiation of OP/1/A/1102/004 to reduce power by 10%FP.

- A. 1st part is correct per AP/21. 2nd part is incorrect but plausible because if DEI were $1.1 \mu\text{ci/gm}$ instead of gross activity, it would be correct.
- B. Correct (see above)
- C. 1st part is incorrect but plausible because in step 4.10 it states that if $1.0 \mu\text{ci/gm}$ DEI were exceeded, evaluate the need to reduce reactor power to maintain RCS activity within limits. 2nd part is incorrect but plausible because if DEI were $> 1.0 \mu\text{ci/gm}$ it would be correct.
- D. 1st part is incorrect but plausible because in step 4.10 it states that if $1.0 \mu\text{ci/gm}$ DEI were exceeded, evaluate the need to reduce reactor power to maintain RCS activity within limits. 2nd part is correct.

Technical Reference(s): AP/21 High Activity in RCS
 TS 3.4.11
(Attach if not previously provided) OP-OC-CH-RC. Rev. 08

Proposed references to be provided to applicants during examination: None

Learning Objective: OP-OC-CH-RC Obj. R10

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

Question History: Last NRC Exam _____
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 X
 55.43 _____

Comments:

076 High Reactor Coolant Activity
Ability to determine and interpret the following as they apply to the High Reactor Coolant Activity:
Corrective actions required for high fission product activity in RCS (CFR 43.5 / 45.6)
2.8/3.4
Author: Ken Schaaf

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4. Subsequent Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
NOTE	
The unit of measure $\mu\text{Ci/ml}$ may be used interchangeably with $\mu\text{Ci/gm}$ when comparing chemistry sample results with Tech Spec limits and conditions of this procedure. {2}	
4.1 IAAT DOSE EQUIVALENT I-131 > 1.0 $\mu\text{Ci/gm}$, THEN perform Steps 4.2 - 4.3.	1. __ Notify OSM to make notifications per OMP 1-14 (Notifications). 2. GO TO Step 4.4.
4.2 Notify OSM to reference the following: __ RP/0/B/1000/001 (Emergency Classification) __ OMP 1-14 (Notifications)	
4.3 __ Initiate action per T.S. 3.4.11 (RCS Specific Activity).	
4.4 __ Notify STA.	
4.5 __ Notify SOC that Unit 1 is unavailable for load following.	
4.6 __ Maximize letdown.	
4.7 __ Notify RP to obtain a RB sample.	
4.8 __ Notify Primary Chemistry to determine DOSE EQUIVALENT Xe-133.	
4.9 __ IAAT primary to secondary leak rate of ϵ 30 gpd (0.02082 gpm) is indicated using OAC point O1P1599 (EST TOTAL PRI TO SEC LEAKRATE), THEN notify Secondary Chemistry to determine Hotwell I-131. {3}	

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ACTION/EXPECTED RESPONSE		RESPONSE NOT OBTAINED
4.10 __ IAAT any of the following limits are exceeded:		
	Parameter	Limit
	RCS DEI	1.0 $\mu\text{Ci/gm}$
	RB Iodine	20 DAC
	Hotwell I-131 (if primary to secondary leak is indicated)	$1.0 \times 10^{-6} \mu\text{Ci/gm}$
THEN evaluate the need to reduce Rx power to stay within the limits in the table above.		
4.11 __ IAAT DOSE EQUIVALENT Xe-133 > 280 $\mu\text{Ci/gm}$, AND RCS Tave \leq 500 °F, THEN initiate action to comply with T.S. 3.4.11 (RCS Specific Activity).		
4.12 __ IAAT Rx power has a Steady State History of \pm 5% FP for at least 72 hours prior to obtaining the chemistry sample that indicated high activity, THEN perform Steps 4.13 - 4.15.		1. __ Determine time when 72 hours of steady state operation will occur. _____ 2. __ Request DEI sample for time listed above. 3. __ GO TO Step 4.16.
4.13 __ Verify DEI \geq 0.5 $\mu\text{Ci/gm}$.		__ GO TO Step 4.15.
NOTE		
The rate of Rx power reduction should be limited to \leq 3% FP/hr, however, this rate may be exceeded to comply with the limits of Tech Specs or as deemed necessary by the OSM.		
4.14 __ Initiate OP/1/A/1102/004 (Operation At Power) to reduce Rx power by 10% FP.		

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Question 25

Unit 2 plant conditions:

- Reactor power = 100%
- 2SA-18/A-11, TURBINE BSMT WATER EMERGENCY HIGH LEVEL, is in alarm
- Turbine Building flooding is confirmed
- The EOP is entered

In the Turbine Building Flooding tab of the EOP, Auxiliary Feedwater pumps ___(1)___ required to be utilized to fill the Steam Generators in addition to the Main Feedwater pumps.

While maximizing feed to the SGs, the feed rate limits of Rule 7 (SG Feed Control) ___(2)___ apply while maintaining $T_{ave} \geq 532$ °F.

- A. 1) are
2) do
- B. 1) are
2) do not
- C. 1) are not
2) do
- D. 1) are not
2) do not

Proposed Answer: A

Licensee to verify interpretation of the TBF Note & Step 1.Explanation (Optional):

- A. 1st part is correct. 2nd part is correct.
- B. 1st part is correct. 2nd part is incorrect but plausible because a note in the TBF tab of the EOP before step 1 states "Feeding to 95% O.R. in Step 1 supersedes guidance in Rule 7 (SG Feed Control), Table 4 (SG Level Control Points)).
- C. 1st part is incorrect but plausible because Main Feedwater is still available and can provide feedwater flow in excess of the limits of rule 7 without emergency feedwater. 2nd part is correct.
- D. 1st part is incorrect but plausible because Main Feedwater is still available and can provide feedwater flow in excess of the limits of rule 7 without emergency feedwater. 2nd part is incorrect but plausible because a note in the TBF tab of the EOP before step 1 states "Feeding to 95% O.R. in Step 1 supersedes guidance in Rule 7 (SG Feed Control), Table 4 (SG Level Control Points)).

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Technical Reference(s): EOP TBF

(Attach if not previously provided) OP-OC-EAP-TBF. Rev. 10a

Proposed references to be provided to applicants during examination: None

Learning Objective: OP-OC-EAP-TBF Obj. R2&3

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

Question History: Last NRC Exam _____
 (Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 X
 55.43 _____

Comments:

BW/A07 Flooding

Knowledge of the operational implications of the following concepts as they apply to the (Flooding):

Components, capacity, and function of emergency systems
 (CFR 41.8 / 41.10 / 45.3)
 3.5/3.5

Author: Ken Schaaf

Rule 7

Table 4			
SG Level Control Points			
NOTE			
Flow may be throttled as necessary to control cooldown during the approach to the SG Level Control Point.			
Plant Condition	Main FDW Pump	EFDW Pump	SSF ASW Pump
<u>All</u> SCMs > 0°F AND <u>any</u> RCP on	25" - 35" [55" - 65" acc] S/U level	30" [60" acc] XSUR (use MFDW setpoint if feeding via S/U CVs)	30" [60" acc] XSUR
<u>All</u> SCMs > 0°F AND <u>all</u> RCPs off	50% [50% acc] Operating Range	240" [270" acc] XSUR (use MFDW setpoint if feeding via S/U CVs)	240" [270" acc] XSUR
<u>Any</u> SCM = 0°F AND NO SSF Event*	95% [95% acc] Operating Range	LOSCM setpoint (Band: +0"/-5")	LOSCM setpoint (Band: +0"/-5")

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		(Turn-on code "EFW" or Per Table 5)	(Turn-on code "EFW" or Per Table 5)
<u>Any</u> SCM = 0°F AND SSF Event*	N/A	240" [270" acc] XSUR	Per AP/25
Superheated with CETCs ≤ 1200°F	95% [95% acc] Operating Range	LOSCM setpoint (Band: +0"/-5") (Turn-on code "EFW" or Per Table 5)	LOSCM setpoint (Band: +0"/-5") (Turn-on code "EFW" or Per Table 5)
Superheated with CETCs > 1200°F	Per Encl 5.15 (ICC Full Range SG Level)	Per Encl 5.15 (ICC Full Range SG Level)	Per Encl 5.15 (ICC Full Range SG Level)

SSF Event* - SSF activated per AP/25 with SSF RC Makeup required. {31}

TBF

Turbine Building Flood

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
---------------------------------	------------------------------

<p><u>Unit Status</u> AP/10 is being used in parallel with this tab.</p>

<p><u>NOTE</u> Feeding to 95% O.R. in Step 1 supercedes guidance in Rule 7 (SG Feed Control), Table 4 (SG Level Control Points).</p>
--

<p>___ Initiate feeding SGs to 95% O.R. at maximum allowable rate using all available feedwater sources while maintaining the following:</p> <ul style="list-style-type: none"> • Initial T_{ave} ≥ 532°F - maintain T_{ave} ≥ 532°F during SG fill. • Initial T_{ave} < 532°F - minimize cooldown during SG fill utilizing TBVs as required. 	
--	--

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Question 26

Unit 3 initial plant conditions:

- Reactor power = 100%
- SBLOCA occurs

Current conditions:

- Reactor has tripped
- ALL SCM's = "0"
- 3C HPI pump fails to start
- 3-HP-409 is open
- "A" HPI Header flow = 485 gpm
- "B" Cross Header flow = 410 gpm
- Seal injection = 40 gpm

1) HPI flow __ (1) __ have to be throttled in accordance with Rule 6 (HPI).

2) HPI flow __ (2) __ be adequate for the worst case SBLOCA.

- A. (1) does
(2) will
- B. (1) does
(2) will not
- C. (1) does not
(2) will
- D. (1) does not
(2) will not

Proposed Answer: **C**

Licensee to verify actual HPI flow limit.

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Explanation (Optional):

- A. 1st part is incorrect but plausible because if one HPI pump were operating in a header, it would be correct. 2nd part is correct.
- B. 1st part is incorrect but plausible because if one HPI pump were operating in a header, it would be correct. 2nd part is incorrect but plausible because if HP-409 were closed, it would be correct.
- C. 1st part is correct. 2nd part is correct.
- D. 1st part is correct. 2nd part is incorrect but plausible because if HP-409 were closed, it would be correct.

Technical Reference(s): EOP LOSCM
EOP Rule 2 LOSCM

(Attach if not previously provided) OP-OC-EAP-TBF. Rev. 10a

Proposed references to be provided to applicants during examination: None

Learning Objective: OP-OC-EAP-LOSCM Obj. R43

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

Question History: Last NRC Exam _____
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 X
55.43 _____

Comments:

BW/E03 Inadequate Subcooling Margin
EA2.2 Ability to determine and interpret the following as they apply to the (Inadequate Subcooling Margin):

Adherence to appropriate procedures and operation within the limitations in the facilities license and amendments.

(CFR 43.5 / 45.13)

3.5/4.0

Author: Ken Schaaf

Oconee Nuclear Station 2013-301 Written Examination

Rule 6

HPI

HPI Pump Throttling Limits

- HPI must be throttled to prevent violating the RV-P/T limit.
- HPI pump operation must be limited to two HPIPs when only one BWST suction valve (3HP-24 or 3HP-25) is open.
- HPI must be throttled ≤ 475 gpm/pump (including seal injection for A header) when only one HPI pump is operating in a header.
- Total HPI flow must be throttled ≤ 950 gpm including seal injection when 3A and 3B HPI pumps are operating with 3HP-409 open.
- Total HPI flow must be throttled < 750 gpm when all the following conditions exist:
 - LPI suction is from the RBES
 - piggyback is aligned
 - either of the following exist:
 - only one piggyback valve is open (3LP-15 or 3LP-16)
 - only one LPI pump operating
- HPI may be throttled under the following conditions:

HPI Forced Cooling in Progress:	HPI Forced Cooling NOT in Progress:
<p><u>All</u> the following conditions must exist:</p> <ul style="list-style-type: none"> • <u>Core</u> SCM > 0 • CETCs decreasing 	<p><u>All</u> the following conditions must exist:</p> <ul style="list-style-type: none"> • <u>All</u> WR NIs $\leq 1\%$ • <u>Core</u> SCM > 0 • Pzr level increasing • SRO concurrence required if throttling following emergency boration

HPI Pump Minimum Flow Limit

- Maintain ≥ 170 gpm indicated/pump. This is an instrument error adjusted value that ensures a real value of ≥ 65 gpm/pump is maintained. HPI pump flow less than minimum is allowed for up to 4 hours.

Step 9: Verify all the following conditions exist:

- **NO** RCPs operating
- HPI flow in both HPI headers
- Adequate total HPI flow per Figure 1 (Total Required HPI Flow)

Step 9: **RNO GO TO** Step 11.

The first bullet is a verification that RCPs were secured as required within the two minute criteria.

If two minutes were exceeded and the RCPs were not secured, a rapid RCS cooldown will be performed to obtain injection from the CFTs and LPI system. This minimizes the exposure time where a failure of RCPs could result in core uncover. See RNO for step 9.

The purpose of the second and third bullets in this step is to verify HPI flow and provide actions to cool down the plant as fast as possible in the unlikely event that sufficient HPI flow is not available/degraded.

Successful mitigation of Small Break LOCA (SBLOCA)

One HPI train provides sufficient flow to mitigate most small break LOCAs (per DBD). SBLOCAs require the proper operation of at least one HPI pump. The curve is conservative from an actual HPI pump head curve. However, for cold leg breaks located on the discharge of the reactor coolant pumps or an HPI line break downstream of the last check valve, some HPI injection will be lost out the break; for these two cases, two HPI trains are required.

Two HPI trains are required to mitigate specific small break LOCAs, if no credit for enhanced steam generator cooling is assumed in the accident analysis. However, if equipment not qualified as QA-1 (i.e., an atmospheric dump valve (ADV) flow path for a steam generator) is credited for enhanced steam generator cooling, the safety analyses have determined that the capacity of one HPI train is sufficient to mitigate a small break LOCA **if reactor power is $\leq 75\%$ RTP** (TS 3.5.2).

Therefore, to cover all potential cases; HPI flow must be established in both headers in order to avoid the necessity of a rapid cooldown.

At least two HPIPs must be running in order to supply flow in both headers.

With only one HPIP operating, opening the crossover valves (HP-409 or HP-410) is not allowed per Rule 2. The reason this is not allowed is because this would increase the possibility of HPIP runout and would require taking multiple instrument errors into consideration. It would be difficult to determine true HPI flow with this set of plant conditions

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- . If the total HPI flow being delivered is less than the curve, if flow cannot be aligned through both HPI headers, OR RCPs remain on > 2 minutes; then action must be taken to increase the amount of core cooling available. See RNO for step 9.*

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Question # 27

Unit 2 initial conditions:

- Main steam line break occurred on the 2A SG outside of containment
- The EHT tab of the EOP was completed
- The crew transitioned to the Forced Cooldown (FCD) Tab

Current plant conditions:

- The decision has been made to perform a natural circulation cooldown
- SCM is currently 105 °F

Level in the 2B SG should be maintained at __ (1) __ while performing the cooldown and the current RCS subcooling margin __ (2) __ meet the minimum requirement in the FCD tab to begin RCS depressurization as part of the cooldown.

- A. (1) 240" XSUR
(2) does
- B. (1) 240" XSUR
(2) does not
- C. (1) 270" XSUR
(2) does
- D. (1) 270" XSUR
(2) does not

Proposed Answer: B

Could give them a conditions that would make ACC in effect which would make D correct.

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Explanation (Optional):

- A. 1st part is correct. 2nd part is incorrect but plausible because 105 °F is a significant SCM.
- B. Correct
- C. 1st part is incorrect but plausible because if the break was in containment, it would be correct. 2nd part is incorrect but plausible because 105 °F is a significant SCM.
- D. 1st part is incorrect but plausible because if the break was in containment, it would be correct. 2nd part is correct.

Technical Reference(s): EOP FCD
EOP Rule 7 SG Feed

(Attach if not previously provided) OP-OC-EAP-FCD. Rev. 7a

Proposed references to be provided to applicants during examination: None

Learning Objective: OP-OC-EAP-FCD Obj. R5

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

Question History: Last NRC Exam _____
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 X
55.43 _____

Comments:

**BW/E09 Natural Circulation Cooldown
EA2. Knowledge of the interrelations between the (Natural Circulation
Cooldown) and the following:**

Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.

(CFR 41.7 / 45.7)

4.0/4.0

Author: Ken Schaaf

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Question #28

Unit 1 initial conditions:

- Reactor power is in the power range
- 1RC-1 fails open
- RCS pressure = 2200 psig decreasing

Current plant conditions:

- RCS pressure = 2100 psig decreasing
- 1RC-1 and 1RC-3 fail to close
- 1AP-44 "Abnormal Pressurizer Pressure Control" is entered

In accordance with 1AP-44, the ___(1)___. IF the reactor is tripped, the ___(2)___ stopped, then if pressure continues to decrease additional RCPs are stopped.

- A. (1) reactor is required to be tripped regardless of power
(2) 1A1 RCP ONLY is
- B. (1) reactor is required to be tripped regardless of power
(2) 1A1 and 1A2 RCPs are
- C. (1) reactor is required to be tripped only if above 70% power
(2) 1A1 RCP ONLY is
- D. (1) reactor is required to be tripped only if above 70% power
(2) 1A1 and 1A2 RCPs are

Proposed Answer: B

Explanation (Optional):

- A. 1st part is correct. 2nd part is incorrect but plausible because the spray line tap is off of the 1A1 RCP discharge.
- B. 1st and 2nd part is correct.
- C. 1st part is incorrect but plausible because 3 RCP operation at power is allowed with some restrictions. Also, there is an ICS runback to 74% if a RCP trips and the Operations at Power procedure (OP/1/A/1102/004) has a step 2.44 where additional actions prior to exceeding 70% power. It's a familiar number associated with 3 RCP operations. 2nd part is incorrect but plausible because the spray line tap is off of the 1A1 RCP discharge.

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D. 1st part is incorrect but plausible because 3 RCP operation at power is allowed with some restrictions. Also, there is an ICS runback to 74% if a RCP trips and the Operations at Power procedure (OP/1/A/1102/004) has a step 2.44 where additional actions prior to exceeding 70% power. It's a familiar number associated with 3 RCP operations. 2nd part is incorrect.

Technical Reference(s): 1AP-44 Rev 2
 OP/1/A/1102/004 Rev 137
 OP-OC-EAP-APG

(Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: OP-OC-EAP-APG Obj. R9

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

Question History: Last NRC Exam _____
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 X
 55.43 _____

Comments:

003 Reactor Coolant Pump
A1.06 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RCPS controls including:
PZR spray flow.
(CFR 41.5 / 45.5)
2.9/3.1

Author: Ken Schaaf

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Question # 29

Unit 3 plant conditions:

- Reactor power is in the power range
- 3 RCPs operating

To meet the interlock to start the 4th RCP, the maximum reactor power is (1). When the idle RCP is started, the indicated Tc in that loop is initially expected to (2).

- A. (1) 40%
(2) remain the same
- B. (1) 40%
(2) increase
- C. (1) 50%
(2) remain the same
- D. (1) 50%
(2) increase

Proposed Answer: C

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Explanation (Optional):

- A. 1st part is incorrect but plausible because this is the procedural maximum power to start a RCP. 2nd part is correct.
- B. 1st part is incorrect but plausible because this is the procedural maximum power to start a RCP. 2nd part is incorrect but plausible because if there were no reverse flow, it would be correct.
- C. Correct.
- D. 1st part is correct. 2nd part is incorrect but plausible because if there were no reverse flow, it would be correct.

Technical Reference(s): OP-OC-PNS-CPM Rev 18
 OP/3/A/1103/006 Rev 54

(Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: OP-OC-PNS-CPM Obj. R19

Question Source: Bank # _____

 Modified Bank # (Note changes or attach parent)

 New X

Question History: Last NRC Exam _____

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

 Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X

 55.43

Comments:

003 Reactor Coolant Pump
K4.02. Knowledge of RCPS design feature(s) and/or interlock(s)
which provide for the following:
Prevention of cold water accidents or transients.
(CFR 41.7)
2.5/2.7

Author: Ken Schaaf

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Question 30

Unit 1 initial conditions:

- Reactor power = 100%
- A loss of letdown occurs
- 1AP/32 Loss Of Letdown is entered

When calculating HPI pump flow using control room indications, the combination of Seal Injection and Makeup flow shall be a minimum of ___(1)___ or a cumulative time limit of 4 hours is place on their operation. This time limit is based on ___(2)___.

- A. (1) 35 gpm
(2) wear on the impeller seal ring
- B. (1) 35 gpm
(2) vapor binding of the pump
- C. (1) 65 gpm
(2) wear on the impeller seal ring
- D. (1) 65 gpm
(2) vapor binding of the pump

Proposed Answer: A

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- A. Correct.
- B. 1st part is correct. 2nd part is incorrect but plausible because minimum pump flows can be based on overheating the pump resulting in cavitation and vapor binding.
- C. 1st part is incorrect but plausible because recirc flow plus seal injection plus makeup flow has to be at least 65 gpm or the 4 hour time limit is placed on their operation. 2nd part is correct.
- D. 1st part is incorrect but plausible because recirc flow plus seal injection plus makeup flow has to be at least 65 gpm or the 4 hour time limit is placed on their operation. 2nd part is incorrect but plausible because minimum pump flows can be based on overheating the pump resulting in cavitation and vapor binding.

Explanation (Optional): Recirc flow is not monitored in the control room so it is assumed to be 30 gpm and added to the combination of seal injection and makeup flows. The total flow has to be a minimum of 65 gpm. What the operator has indication of in the control room has to add up to at least 35 gpm.

Technical Reference(s): OP-OC-PNS-HPI Rev 28b
OP/1/A/1104/002 Rev 159
AP/1A/1700/032 Rev 6

(Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: OP-OC-PNS-HPI Obj. R4

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

Question History: Last NRC Exam _____
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 X
55.43 _____

Comments:

004 Chemical and Volume Control
G2.1.32 Ability to explain and apply system limits and precautions.
(CFR 41.10/43.2/45.12)
3.8/4.0
Author: Ken Schaaf

Precautions & Limitations

NOTE: When calculating HPI pump flow, assume HPI pump recirc flow is 30 gpm.

2.20 HPI Pump flow limits during normal operation: (OSC-7709)

- > 65 gpm: Indefinite operation
- 30 - 65 gpm: ≤ 4 hrs or RCP starts.

1/AP/32

NOTE

The running HPI pump may operate below 65 gpm for up to 4 hours. HPI pump time of operation below minimum flow is cumulative.

1.1 ___ Verify HPI pump flow ≥ 65 gpm.
 $\frac{30}{\text{Recirc}} + \frac{\quad}{\text{SI}} + \frac{\quad}{\text{MU}} = \text{___ gpm}$

___ Log beginning time for HPI pump flow below minimum.

OP-OC-PNS-HPI

(OBJ. R4) Normal Injection

High Pressure Injection Pumps

Three high pressure injection pumps are provided with the capability to take suction from the Letdown Storage Tank during normal operation and inject coolant into the Reactor Coolant System.

HPI Pump Auto Start Signals:

ES Channel 1 – A/B HPIPs

ES Channel 2 – B/C HPIPs

Main Feeder Bus Monitor Panel signal
(Reference EL-EPSL).

Low Seal Injection Flow (A/B HPIPs **Only**)

≤ 22 gpm U1, ≤ 30 gpm U2/3

All auto starts are blocked when on PSW power.

Each high pressure injection pump is a 600 HP, 24 stage, vertical centrifugal pump with a design discharge pressure of 3040 psi @200°F, 3120 psi @ 150°F.

Continuous operation of an HPI pump requires flow through the pump to be ≥ 65 gpm. During normal operation this is accomplished by having seal injection flow (32 gpm or 40 gpm) plus 35 to 45 gpm makeup flow and ~ 30 gpm minimum recirc flow. If flow lowers into the 30 gpm to < 65 gpm range we are limited to 4 hours of operation after which an engineering evaluation is required. A minimum flow of 30 GPM is insured on each HPI pump when running by a minimum recirculating line for each pump and flow of 30-65 gpm is permitted for up to 4 hours.

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Question 31

Unit 2 Initial Conditions:

- The Reactor is in MODE 6.
- The Fuel Transfer Canal is flooded.

Unit 2 Current Conditions:

- All LPI pumps trip and cannot be restored.
- The crew has entered AP/2/A/1700/026, Loss of Decay Heat Removal.
- 2LP-1 and 2LP-2 have the ability to be opened.

Which ONE of the following describes the appropriate actions to take in accordance with AP/2/A/1700/026?

- A Initiate Enclosure 5.7 (DHR Using SF Cooling) and align the "B" SF cooling pump to take suction from the decay heat drop line and discharge to the SF pool.
- B Initiate Enclosure 5.7 (DHR Using SF Cooling) and align the "B" SF cooling pump to take suction from the FTC deep end and discharge to the SF pool.
- C Align HPI for injection mode with water flowing through the core and out the lowest RCS opening into the FTC.
- D Align HPI for injection mode with water flowing through the core and out the lowest RCS opening into the RB Basement.

Proposed Answer: A

K/A Match Analysis

The question requires the applicant to determine the appropriate general strategy to take when normal DHR is lost during refueling.

Answer Choice Analysis

- A. CORRECT.
- B. INCORRECT. Plausible because this would be correct if 2LP-1 and 2LP-2 could NOT be opened.
- C. INCORRECT. Plausible because this would be correct if the FTC was NOT flooded and the canal seal plate WAS installed.
- D. INCORRECT. Plausible because this would be correct if the FTC was NOT flooded and the canal seal plate was NOT installed.

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Technical Reference(s): Lesson Plan OP-OC-PNS-LPI, Rev 25a, page 33-34, 45
(Attach if not previously provided) Lesson Plan OP-OC-TA-DHR, Rev 09c, pages 36-37
(including version/revision number) Lesson Plan OP-OC-FH-FHS, Rev 18, page 22
AP/2/A/1700/26, Rev 21, Section 4C
AP/2/A/1700/26, Rev 21, Section 4D
AP/2/A/1700/26, Rev 21, Enclosure 5.7

Proposed references to be provided to applicants during examination: None

Learning Objective: Lesson Plan OP-OC-PNS-LPI Obj. R31
Lesson Plan OP-OC-TA-DHR Obj. R14, R16, R18
Lesson Plan OP-OC-FH-FHS Obj. R47

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

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content: 55.41 X
55.43 -

Comments:

005K3.07 Knowledge of the effect that a loss or malfunction of the RHRS will have on the following: Refueling operations

3.2/3.6

Author: Amanda Toth

Supporting References (excerpted from lesson plans)

OP-OC-PNS-LPI, Rev 25a, page 33-34

Operation of the LPI System during Refueling

Aside from cooling down and maintaining RCS temperature in preparation for refueling, the LPI System is used to fill the Fuel Transfer Canal (FTC) simultaneously with the SFC system once the reactor vessel head has been removed.

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The fuel transfer canal and incore instrument handling tank are filled from the BWST by simultaneously using the Spent Fuel Cooling System and the LPI system. Refer to (OC-PNS-LPI-8)

The B SF Cooling Pump will be secured and then realigned to fill the FTC from BWST at a flow rate of 800 to 1000 gpm by throttling SF-23.

The B Train of the LPI System suction will be aligned from the BWST via LP-22. Two LPI Pumps will be used. Flow rate to the FTC will be maintained at 2400 to 3000 gpm while DHR is maintained at 1800 to 2000 gpm.

Total flow rate into the FTC will therefore be 3200-4000 gpm.

This flow rate is made possible by the BWST vent mod.

WHEN PZR level is 300 inches **OR** BWST reaches 9 feet, the B LPI pump is secured and realigned for normal DHR.

During refueling, the water level is substantially above the reactor vessel flange and there is mixing of the water in the reactor and the fuel transfer canal. **The Low Pressure Injection System will be in the normal decay heat removal mode and the Spent Fuel Cooling System in its normal operating mode.**

Once refueling has been completed, both the LPI and the SFC system is used simultaneously to drain the FTC. Refer to (OC-PNS-LPI-9)

The Refueling Mode of the Spent Fuel Cooling System will be secured and the lineup will be made to drain the FTC to the BWST using the "2B" Train of the LPI System. Suction will be from the "2A" Hot Leg via the normal LPI Drop Line. Two LPI Pumps will be used. Flowrate to the BWST will be maintained at 2400 to 3000 gpm while DHR is maintained at 1800 to 2000 gpm.

2LP-14 will be used to control flowrate to the BWST.

2LP-12 will be used to control DHR flowrate.

While the "B" LPI Train is draining the FTC, the "B" SF Cooling Pump will be aligned to take suction from the deep end of the FTC and discharge to the BWST. Flow rate through this flowpath will be 800 -1000 gpm.

SF-51 will be throttled to maintain this flow rate.

Total flow rate into the BWST will therefore be 3200 to 4000 gpm.

This flow rate is made possible by the recent BWST vent MOD.

WHEN PZR level = 100";

Throttle 2LP-14 to 800 - 1000 gpm

Stop second LPI Pump

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The LPI System may be used to pump the shallow portion of the FTC down to just below the RV flange by opening LP-40 ("B" LPI Train Recirc) and then throttling open the LPI Recirculation Line (LP-42) to the BWST.

NOTE: LP-40 and LP-42 were replaced in answer to a recent PIP which related a problem with LP-40 leaking past the seat, due to not being fully closed by the operator. These valves are now 90° ball valves.

Once the shallow end of the FTC is empty, the SFC System is used to complete pumping the deep end of the FTC back to the BWST.

OP-OC-PNS-LPI, page 45

(OBJ.R31) Refer to AP/1-2-3/A/1700/26, Loss of DHR.

AP is subdivided into four cases:

Section 4A: RCS Intact, and Full, With SGs Available

Section 4B: RCS Intact, RC Loops Not Full

Section 4C: RCS Vented

Section 4D: Fuel Transfer Canal Flooded

If LPIP flow has degraded:

- LPIP delta P < 111 psid
- LPI decay heat flow < 1000 gpm/header
- LPIP motor amps are fluctuating, then trip the running LPIP and Evacuate the Reactor Building.

The AP will then check the LPI System and supporting systems for failures and systematically step through returning any failed systems to operation.

If LPI is lost during refueling operations and cannot be restored, then utilize the SFP Cooling Pumps to maintain core temperatures via the SFP Coolers recirculating the Transfer Canal to the SFP.

The Loss of Decay Heat Removal AP will be covered in more detail in Lesson Plan TA-DHR

OP-OC-TA-DHR, Rev 09c, pages 36-37

R14, R16, R18) Mitigation of Loss of Decay Heat Removal (AP/A/1700/026)

Purpose: Provides specific guidance (SECTIONS) to mitigate a loss of DHR due to any of the AP entry conditions based on one of the following plant configurations:

RCS Intact and RC Loops Full

RCS Intact and RC Loops NOT full

RCS Vented and FTC NOT Flooded

FTC Flooded (SF-1 or SF-2 capable of being opened)

Entry Conditions:

Loss of RCS Inventory while on LPI DHR as directed from AP/2 (Excessive Leakage)

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Loss of DHR capability as a result of any of the following:

Degraded LPI flow resulting in an increase in temperature outside the control band as indicated on the applicable temperature instrument (Tc, CETC or LPI cooler outlet)

Loss of all running LPI Pump(s) (including loss due to slow power transfer or total loss of power)

Inadequate LPSW to LPI coolers resulting in an increase in temperature outside the control band as indicated on the applicable temperature instrument (Tc, CETC or LPI cooler outlet)

Unit Differences

Unit 1 is most susceptible to loss of suction to the LPI pumps due to the loop raised between 1LP-2 and 1LP-3.

(R13) Prior to draining below 100" PZR level, verify the flange is installed on the Emergency Sump Line "A" (LP-19) and the flange vents are closed. (This is only in Unit 1's procedure.)

This allows for an alternate LPI pump suction flow path via LP-1, LP-2 and LP-105. On Unit 1, the decay heat drop line penetrates the RB at a higher elevation than Units 2 & 3. The centerline of the drop line high point corresponds to an elevation equivalent to ~ 95" on LT-5.

The concern here is that if LT-5 is < 95", and air entrainment occurs in the drop line (due to vortexing, etc.), this air could settle out at the high point of the drop line penetration, and ultimately cause a loss of suction to the LPI pumps. Therefore, the flange is installed to provide an alternate suction flow path (which penetrates the RB at a much lower elevation), which can be used to provide DHR flow while actions are being taken to vent the normal drop line high point in the Penetration Room.

AP/2/A/1700/26, Rev 21, Section 4D, page 7

Section 4D

Fuel Transfer Canal Flooded

ACTION/EXPECTED RESPONSE		RESPONSE NOT OBTAINED
Unit Status		
LPI pumps OR LPI flow have been lost.		
19.	<input type="checkbox"/> Verify LPI Pumps were secured due to loss of inventory.	<input type="checkbox"/> GO TO Step 21.
20.	<input type="checkbox"/> GO TO Step 62.	
21.	<input type="checkbox"/> IAAT any of the following exist: <input type="checkbox"/> Attempts to restore DHR are unsuccessful <input type="checkbox"/> Visual observation indicates RCS	

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Encl	boiling/steaming THEN initiate 5.7 (DHR Using SF Cooling).	
------	---	--

AP/2/A/1700/26, Rev 21, Section 4C, pages 15 and 73

Section 4C

RCS Vented and Fuel Transfer Canal NOT Flooded

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
Unit Status	
LPI pumps OR LPI flow have been lost.	
39. ___ Verify LPI Pumps were secured due to loss of inventory.	___ GO TO Step 41.
40. ___ GO TO Step 84.	
21. ___ IAAT any of the following exist: ___ CETC > 180 <input type="checkbox"/> F ___ Visual observation indicates RCS boiling/steaming ___ It is determined that the failure CANNOT be restored. THEN GO TO Step 212.	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
Unit Status	
LPI DHR is NOT available.	
212. ___ IAAT all of the following exist: ___ All leaks are isolated/repaired (if leak existed) ___ LPI/LPSW is available for DHR ___ RV level > 10"	

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<p>THEN GO TO Step 287.</p>	
<p>213. Maximize RB Cooling by performing the following:</p> <ul style="list-style-type: none"> ___ Ensure all available RBCUs operating in HIGH. ___ Open 2LPSW-18. ___ Open 2LPSW-21. ___ Open 2LPSW-24. ___ Ensure all available RB AUX COOLERS in service. 	
<p>214. ___ Verify HPI system is available.</p>	<p>___ GO TO Step 238</p>
<p align="center">NOTE</p> <p>The following steps will align HPI in injection mode for core cooling. Water will flow through the core and out the lowest RCS opening into the RB basement or Fuel Transfer Canal.</p>	
<p>215. ___ Verify HPI system is available.</p>	<p>___ GO TO Step 217</p>
<p>216. ___ GO TO Step 224.</p>	

AP/2/A/1700/26, Rev 21, Enclosure 5.7

DHR Using SF Cooling

<p align="center">NOTE</p> <ul style="list-style-type: none"> • If 2LP-1 and 2LP-2 can be opened, this Enclosure aligns B SF Cooling Pump suction to the decay heat drop line with discharge to the SF Pool. • If 2LP-1 or 2LP-2 CANNOT be opened, this enclosure aligns B SF Cooling Pump suction to the FTC deep end with discharge to the SF Pool.

OP-OC-FH-FHS, Rev 18, page 22

During refueling, Operations uses the "A" or "C" SFC pump taking a suction from the SF pool and discharging back to the SF pool. The LPI system is lined up for normal decay heat removal taking suction off the hot leg and discharging to the reactor vessel. To keep the canal and the SF pool at equilibrium boron, the "B" SF cooling pump is aligned to take suction on the fuel transfer canal and discharges to the SF pool. The water then returns to the canal through SF-1 and SF-2 and the transfer tubes. To enhance water clarity during fuel loading and unloading "B" SF pump suction may be aligned to the decay heat drop line through LP-24.

(OBJ. R47) AP/1700/26 (Loss of Decay Heat Removal) Section 4D (FTC Flooded)

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Question # 32

Unit 1 conditions:

- A Reactor trip from 100% power has occurred due to a Small-Break LOCA.
- Enclosure 5.1, step 73 is in progress.
- 1HP-24 and 1HP-25 failed CLOSED.
- '1A', '1B' and '1C' HPI Pumps are running.
-

Which ONE of the following completes the statements:

- 1) '1A' and '1B' LPI pumps __ (1) __ required to be started.
- 2) One HPI pump __ (2) __ required to be secured.

- A 1) Are
 2) Is
- B 1) Are
 2) Is not
- C 1) Are not
 2) Is
- D 1) Are not
 2) Is not

Proposed Answer: B

K/A Match Analysis

This question requires the applicant to identify that the BWST suction to the HPI pumps have failed closed. When the BWST suction to the HPI pumps fail closed, the LPI pumps' discharge (via a line at the discharge of the LPI cooler) provides the suction to the HPI pumps. Since there is no indication that one flow path is unavailable, there is no restriction on the number of operating HPI pumps.

Answer Choice Analysis

- A. INCORRECT. The first part is correct. The second part is incorrect because there is no indication that only one flow path is available.
- B. CORRECT.
- C. INCORRECT. The first part is incorrect because with both suction valves failing closed, LPI pumps are required to be started to provide suction to the HPI pumps. The second part is incorrect because there is no indication that only one flow path is available.
- D. INCORRECT. The first part is incorrect because with both suction valves failing closed, LPI pumps are required to be started to provide suction to the HPI pumps. The second part is

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correct.

Technical Reference(s): Lesson Plan OP-OC-PNS-LPI, Rev 25a, pages 40-41
(Attach if not previously provided) Lesson Plan OC-PNS-HPI, Rev 28b, pages 11-12
(including version/revision number) EOP Enclosure 5.1, step 73

Proposed references to be provided to applicants during examination: None

Learning Objective: Lesson Plan OP-OC-PNS-LPI Obj. R26

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

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis

10 CFR Part 55 Content: 55.41
55.43

Comments:

006K6.01 Knowledge of the effect of a loss or malfunction on the following will have on the ECCS: BIT/borated water sources

3.4/3.9

Author: Amanda Toth

Supporting References (excerpted from lesson plans)

From OP-OC-PNS-LPI, Rev 25a, page 40-41

During LOCAs, the LPI System can also supply suction to the HPI System. The connection to the HPI System is via a line located at the discharge of each LPI cooler to the suction header of an HPI Pump.

These paths are normally isolated with an electrically operated valve in each line. They can be used to provide additional cooling for the HPI System during emergency injection, for providing a long-term source of water during emergency injection, or for providing an alternate suction source for the HPI Pumps if the suction valves from the BWST should fail closed. The manual isolation valves in this flowpath, LP-54 and LP-56, are kept locked open.

On an actuation of ES Channels 1 and 2, if HP-24 and HP-25 fail to open, the CRO will align the HPI/LPI systems in the piggyback mode to ensure a suction path to the HPI Pumps

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from the BWST. Licensed Operators are required to perform this lineup using Rule 2 of the EOP. Failure to perform this lineup in a timely manner could result in loss of LDST inventory and eventual damage to the HPI Pumps.

Power supplies to LP-15 and 16 have been moved to safety related power supplies due to their critical role during Piggyback and the dependence on these valves.

From OC-PNS-HPI, Rev 28b, pages 11-12:

Design Bases for HPI System

During emergency operation, the system is designed to supply borated water to the RCS and RCP seals. The water added directly to the system makes up for water lost due to a primary side leak (Core Flood line, RCP Discharge, HPI Nozzle, SGTL, etc.) or for shrinkage of the RCS due to cooling caused by a secondary side break. Maintaining RCP seal integrity is important for all but LBLOCAs.

The system shall be capable of supplying required flow rates via automatic actuation within 48 sec. of an ES signal actuation, and withstand credible single failure criteria. For most break sizes, one train of HPI can provide sufficient HPI flow. For certain size SBLOCA's that could occur at the RCP discharge, both trains of HPI are required to mitigate the accident. Thus, three HPI pumps must be operable to ensure adequate cooling is available for the design basis RCP discharge SBLOCA.

Additionally, in the event one HPI train fails to actuate due to a single failure, operator actions from the Control Room are required to cross-connect the HPI discharge headers within 10 minutes in order to provide HPI flow through a second HPI train.

The discharge of the HPI Pumps shall be throttled to limit any pumps flow rate to 475 gpm within 10 minutes of an ES event (Operator critical action). This limit includes maximum flow instrument error and minimum recirc flow. This prevents pump runout and assures adequate NPSH.

Flow shall be controlled from the Control Room either by stopping pumps or by throttling valves HP-26, HP-27, HP-409, and HP-410.

To ensure adequate NPSH during piggyback operation, ensure one HPI Pump per header when only 1 LPI/HPI path is capable of supplying flow. While in piggyback operations with suction from the RBES, total HPI flow, not the number of running HPI pumps, is the most limiting factor.

The system also provides various essential functions for normal operation, such as letdown for water purification, and boron concentration control.

E. Redundant components and alternate flow paths are provided to improve system reliability.

From Enclosure 5.1, step 73 RNO:

Open: ___ 1HP-24 ___ 1HP-25	1. ___ IF both BWST suction valves (1HP-24 and 1HP-25) are closed, THEN: A. ___ Start 1A LPI PUMP. B. ___ Start 1B LPI PUMP. C. Open:
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	<p>___ 1LP-15 ___ 1LP-16 ___ 1LP-9 ___ 1LP-10 ___ 1LP-6 ___ 1LP-7</p> <p>D. ___ IF two LPI Pumps are running <u>only</u> to provide HPI pump suction, THEN secure one LPI pump.</p> <p>E. ___ Dispatch an operator to open 1HP-363 (Letdown Line To LPI Pump Suction Block) (A-1-119, U1 LPI Hatch Rm, N end).</p> <p>F. ___ GO TO Step 74.</p> <p>2. ___ IF only one BWST suction valve (1HP-24 or 1HP-25) is open, THEN:</p> <p>A. ___ IF three HPI pumps are operating, THEN secure 1B HPI PUMP.</p> <p>B. ___ IF < 2 HPI pumps are operating, THEN start HPI pumps to obtain two HPI pump operation, preferably in opposite headers.</p> <p>C. ___ GO TO Step 75.</p>
74. ___ Ensure at least two HPI pumps are operating.	

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Question # 33

Unit 1 conditions:

- Quench Tank pressure is 25 psig due to 1 RC-66 (PORV) leaking by.
- Quench Tank pressure is being lowered IAW Enclosure 4.5 (Lower QT Pressure) of OP/1/A/1104/17, Quench Tank Operations.
- 1GWD-12 (QUENCH TANK VENT INSIDE RB) is OPEN.
- The Gaseous Waste Disposal system is operating normally IAW OP/1/A/1104/18.
- Vent Header Pressure is +3 inches.
- GWD-1 is in AUTO.

Which ONE of the following completes the statements:

- 1) IAW 1/A/1104/17, 1GWD-13 (QUENCH TANK VENT INSIDE RB) should be ___(1)___ to lower Quench Tank pressure, and
- 2) GWD-1 is initially expected to ___(2)___ in response to the current vent header pressure.

- A 1) cycled open and closed
 2) open
- B 1) cycled open and closed
 2) close
- C 1) opened continuously
 2) open
- D 1) opened continuously
 2) close

Proposed Answer: B

K/A Match Analysis

The conditions given in the stem indicate that the vent header is over pressurized. As a result of vent header pressure exceeding +2 inches, GWD-13 is required to be cycled so as to not over pressurize the vent header. Additionally, since Vent Header Pressure is +3 inches, GWD-1 will respond to a pressure that's above its normal setpoint (the vent header pressure is normally set slightly negative).

Answer Choice Analysis

- A. INCORRECT. First part is correct. Second part is plausible if the applicant does not understand how GWD-1 controls at setpoint.

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Supporting References (excerpted from lesson plans)

From PNS-PZR, Rev 21, page 24:

A. Relief Valve Effluent

1. Effluent from the PORV and code safety valves discharges into the quench tank which condenses and collects the relief valve effluent. The tank is protected against overpressure by a rupture disc sized for the total combined relief capacity of the two codes and the PORV.
 - a) The tank contents can be cooled by the component drain pump and quench tank cooler of the Coolant Storage System. The tank fluid is circulated from the tank through the cooler and returned to the tank by spraying into the tank vapor space.

The quench tank can be remotely vented to the Gaseous Waste Disposal System.

Excess water can be transferred to the bleed holdup tanks or the LDST. Quench Tank pressure, level, and/or temperature can be used as a diverse indication of an open relief valve, as well as leakage past the seat.

From OC-OP-WE-GWD, Rev 13a, Page 13-14:

F. GWD Vent Header Pressure Control Valve (GWD-1)

1. Allows bleed-back (recirc) of gas from in-service GWD tank back to vent header to maintain operator set vent header pressure
 - Necessary, since GWD compressor has excess capacity over what is required to maintain vent header negative during normal operation.
2. Controlled by Bailey Station in Control Room
3. Usually in AUTO, and vent header pressure is maintained slightly negative.
 - a) If the setpoint too low – air will be drawn into vent header, wasting gas tank space, and minimizing holdup time.
 - b) If setpoint too high – gas will be allowed to escape into Aux. Bldg., NOT ALARA.
 - c) Correct setting is best learned by experience.
4. All the tanks on the primary side are physically tied together through a common vent header. Any pressure changes that happen to one tank are felt/transferred to the other tanks tied to the vent header. When one tank is drained the low pressure created in that tank will transfer gas from to that tank from tanks on the header with relatively higher gas space pressure. The close coupling of these tanks makes an affect on one tank felt by all. To manage the header that couples them together it is critical to keep this in mind.
5. The operating GWD compressor(s) maintains a negative pressure (within its capacity) on the vent header. As the pressure on the vent header goes more negative below the 'Bailey' setpoint GWD-1 opens to allow pressure from the in service gas tank back into the vent header to control pressure.
 - a) Allowing bleed-back (recirc) of gas from in-service GWD tank back to vent header permits the system to maintain a controlled pressure in the vent header.
 - This bleed back pressure is necessary, since GWD compressor has excess capacity over what is required to maintain vent header negative during normal operation.

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6. In AUTO, the Bailey setpoint adjusted to maintain the vent header pressure slightly negative (~0").
 - a) If the setpoint is set to low – air will be drawn into the vent header, causing the gas tank to be filled with clean air and wasting the available tank space, and minimizing holdup time prior to release.
 - b) By observing the initial demand for GWD-1 before an evolution begins you can determine the amount of reserve capability that you have in the vent header to deal with additional gas entering it. This allows the crew the ability prior to an evolution such as venting the QT to determine if an additional compressor is necessary prior to initiating the evolution
 - c) If setpoint is set to high – gas will be allowed to escape into the Aux. Bldg., NOT ALARA and cause Noble gas to be released in the work space of employees.
 - 1) Each tank that is connected to the GWD vent header has its gas space coupled with the other tanks on the GWD header. As a tank level is lowered or raised the resultant pressure change will be felt by the other tanks until GWD-1 can respond. While the level/pressure change is occurring gas is shifting in the vent header between tanks (path of lowest pressure) until GWD-1 responds. The gas compressors will ultimately provide the low pressure area to direct the gas to the in service gas tank unless the amount of gas exceeds their capacity.

From OC-OP-WE-GWD, Rev 13a, Page 18-19:

C.3 (Obj. R4, R15) System Operation OP/1&2 or 3/A/1104/18 “Gaseous Waste Disposal System”

A. Normal Operation

1. Vent Header Split (Isolation Valves Closed)
2. For each system:
 - a) one compressor on
 - b) one tank in service
 - c) GWD-1 in auto and set slightly negative (as indicated by AB-3 gauge and in service tank stable or slowly increasing)
 - d) Compressor removes gases from vent header, discharging to tank.
 - e) GWD-1 allows “recirc” to vent header of enough gas of the in service gas tank to maintain operator set vent header pressure.
 - 1) As pressure in the vent header increases above setpoint, the controller for vent header pressure will cause GWD-1 valve position to be decreased. With less gas being recircled to the vent header, then the vent header pressure should decrease and return header pressure to setpoint. The closing down of GWD -1 will cause gas to be retained in the inservice GWD tank and the GWD tank pressure will increase.

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- 2) As pressure in the Vent header decreases below setpoint, the controller for vent header pressure will cause GWD-1 valve position to open up. With more gas being recircled to the vent header, then the vent header pressure should increase and return header pressure to setpoint. The opening up of GWD -1 will cause gas to be released from the inservice gas tank, returning the gas to the vent header, and the gas tank pressure will decrease.

From OP/1/A/1104/17 (Quench Tank Operation), Rev 47, Limits and Precautions

- 2.1 Use of this procedure can affect reactivity management due to the following: (R.M.)
- Changes in RC BHUT Boron concentration
 - Changes in LPI/RCS Boron concentration
- 2.2 Maximum Quench Tank pressure is 49 psig.
- 2.3 Maximum Quench Tank temperature is 180°F.
- 2.4 Quench Tank level shall be maintained within the band of 80 - 100 inches when RCS pressure > 45 psig.

From OP/1/A/1104/17 (Quench Tank Operation), Rev 47, Enclosure 4.4, Lower QT Pressure

1. Initial Conditions
- 1.1 IF required, lower Quench Tank temperature per Enclosure 4.3 (QT Recirc).
- 1.2 Review Limits and Precautions.
2. Procedure
- 2.1 Open 1GWD-12 (QUENCH TANK VENT INSIDE RB).
- NOTE:
- Cycling 1GWD-13 will prevent exceeding Vent Header Pressure limit.
 - Do NOT exceed +2 inches Vent Header Pressure.
- 2.2 Cycle 1GWD-13 (QUENCH TANK VENT OUTSIDE RB) as required to lower QT pressure.
- 2.3 WHEN Quench Tank at desired pressure, perform the following:
- Ensure closed 1GWD-12 (QUENCH TANK VENT INSIDE RB).
 - Ensure closed 1GWD-13 (QUENCH TANK VENT OUTSIDE RB).

From OP/3/A/1104/18 (GWD Tank System), Rev 78, Limits and Precautions:

- 2.1 The quantity of radioactivity contained in each GWD Tank shall be $\leq 3.8E5$ curies noble gases (considered as Xe-133).
- 2.2 An isolated GWD tank must be sampled for hydrogen within 24 hours of isolation.
- 2.3 Maximum hydrogen concentration in GWD system is 3% (SLC 16.11.14 "Explosive Gas Mixture").
- 2.4 If GWD tank hydrogen concentration > 3% but $\leq 4\%$ volume, concentration must be reduced to $\leq 3\%$ within 48 hours (SLC 16.11.14 "Explosive Gas Mixture").
- 2.5 If GWD tank hydrogen concentration > 4%, all additions to the tank must be suspended and concentration reduced to $\leq 3\%$ within 24 hours (SLC 16.11.14 "Explosive Gas Mixture").
- 2.6 GWD Tank pressure shall be ≤ 85 psig.

References Provided to Applicant

- None

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Question 34

Proposed Question:

Unit 1 Conditions:

- OP/1/A/1108/008, Component Cooling System, Enclosure 4.6, CC System Temperature Check is in progress.
- 1HP-7 (Letdown Control) is being throttled open.

In accordance with OP/1/A/1108/008 Enclosure 4.6, which ONE of the following describes the temperature limit on the CC System and the location to monitor CC system temperature?

- 1) CC System Letdown Cooler outlet temperature is controlled less than ___(1)___ deg-F, and
 - 2) CC System Letdown Cooler outlet temperature is monitored by ___(2)___
- A 1. 225
 2. OAC indication ONLY
- B 1. 225
 2. OAC indication AND Control Room temperature gage
- C 1. 140
 2. OAC indication ONLY
- D 1. 140
 2. OAC indication AND Control Room temperature gage

Proposed Answer: A

K/A Match Analysis

The question requires the applicant to recognize that the CC system temperature, as monitored by the OAC indication of the CC System Letdown Cooler Outlet temperature, should be kept below 225F.

Answer Choice Analysis

- A. CORRECT.
- B. INCORRECT. First part is correct. Second part is plausible because other CC system parameters (surge tank level, CC flow) have both an OAC indication and a gage in the control room.
- C. INCORRECT. First part is plausible because CC is required to be initiated prior to starting letdown with RCS temperature >120F in order to keep the anion resin from exceeding 140F and the cation resin from exceeding 250F (to prevent degradation). Applicant could plausibly think that the temperature limit for the CC system during the temperature check is similarly restricted. Second part is correct.
- D. INCORRECT. First part is plausible because CC is required to be initiated prior to starting letdown with RCS temperature >120F in order to keep the anion resin from

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exceeding 140F and the cation resin from exceeding 250F (to prevent degradation). Applicant could plausibly think that the temperature limit for the CC system during the temperature check is similarly restricted. Second part is plausible because other CC system parameters (surge tank level, CC flow) have both an OAC indication and a gage in the control room.

Technical Reference(s): Lesson Plan OC-OP-PNS-CC, Rev 14a, page 8, 11-12
(Attach if not previously provided) OP/1/A/1108/008, Component Cooling System, Rev 068,
(including version/revision number) Enclosure 4.6, CC System Temperature Check

Proposed references to be provided to applicants during examination: None

Learning Objective: Lesson Plan OC-OP-PNS-CC Obj. R2
Lesson Plan OC-OP-PNS-CC Obj. R5

Question Source: Bank # _____
Modified Bank # 2010-302 Q34 (Note changes or attach parent)
New _____

MODIFIED from 2010-301 Exam (changed the conditions to ask how CC system temperature is controlled during CC system temperature check. Kept part of question regarding where CC system temperature is monitored.)

Question History: Last NRC Exam: 2010-302

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

10 CFR Part 55 Content: 55.41 X
55.43 —

Comments:

008A1.02 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CCWS controls including: CCW temperature

2.9/3.1

Author: Amanda Toth

Supporting References (excerpted from lesson plans)

OP-OC-PNS-CC, Rev 14a, pages 8:

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- C.4 **(OBJ. R2)** Four components cooled by CC and reasons why various CC System components require cooling.
- A. CRD stators generate heat when energized. Even when the CRD stators are **NOT** energized, CC needs to be flowing through the CRD stators if the RCS temperature is $> 190^{\circ}\text{F}$ to prevent damage to the stator windings
 - B. RCP seal coolers cool RCS to minimize the temperature of the water to RCP seals following a loss of seal injection to the RCPs.
 - C. Quench tank coolers provide capability of cooling contents of quench tank after the pressurizer relief valve(s) have lifted.
 - D. Letdown coolers provide cooling of RCS letdown when the RCS is greater than 120°F to prevent damage to the purification demineralizer resins and cavitation of the HPI Pumps.
 - 1. Ion exchange resins are very sensitive to elevated temperatures.
 - a) The anion resin begins to decompose slowly at about 140°F thereby releasing a weak base compound similar to ammonia.
 - b) The cation resins do not begin to break down until temperatures exceed 250°F at which time the resin begins to release sulfuric acid.
 - 2. As the temperature of the water passing through the demineralizer increases, the affinity of the resin to attach boron atoms will decrease.
 - a) If the demineralizer is already boron saturated and temperature begins to increase, the demineralizer will actually release boron into the water. Therefore changing letdown temperature becomes a reactivity management concern.

OP-OC-PNS-CC, Rev 14a, pages 11-12:

- A. Component Coolers
 - 1. Purpose
 - a) The purpose of the component coolers is to transfer heat from the CC system to the LPSW system. Each component cooler is designed to remove the total CC system heat load for a unit.
 - 1) Both Letdown coolers are valved in with CC flow throttled to the flow rate. If one letdown cooler is isolated, the flow will not exceed 400 gpm in the on-service cooler. The flow to both coolers is ≈ 200 gpm each.
 - 2. **(OBJ. R5)** Description
 - a) The component coolers are single pass tube and shell heat exchangers. CC water flows through the carbon steel shell and LPSW flows through the alloy tubes.
 - b) The inlet of the component coolers is connected to the reactor building return header and the outlet of the component coolers is connected to the suction of the CC pumps. Manually operated isolation valves are provided on both the tube and shell sides of the coolers for isolation and maintenance.

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- c) A vent, drain, and relief valve are provided on the shell side of each cooler. A vent and a drain are also provided on the tube side of each cooler. Shell side vent and drain effluent is routed to the component cooling drain tank. Relief valve effluent is routed to the low activity waste tank.
- d) A temperature transmitter measures cooler CC outlet temperature and provides a signal to the cooler LPSW control valve to control LPSW flow through the coolers.

3. Specifications

a) Shell Side

Design Pressure	150 psig
Design Temperature	225°F
Inlet Temperature	153°F
Outlet Temperature	100°F
Shell Material	Carbon Steel

b) Tube Side

Design Pressure	150 psig
Design Temperature	300°F
Inlet Temperature	75°F
Outlet Temperature	106°F
Tube/Tube Sheet Material	Alloy

OP/1/A/1108/008, Component Cooling System, Rev 068, Enclosure 4.6, CC System Temperature Check:

- **NOTE:**

- This Enclosure used to identify operating limits of CC System
- Decreasing letdown flow will affect core reactivity by decreasing letdown temperature which causes purification IX to initially remove boron. (R.M.)
- Increasing letdown flow will affect core reactivity by increasing letdown temperature which causes purification IX to initially release boron. (R.M.)
- **NOTE:** CC System Letdown Cooler outlet temperature should **NOT** exceed 225°F while performing this temperature check. {2}
- Step 2.9 Increase letdown flow by throttling 1HP-7 (LETDOWN CONTROL) to ≈ 80 gpm.
 - Continue to monitor OAC CC Letdown Cooler outlet temperature. {2}

Parent Question from NRC Exam 2010-302 (Q34)

Which ONE of the following:

- 1) would result in an increase in CC Cooler outlet temperature?
and
 - 2) describes how the increasing CC Cooler Outlet temperature can be monitored from the Unit 1 Control Room?
- A. Throttling open 1HP-7 (Letdown Control) / OAC indication ONLY
- B. Throttling open 1HP-7 (Letdown Control) / OAC AND temperature gage in Control Room
- C. Placing 1HP-14 (LDST Bypass) in "BLEED" / OAC indication ONLY

- D. Placing 1HP-14 (LDST Bypass) in “BLEED” / OAC AND temperature gage in Control Room

Answer A Discussion

Correct. Throttling HP-7 open will increase letdown flow. CC cools the letdown coolers therefore increased letdown flow will result in an increase in CC cooler outlet temperatures. The only CC temperature indications available in the control room are those on the OAC.

Answer B Discussion

Incorrect. First part is correct. Second part is plausible since it would be correct if asking about other CC parameters, (CC flow, CRD pump pressure, CC tank level, etc.) however CC Cooler outlet temp is only available on the OAC computer.

Answer C Discussion

Incorrect. First part is plausible since CC is cooling letdown and anything that affects letdown flow would impact CC cooler outlet temp. Placing 1HP-14 to bleed does not change the actual amount of letdown flow (only the flowpath) therefore it would not impact CC temperatures. The misconception that diverting letdown to a BHUT would result in an increase in letdown flow OR the misconception of where the letdown coolers were actually located in the letdown flowpath could result in making this choice. Second part is correct.

Answer D Discussion

Incorrect. First part is plausible since CC is cooling letdown and anything that affects letdown flow would impact CC cooler outlet temp. Placing 1HP-14 to bleed does not change the actual amount of letdown flow (only the flowpath) therefore it would not impact CC temperatures. The misconception that diverting letdown to a BHUT would result in an increase in letdown flow OR the misconception of where the letdown coolers were actually located in the letdown flowpath could result in making this choice. Second part is plausible since it would be correct if asking about other CC parameters, (CC flow, CRD pump pressure, CC tank level, etc.).

Cognitive Level

Comprehension

Job Level

RO

QuestionType

NEW

Development References Student

Obj. PNS-CC R8

PNS-CC

Basis for meeting the KA

Requires ability to predict the impact of HPI system operations on CC temperatures and the ability to monitor CC temperatures during the HPI system operations to ensure the design temperatures of the CC coolers are not exceeded.

SYS008

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A1.02

KA_desc

Ability to predict and/or monitor changes in parameters to prevent exceeding design limits) associated with operating the

CCWS controls including : (CFR: 41.5 / 45.5) CCW temperature

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Question # 35

Unit 1 Initial Conditions:

- '1B' CC Pump is in ON and is running.
- '1A' CC Pump is in AUTO and is off.
- The air supply to 1CC-8 completely severs.

Unit 1 Current Conditions:

- Operators have manually re-opened 1CC-8.

Which ONE of the following completes the following statements concerning the CC pumps?

- 1) '1B' CC Pump ___(1)___ when the air supply to 1CC-8 severs.
- 2) '1A' CC Pump ___(2)___ automatically start when 1CC-8 is manually opened.

- A 1) trips
 2) will
- B 1) trips
 2) will NOT
- C 1) remains running
 2) will
- D 1) remains running
 2) will NOT

Proposed Answer: A

K/A Match Analysis

Requires the applicant to determine when CC pumps automatically trip and restart.

Answer Choice Analysis

- A. CORRECT. All running CC pumps trip when 1CC-8 closes. Since both pumps receive a trip signal, a low-flow condition exists in the CC system. When 1CC-8 reopens, both pumps will start.
- B. INCORRECT. First part is correct. Second part is plausible because 1B CC Pump is in AUTO rather than ON prior to the loss of instrument air.
- C. INCORRECT. First part is plausible because the applicant is required to know that the pump trip signal comes from the 1CC-8 valve position. If the applicant thinks that the pump trip signal is actually generated by ES channel 5 and 6 (which causes automatic closure of 1CC-8) they would not recognize that the pump would have tripped. Second part is correct.
- D. INCORRECT. First part is plausible because the applicant is required to know that the pump trip signal comes from the 1CC-8 valve position. If the applicant thinks that the pump trip signal is actually generated by ES channel 5 and 6 (which causes automatic

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B. Controls

1. (OBJ. R8) CC Pump Control Switches

- a) A three position OFF-AUTO-ON control switch is provided for each pump.
- b) A CC pump can be manually started by placing its control switch in the ON position.
- c) When a pump's control switch is in AUTO, the pump will automatically start if either:
 - 1) Component cooling header flow decreases below 575 gpm, or
 - 2) A sustained loss of power of 20 seconds or longer occurs on both main feeder buses (MFBMP).
 - 3) Once started, a CC pump will continue to run until the operator places its control switch in the OFF position or containment isolation valve CC-7 or 8 closes.
2. CC-1/HP-1 and CC-2/HP-2, Letdown Cooler Inlet Valve Open/Close Switches
3. CC-7 and CC-8, Component Cooling Return Penetration Block Valve, Open/Close Switches
4. CC-3, 4, 5 and 6, RCP Seal Cooler Outlet Valve, Open/Close Switches

OC-OP-PNS-CC, Rev 14a, page 18:

C. (OBJ. R13) Reopening CC-8 Manually Due to a Loss of Instrument Air to the Valve

1. CC-8 must be re-opened as soon as possible if it shuts during power operation.
 - a) CRD stator temperature will exceed 180°F within 4 minutes of a loss of CC flow.
 - b) The reactor must be manually tripped if two or more individual CRD stator temperatures exceed 180°F.
 - c) Also, a loss of cooling will result in HP-5 automatically closing at 135°F letdown temperature.
 - d) CC pumps will trip when CC-8 closes, and will automatically restart after it has been re-opened.
2. (OBJ. R14) CC-8 is reopened manually after it closes by:
 - a) Placing the selector lever in the MANUAL position and then rotating the handwheel in the open direction (counterclockwise). The lever does not have to be held in the manual position while operating the valve.
 - b) If containment integrity is required, the operator must stay with the valve while it is open in manual, and return the lever to AUTO once the situation has been corrected. This returns the valve to automatic. Otherwise, the valve will be inoperable remotely.

OC-OP-PNS-CC, Rev 14a, page 18:

C.5 (OBJ. R15) Interlocks Associated With the CC System

- A. If in AUTO, the standby CC Pump starts at 575 GPM flow.

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- B. If de-energized, the CRDs cannot be energized if CC flow is less than 138 GPM to the CRDs.
- C. A reactor coolant pump cannot be started if CC flow is less than 575 GPM. Low CC flow will not affect a running RCP.
- D. Letdown cooler CC inlet valve CC-1 (CC-2) must be open before letdown cooler inlet valve HP-1 (HP-2) will open.
- E. CC-7 and 8 close on actuation of ES Channels 5 and 6 (respectively)
- F. If CC-7 or CC-8 goes closed, the CC pumps will trip and automatically restart when CC-7 and CC-8 are reopened.
- G. Upon receiving MFBMP activation (both MFB's de-energized for ≥ 20 seconds), both CC pumps will receive a start signal.

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Question # 36

Which ONE of the following, concerning Unit 1 Group C Pressurizer heaters, completes the statement:

Unit 1 Group C Pressurizer heaters are powered from MCC ___(1)___, and ___(2)___ be controlled from the Unit 1 Aux Shutdown Panel.

- A 1) 1XSF
 2) can
- B 1) PXSF
 2) can
- C 1) 1XSF
 2) can NOT
- D 1) PXSF
 2) can NOT

Proposed Answer: D

K/A Match Analysis

Requires the applicant to know the bus power supply for the Unit 1 Group C PZR heaters and where the Unit 1 Group C PZR heaters can be controlled from.

Answer Choice Analysis

- A. INCORRECT. First part is plausible because this is the normal supply for the Group B PZR heaters (which is also operated from the SSF). Second part is plausible because Group B can be controlled from both the SSF and the Aux SD Panel, whereas Group C can be controlled from the Aux SD Panel only.
- B. INCORRECT. First part is correct. Second part is plausible because Group B can be controlled from both the SSF and the Aux SD Panel, whereas Group C can be controlled from the Aux SD Panel only.
- C. INCORRECT. First part is plausible because this is the normal supply for the Group B PZR heaters (which is also operated from the SSF). Second part is correct.
- D. CORRECT.

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Technical Reference(s): Lesson Plan OP-OC-PNS-PZR, Rev 21, Page 18-19
(Attach if not previously provided) _____
(including version/revision number) _____

Proposed references to be provided to applicants during examination: None

Learning Objective: Lesson Plan OP-OC-PNS-PZR Obj. R7

Question Source: Bank # _____
Modified Bank # 2010-301 Q57
New _____

MODIFIED from 2010-301 Q57 (Changed which heater group the question is about and added a part about where Group C can be controlled from.) Original Question is at the end for reference.

Question History: Last NRC Exam: 2006-301

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

10 CFR Part 55 Content: 55.41 X
55.43 _____

Comments:

010K2.01 Knowledge of bus power supplies to the following: PZR heaters
3.0/3.4

Author: Amanda Toth

Supporting References (excerpted from lesson plans)

OP-OC-PNS-PZR, Rev 21, Page 18-19:

(OBJ.R7) PZR Heaters

Heaters:

- a) replace heat lost during normal steady state operation
- b) raise the pressure to normal operating pressure during Reactor Coolant System heatup from the cooled down condition
- c) restore system pressure following transients.

Eleven Groups (A thru K) of electric heaters, divided into four banks (BANK 1, 2, 3, & 4), are assembled into three removable horizontal heater assemblies.

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ICS median selected narrow range (NR) RCS pressure signal controls these four banks of heaters.

Bank 1- (Group A; 126 kW) heaters utilizes SCR proportional control and will normally operate at an adjustable voltage capacity to replace heat lost, thus maintaining pressure at set point.

- a) Spray valve leakage, bypass flow, and insulation losses effect actual heating requirements
- b) Group K, part of Bank 2, stays on continuously (for bypass flow) and is generally associated with Bank 1 controls (ON/OFF Sw).

Banks 2, 3, and 4 – use Auto/On/Off control; Auto – cycles at setpoints

A recent analysis has determined that a minimum of ~400 kilowatts of heaters should be available from an emergency power source within two hours after loss of off-site power (LOOP) in order to establish and maintain natural circulation in MODE 3.

Heater Bank 2 – Groups B and D-can be controlled from each unit's Aux Shutdown Panel

OFF / NORMAL / ON – switch positions are selected; in NORMAL control can only be from the Control Room

Heater Bank 2- Group B and C- can be controlled from the SSF.

Group B is normally powered from MCC 1, 2, 3XSF, which is normally fed from load center 1, 2, 3X8.

However, when Group B is being powered from the SSF Diesel, it can only be operated from the SSF Unit Control Board.

Group C heaters are only powered from the SSF and can only be operated from the SSF Unit Control Board.

- o **Power is supplied from OTS1 thru PXSF transformer to PXSF MCC (located in the D/G room)**
- o Separate breaker for each unit's Group C heaters on PXSF MCC
- o Credit is NOT taken for Group C to satisfy TS 3.4.9.
- o Separate ON-OFF switches exist for both the Group C and Group B heaters for each unit in the SSF control room.

Low level heater cutoff -85" SSF PZR level-uncompensated

The heaters for each unit are normally supplied from non-safety related motor control centers (MCCs) XH, XI, XJ, and XK. They are divided among the three 4160 volt ES buses such that the loss of one entire 4160 volt bus will not preclude the capability to supply sufficient heaters to maintain natural circulation in MODE 3.

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Parent Question from Exam 2010-301

Which ONE of the following describes the power supply for the Unit 1 Group B Pressurizer heaters?

- A. 1XH
- B. 1XI
- C. 1XJ
- D. 1XSF

Question 57

T2/G2 - CPW

011K2.02, Pressurizer Level Control System

Knowledge of bus power supplies to the following:

PZR heaters.

(3.1/3.2)

K/A MATCH ANALYSIS

Question requires knowledge of power supply to Group B Pressurizer heaters as well as the Operators ability to control Group B heaters

ANSWER CHOICE ANALYSIS

Answer: A

A. CORRECT: The normal supply for Group B is 1XSF.

B. Incorrect: Plausible since 1XH is the normal supply to group E pressurizer heaters.

C. Incorrect: Plausible since 1XI is the normal supply to Group F pressurizer heaters.

D. Incorrect: Plausible since 1XJ is the normal supply to Group D pressurizer heaters.

Technical Reference(s): **PNS-PZR**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **PNS-PZR R7**

Question Source: **NEW**

Question History: Last NRC Exam **N/A**

Question Cognitive Level: **Knowledge and Fundamentals**

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Question # 37

Unit 1 Initial Conditions:

- Reactor power = 100%
- 1B RPS Channel RCS Pressure fails low.
- 1B RPS Channel is inadvertently placed in "Shutdown Bypass".

Unit 1 Current conditions:

- 1C RPS Channel loses power.

With no additional operator actions, based on the above conditions, which ONE of the following describes the impact (if any) on Unit 1 and the RPS Trip Logic?

Unit 1 ___(1)___ trip, and the RPS trip logic is ___(2)___.

- A 1. will
 2. 2/3
- B 1. will
 2. 2/4
- C 1. will NOT
 2. 2/3
- D 1. will NOT
 2. 2/4

Proposed Answer: B

K/A Match Analysis

The question requires the applicant to know the difference between Shutdown Bypass and Manual Bypass, and to determine the effect that a failed channel that is not bypassed will have on the unit. In this case, Channel B sees a low pressure; however, the channel does not trip due to the low pressure because 2.MIN shows that the Channel B pressure is faulted. However, placing Channel B in SHUTDOWN BYPASS (vice MANUAL BYPASS, this is the "malfunction") will lower the HIGH pressure setpoint, and the 3 other channels that feed Channel B will cause Channel B to trip (3 channels being above the high pressure setpoint for B). Channel B being in Shutdown bypass keeps a 2/4 trip logic.

Separately, a loss of power to Channel C causes channel C to trip. The combination of Channel B tripping and Channel C tripping will cause a reactor trip.

Answer Choice Analysis

- A. INCORRECT. The first part is correct. The second part is plausible if the applicant thinks that the Shutdown Bypass changes the coincidence logic to 2/3. (The Manual Bypass actually changes the coincidence logic to 2/3.)
- B. CORRECT.
- C. INCORRECT. The first part is plausible if the applicant thinks Channel B will not trip, since Shutdown Bypass actually bypasses the low pressure trip and Channel B pressure is failed low. However, the 2.MAX function will actually discount the Channel B pressure

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and use the 3 remaining pressures to trip that Channel. The second part is plausible if the applicant thinks that the Shutdown Bypass changes the coincidence logic to 2/3. (The Manual Bypass actually changes the coincidence logic to 2/3.)

- D. INCORRECT. The first part is plausible if the applicant thinks Channel B will not trip, since Shutdown Bypass actually bypasses the low pressure trip and Channel B pressure is failed low. However, the 2.MAX function will actually discount the Channel B pressure and use the 3 remaining pressures to trip that Channel. The second part is correct.

Technical Reference(s): Lesson Plan OC-OP-IC-RPS, Rev 18b,
(Attach if not previously provided) pages 12-13, 18-19, 53-54
(including version/revision number) _____

Proposed references to be provided to applicants during examination: None

Learning Objective: Lesson Plan OC-OP-IC-RPS Obj. R4
Lesson Plan OC-OP-IC-RPS Obj. R5
Lesson Plan OC-OP-IC-RPS Obj. R7
Lesson Plan OC-OP-IC-RPS Obj. R16

Question Source: Bank # _____
Modified Bank # 2010-301 Q37
New _____

MODIFIED from 2010-301 Q37 (Changed conditions of the stem such that the failed channel is placed in Shutdown Bypass and a subsequent channel loses power. Also made question applicable to digital RPS.) Original Question is at the end for reference.

Question History: Last NRC Exam: 2010-301

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

10 CFR Part 55 Content: 55.41 X
55.43 _

Comments:

012K6.04 Knowledge of the effect of a loss or malfunction of the following will have on the RPS: Bypass-block circuits.

3.3/3.6

Author: Amanda Toth

Supporting References (excerpted from lesson plans)

OC-OP-IC-RPS, Rev 18b, page 12-13 –

Each RPS channel powers four Rx Trip Relays (RTR's) associated with that channel (physically located one per cabinet in RPS A, B, C, and D).

Each RPS channel contains four physically separated relays, each powered from and actuated by the logic of a different RPS channel. When a particular RPS channel determines that a trip condition has been reached, a trip output is generated by that channel and the four trip relays associated with that channel are de-energized. For example, if channel A senses a trip condition, relays AA, BA, CA, and DA will all be de-energized.

RPS Ch A cabinet includes relays AA, AB, AC, and AD. AB relay coil is powered by an RPS Ch B binary output, AC coil powered by an RPS Ch C binary output, etc. Each channel's associated relay and wiring is physically separated from the other channels.

The output contacts of the four RTR's are wired to provide two-out-of-four coincidence logic to de-energize the under-voltage trip coils and energize the shunt trip coils in order to trip the CRD breaker associated with each RPS channel. Each breaker undervoltage coil circuit is monitored by a shunt trip relay as a back-up RPS trip. If the undervoltage coil power is removed due to either an RPS automatic or manually initiated trip, the shunt relay will cause the shunt trip coil to be energized and trip the breaker.

RPS Channel A provides "two-out-of-four trip" relay logic to CRD breaker A, RPS B provides "two-out-of-four trip" relay logic to CRD breaker B, etc.

The RTR circuitry is a "de-energize to trip" fail safe design. For loss of power and fail low binary output failures, the affected channel(s) RTR's fail to the tripped condition.

If two or more channels indicate a valid trip condition, all four CRD breakers will trip.

The CRD breakers operate in a one-out-of-two-taken twice configuration to remove power to the CRD, thus tripping the reactor.

OC-OP-IC-RPS, Rev 18b, page 18 –

General Description

(OBJ R2) Two different methods are available to provide protection to the RCS, fuel and fuel cladding:

Automatic system actuation (Digital RPS) to trip the CRD breakers and therefore, "trip the reactor".

Manual system actuation (MANUAL TRIP Pushbutton) to trip the CRD breakers and therefore, "trip the reactor".

RPS consists of four identical protective channels designated as RPS Channel A, B, C, and D.

(Obj. R16) Central to each RPS Channel is the Reactor Trip Component (RTC) for the TXS system.

If any two RPS channels trip, the RTC logic in each channel actuates to remove 120 VAC power from its associated CRD trip devices.

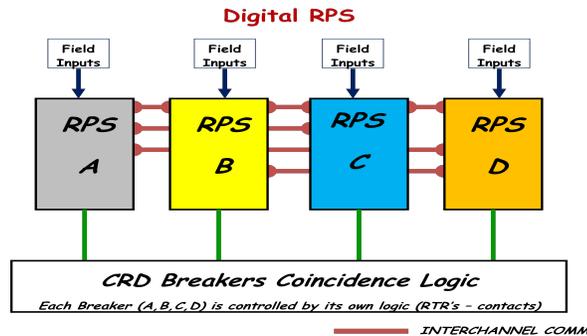
The RTC is made up of two digital output modules (451's) and four RTR's, all contained within the respective RPS channel's cabinet.

Each RTC receives a channel trip signal from that respective channel and channel trip signals from the digital output modules in the other three RPS channels.

TXS (Digital) RPS Common Logic Features:

RPS channels exchange process variables / signal inputs via fiber-optic data links.

Enables each protective channel to perform validation checks, on-line signal monitoring & signal selection when processing the RPS functions.



(Obj. R7) Analog input - 2.MIN and 2.MAX

Because the channels share input information, each channel can utilize a second minimum (2.MIN) or second maximum (2.MAX) for analog signal selection.

Digital RPS typically uses 2.MIN or 2.MAX Function Blocks for analog process input signal selection and signal validation.

For 2.MIN signal selection, each channel uses the second lowest measurement to compare with the low set point value and then determines the trip status of that channel for a "low trip" parameter.

Similarly, 2.MAX uses the second highest measurement to compare with the high set point value and then determines the trip status of that channel for a "high trip" parameter.

The associated function will not trip on one deviated (out of tolerance) signal (uses 2.MIN OR 2.MAX signal for processing) in the process measurement and thereby minimizes inadvertent RPS channel trips.

If a signal is assigned a FAULT status, the signal will be excluded from processing in 2.MIN or 2.MAX (for tripping the channel) but will be included in the Trip Function alarm logic for that channel. A Faulted signal will also provide a Trouble statalarm and OAC alarm.

Example: ONLY RPS Channel A RC pressure goes high. Would get a functional trip (statalarm, etc.) and Channel Trouble alarms (channel check logic), but would NOT get a Channel A trip. Would require a second high pressure signal in another channel to satisfy the 2.MAX logic to trip the A Channel (would also get trips on all four channels).

Protective Functions Bypasses

(Obj. R5) Shutdown Bypass

Specific RPS protective functions for a channel can be bypassed with a Shutdown Bypass key-switch located in that channel's RPS cabinets.

SD Bypass provides capability to

Perform CRD testing prior to startup.

Reset CRD bkrs when plant is shut down (pull Group 1 to 50%).

Positioning keyswitch to BYPASS bypasses the following trips:

- Low Pressure
- Variable Low Pressure
- Flux/Flow/lmb
- RCP Power/Flux

In addition to bypassing the four trip parameters above, SD BYPASS provides protection so that the plant cannot be operated normally with portions of the RPS in SD Bypass.

Automatically inserts a high RCS pressure trip set point of ≤ 1710 psig (TS ≤ 1720 psig)

While the normal high pressure trip of ≤ 2345 psig is not electrically bypassed it is basically nonfunctional because RPS will trip before the setpoint can be reached.

Automatically inserts a high power trip set point of 4%.

While the normal high flux trip of $\leq 104.75\%$ power is not electrically bypassed it is basically nonfunctional because RPS will trip before the setpoint can be reached.

The TS setpoint of ≤ 1720 psig is selected so the plant must first be shutdown, using normal procedures, before SD Bypass can be initiated. The normal low pressure trip is 1800 psig, so the plant must first be maneuvered below the normal low pressure trip point before going to SD bypass (1710 psig is the actual setpoint for conservatism).

TS requires a setpoint of $\leq 5\%$ power when shutdown thus, preventing significant power production (natural circ flow would be available to remove up to 5% of rated power).

When the SD Bypass position is selected, a Statalarm on panel 1SA-5 indicates the specific channel is in SD Bypass.

Selecting SD Bypass at full power will result in a trip of the associated RPS channel on high RC pressure and high flux.

(Obj. R6) Manual Bypass

A Manual Bypass key switch located in each RPS channel bypasses all automatic trip functions associated with that channel.

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RPS will initiate a reactor trip if any two of the four RPS channels trip; this constitutes a two-out-of-four logic. If the automatic trip functions of one channel are bypassed, two RPS channels are still required to actuate a reactor trip, but only three channels are left available. The trip logic with one channel in Manual Bypass becomes two-out-of-three.

Only one RPS channel per unit may be legally placed in BYPASS at a time (prevents the trip logic from being reduced to below 2 out-of-3 logic).

Manual Bypass is used, normally, for testing individual RPS channels while the plant is operating (so that the likelihood of inadvertent reactor trip is reduced); but it may also be used to bypass an inoperable channel due to a component failure in that channel.

The Manual Bypass keyswitch input is routed to all trip function logic circuits AND also has a hardwired 24 VDC input to the RTRs (for that channel) independent of the TXS computers preventing a channel trip.

While an RPS channel is in MANUAL BYPASS, the process signals of that channel will NOT be processed by the other channels (sets signals to a **FAULT** status) to determine trip status.

When a channel is in MANUAL BYPASS, that channel's input will still be included in the channel check and the Trip Function alarm logic.

When Manual Bypass keyswitch is in BYPASS a Statalarm on 1SA-5 will actuate for the specific channel.

Manual Bypass is administratively controlled. There are no interlocks to prevent placing two channels of RPS on the same unit in Manual Bypass at the same time.

Only one key is available for use on each unit. Units 1, 2, and 3 are keyed differently. Keys are non-removable in the BYPASS position.

Parent Question from Exam 2010-301

Unit 2 plant conditions:

- Reactor power = 100%
- 2B RPS Channel Low RCS Pressure Bistable failed in "tripped" state
- 2B RPS Channel in "Manual Bypass"

Current conditions:

- 2C RPS Channel inadvertently placed in "Shutdown Bypass"

Based on the above conditions, which ONE of the following describes the impact (if any) on reactor power and control room alarms?

With NO additional operator actions, reactor power will be _____ and the associated RPS Channel C statalarm for _____ bistable trip will be actuated.

- A. 0% / Low pressure
- B. 0% / High pressure

- C. 100% / Low pressure
- D. 100% / High pressure

T2/G1 – okm/cpw

012A4.03 Reactor Protection System

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

Channel blocks and bypasses.

(3.6/3.6)

K/A MATCH ANALYSIS

Requires the ability to monitor plant response and control room indications that occur when placing RPS Channels in Manual Bypass and Shutdown Bypass

ANSWER CHOICE ANALYSIS

Answer: D

- A. Incorrect: Both parts are incorrect. First part is plausible since there would be a bistable tripped in both the B and C channels however with the B channel in Manual Bypass the failed bistable does not result in RPS logic seeing that channel as actuated. Since it takes 2 channels to actuate, the reactor will still be at power. Second part is plausible in that it would be essentially true if the question were asking which would be bypassed instead of actuated. It would be plausible to believe that the bistables in question were tripped instead of bypassed. If the bistable is bypassed then the statalarm is essentially bypassed.
- B. Incorrect: First part is incorrect but plausible as described in A above. Second part is correct
- C. Incorrect: First part is correct. Second part is plausible in that it would be essentially true if the question were asking which would be bypassed instead of actuated. It would be plausible to believe that the bistables in question were tripped instead of bypassed.
- D. CORRECT: With the B channel in Manual Bypass the failed bistable does not result in RPS logic seeing that channel as actuated therefore there is only one RPS channel tripped. Since it takes 2 tripped RPS channels to generate a Reactor trip the Rx still be at power. When an RPS channel is placed in shutdown bypass, RPS automatically inserts a high RCS pressure trip set point of ≤ 1720 psig therefore the high RCS pressure bistable will have actuated.**

Technical Reference(s): **IC-RPS pgs 8,18,19**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **IC-RPS R5, R6**

Question Source: **NEW**

Question History: Last NRC Exam **N/A**

Question Cognitive Level: **Comprehension or Analysis**

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Question # 38

Given the following Unit 1 conditions:

Time = 1200

- Reactor trips from 100% power due a 1A MSLB.
- Core SCM = 0°F.

Time = 1206

- Tcold reaches lowest value of 398°F.
- Core SCM remains 0°F.
- 1B1 RCP is still running.

Time = 1220

- Tcold = 495°F stable.
- Core SCM = 72°F stable.
- Rule 2 (Loss of SCM) is complete.

Which ONE of the following completes the statement:

In accordance with Rule 8, Pressurized Thermal Shock, HPI can be throttled (using Rule 6) to a minimum of ___(1)___ gpm indicated, per pump, while maintaining ___(2)___ stable.

(Assume that special limits for minimum HPI flow do not apply.)

- A (1) 170
(2) CETCs
- B (1) 170
(2) Tc
- C (1) 65
(2) CETCs
- D (1) 65
(2) Tc

Proposed Answer: B

K/A Match Analysis

Question requires applicant to know the limits on throttling HPI during a PTS scenario. Rule 8 states that when Tc drops >100F in one hour, SCM must be minimized. An acceptable means of doing so is lowering HPI flow. The limits for throttling HPI are given in Rule 6 (minimum flow per pump is given as 170 gpm indicated, which takes into account instrument error, to prevent going below the real limit of 65 gpm at the pump). Since a RCP is running, HPI is throttled to control Tc in an operating loop (vice CETCs) stable for a period of 1 hour.

The assumption was stated so that applicants would not be led to choose '65 gpm' because Rule 6 allows flows less than the limits for up to 4 hours.

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Answer Choice Analysis

- A. INCORRECT. First part is correct. Second part is plausible because this would be the correct answer if no RCPs were running. If applicant does not remember RCP trip criteria (specifically, the part where <2 minutes have elapsed since SCM is lost), they might think that tripping 1B1 RCP is still acceptable. If so, no RCPs would be running, and temperature would be controlled via CETCs.
- B. CORRECT.
- C. INCORRECT. First part is plausible because the 170 gpm indicated limit is to prevent flow from reaching the real limit of 65 gpm. Second part is plausible because this would be the correct answer if no RCPs were running. If applicant does not remember RCP trip criteria (specifically, the part where <2 minutes have elapsed since SCM is lost), they might think that tripping 1B1 RCP is still acceptable. If so, no RCPs would be running, and temperature would be controlled via CETCs.
- D. INCORRECT. First part is plausible because the 170 gpm indicated limit is to prevent flow from reaching the real limit of 65 gpm. Second part is correct.

Technical Reference(s): Lesson Plan OP-OC-EAP-LOSM, Rev 18
(Attach if not previously provided) Rule 6, Rev 39, Page 1 of 1,
(including version/revision number) Rule 8, Rev 39, Page 1 of 1
Lesson Plan OP-OC-EAP-EOP, Rev 16, pages 14-15
OMP 1-18, Rev 33, Attachment 1

Proposed references to be provided to applicants during examination: None

Learning Objective: Lesson Plan OP-OC-EAP-LOSM, Rev 18 Obj. R3
Lesson Plan OP-OC-EAP-EOP, Rev 16 Obj. R27

Question Source: Bank # _____
Modified Bank # 2011-301, Q18 (Note changes or attach parent)
New _____

Original Question at the end.

Question History: Last NRC Exam: 2011-301 Q18

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

10 CFR Part 55 Content: 55.41 X
55.43 —

Comments:

013A1.10 Ability to predict and/or monitor changes in parameters (to Prevent exceeding design limits) associated with operating the ESFAS controls including: T-cold

3.4/3.7

Author: Amanda Toth

Supporting References (excerpted from lesson plans)

Rule 8, Pressurized Thermal Shock, page 1 of 1

NOTE

This rule is invoked under either of the following conditions:

- A cooldown below 400°F T_c at > 100 °F/hr has occurred.
 - HPI has injected through an open or throttled open 1HP-26, 27, 409, 410 with all RCPs OFF.
- SCM must be minimized. The following methods may be used at the discretion of the CR SRO:
 - **Throttling HPI per Rule 6 (HPI)**
 - De-energizing Pzr heaters
 - Using Pzr normal spray
 - Using Pzr aux spray
 - Using PORV
 - Throttling LPI {22}
 - **Once RCS temperature is stable, a 1-hour hold of RCS temperature must be performed unless a LOCA or SGTR is in progress. Use T_c in loop with an operating RCP or use CETCs if **NO** RCPs are operating.**

Rule 6, Page 1 of 1

HPI Pump Minimum Flow Limit

- **Maintain ≥ 170 gpm indicated/pump.** This is an instrument error adjusted value that ensures a real value of ≥ 65 gpm/pump is maintained. HPI pump flow less than minimum is allowed for up to 4 hours.

EAP-EOP, Rev 16 pages 14-15

(OBJ R27) Rules 1 – 8

Rules are written guidance for Operators to follow when certain symptoms/conditions occur.

These conditions are as follows:

Rule 1, ATWS/ Unanticipated Nuclear Power Production contingency actions.

Rule 2, Loss of SCM contingency actions.

Rule 3 Loss of Main or Emergency FDW along with contingency actions for a loss of all FDW.

Rule 4 HPI, Initiation of HPI Forced Cooling. This rule is for HPI operation

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Rule 5 Main Steam Line Break contingency actions.

Rule 6, HPI, provides HPI system operation guidance.

This Rule applies any time that HPI is in operation and the EOP is entered.
The rule contains HPI throttling criteria and minimum flow limits.

Rule 7 SG Feed Control, gives the Crew guidance/information on allowable SG feed rates and levels for different conditions.

This Rule applies anytime the SGs are being fed and EOP entered.

Rule 8 Pressurized Thermal Shock (PTS) gives the Crew guidance to protect against PTS.

Using Rules allows the guidance/information to be more in depth than the previous memory items because the Operator only needs to know which rule is applicable, refer to the rule and then apply the guidance.

Rules 6-8 do not contain check off spaces and may be implemented by the operator with procedure in hand or from memory as required (OMP 1-18). When Rule 6-8 is used by the operator, SRO concurrence to use the Rule is NOT required as it is for Rules 1-5.

OMP 1-18, Rev 33, Attachment 1 (Licensed Operator Memory Items)

1.5. If any valid subcooling margin is $< 0^{\circ}\text{F}$, perform Rule 2, *Loss Of SCM* (OATC only trips RCPs as described in 5.3.5.A.1.d).

1.9. When HPI is operating, the requirements of Rule 6, *HPI* are applicable.

1.11. Maintain the requirements of Rule 8, *PTS* as applicable.

References Provided to Applicant

- None

2011-301 Q18

Given the following Unit 1 conditions:

Time = 1200

- Reactor trips from 100% power due a 1A MSLB
- BOTH SG pressures rapidly decreasing
- Core SCM = 0°F

Time = 1204

- Tcold reaches lowest value of 416°F
- RCP 1A1 is running.

Time = 1215

- Tcold = 498°F stable
- Core SCM = 78°F stable

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- Rule 2 (Loss of SCM) is complete
- 1A SG tube leakage = 5 gpm

___ (1) ___ was the EOP tab that was entered first from Subsequent Actions.

Rule 8 (Pressurized Thermal Shock) ___ (2) ___ required to be invoked.

Which ONE of the following completes the statements above?

- A. 1. Loss of SCM, 2. is
- B. 1. Loss of SCM, 2. is NOT
- C. 1. Excessive Heat Transfer, 2. is
- D. 1. Excessive Heat Transfer, 2. is NOT

Answer A Discussion

Correct.

First part is correct. The LOSCM tab will be entered first based upon the order steps are completed in the Subsequent Actions tab. It will determine in the LOSCM tab that SCM was lost due to EHT and then the transfer to EHT tab will be made from the LOSCM tab.

Second part is Correct. Per Rule 8 if "HPI has injected through an open or throttled open 1HP-26, 27, 409, 410 with all RCPs OFF" then Rule 8 would be invoked. Rule 2 has been complete so RCP have been secured and HPI has been initiated.

Answer B Discussion

Incorrect.

First part is correct. The LOSCM tab will be entered first based upon the order steps are completed in the Subsequent Actions tab. It will determine in the LOSCM tab that SCM was lost due to EHT and then the transfer to EHT tab will be made from the LOSCM tab.

Second part is incorrect and plausible. There are two conditions, either of which require Rule 8. If all RCP's are off with HPI on is not understood then a student could conclude Rule 8 is not applicable in that a cooldown below 400 degrees at > 100 degrees per hour has not occurred.

Answer C Discussion

Incorrect.

First part is incorrect and plausible. EHT has occurred as a result of the MSLB on the 1A SG. A student could reasonable conclude EHT is applicable since it is the cause of the LOSM.

Second part is Correct. Per Rule 8 if "HPI has injected through an open or throttled open 1HP-26, 27, 409, 410 with all RCPs OFF" then Rule 8 would be evoked. Rule 2 has been complete so RCP have been secured and HPI has been initiated.

Answer D Discussion

Incorrect.

First part is incorrect and plausible. EHT has occurred as a result of the MSLB on the 1A SG. A student could reasonable conclude EHT is applicable since it is the cause of the LOSM.

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Second part is incorrect and plausible. There are two conditions, either of which require Rule 8. If all RCP's are off with HPI on is not understood then a student could conclude Rule 8 is not applicable in that a cooldown below 400 degrees at > 100 degrees per hour has not occurred.

Cognitive Level

Comprehension

Job Level

RO

QuestionType

NEW

Question Source

Development References Student References Provided

EAP-LOSCM R5

EOP Rule 8

Basis for meeting the KA

The question requires knowledge of the Subsequent Actions tab and the hierarchy of importance to address LOSCM before EHT. The student must determine that PTS limits are invoked in implementing mitigations strategies.

Basis for Hi Cog

Plant data must be evaluated to determine which EOP tab is entered first.

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Question # 39

Unit 1 Conditions at time 0500:

- Reactor power = 25%.
- 1A and 1C RBCUs operating in HIGH speed.
- 1B RBCU is operable and OFF.

Unit 1 Conditions at time 0501:

- A LOCA occurs and ES channels 1-5 actuate.
- A LOOP occurs.

Unit 1 Conditions at time 0505:

- Offsite power is restored to Unit 1.

Based on the above conditions, which ONE of the following describes RBCU status at time 0506?

1C RBCU is __ (1) __ and 1B RBCU is __ (2) __.

- A 1. operating in LOW speed
 2. operating in LOW speed
- B 1. operating in LOW speed
 2. OFF
- C 1. OFF
 2. operating in LOW speed
- D 1. OFF
 2. OFF

Proposed Answer: D

K/A Match Analysis

The question requires the applicant to know that ES-5/6 actuate on high containment pressure, and then determine the RBCU configuration as a result of the actuation, including the 3 minute time delay. The mixed speed circuit stops all running RBCUs on receipt of ES Ch-5/6 actuation, and after a 3-minute time delay, restarts them in LOW. In the case of a simultaneous LOCA and LOOP, the 3-minute time delay starts when power is restored (rather than when ES actuates).

Answer Choice Analysis

- A. INCORRECT. First part is plausible if the applicant misapplies the 3-minute time delay AND misremembers which RBCU comes restarts from ES-5. (A RBCU restarts from ES-5 and C RBCU restarts from ES-6). Second part is plausible if the applicant misapplies the 3-minute time delay and starts the clock at ES actuation rather than restoration of power.

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Supporting References (excerpted from lesson plans)

OP-OC-PNS-RBC, Rev 22b, pages 15-17:

C.6 (Obj R5, 7, 9) Engineered Safeguards (ES) Operation

A. RB Aux Fans

1. Fans receive no ES signal
2. Continue to run as initially set up
3. Forced Cooldown and HPI Cooldown sections of EP/1800/01, verifies all RB Aux Fans running. This maximizes RB cooling and minimizes the possibility of hydrogen pockets forming in RB atmosphere.

B. RBCUs

1. Receive signal from ES 5 & 6 (3 psig Reactor Building Pressure)
2. All RBCUs go to LOW speed (**Obj. R1**)
 - a) RBCUs are run in low speed due to the higher density RB atmosphere. The higher density atmosphere could cause fan motors to fail in high speed because of the increased horsepower required to move the denser air.
3. Mixed Speed Circuit - (**Obj. R1**) – Refer to Attachment 1
 - a) There is a circuit to prevent mixed speed operations of the A and C RBCUs during an actuation of ES channel 5 ONLY **OR** ES channel 6 ONLY.
 - 1) This circuit removes the control switch for the single unit of the other ES channel causing that RBCUs to immediately **STOP** regardless of the associated control switch position
 - (a) For **ES-5** – the ES-5 RBCUs and **C** RBCU also STOPS
 - (1) ES 5 RBCUs restart after 3 minutes in low speed
 - (b) For **ES-6** – the ES-6 RBCUs and **A** RBCU also STOPS
 - (1) ES 6 RBCUs restart after 3 minutes in low speed
 - 2) **IF** ES channel 5 ONLY, ALL '3' RBCUs will STOP and their switches will have no control.
 - 3) **IF** ES channel 6 ONLY, ALL '3' RBCUs will STOP and their switches will have no control.
 - 4) To regain Switch Control over ALL '3' RBCUs the following must be performed:

NOTE: When Switch Control is re-established the associated RBCU will go to the switch selected speed.

- (a) Tripped ES 5 & 6 Automatic Logic Channels Reset

OR

- (b) Tripped ES Channels 5 & 6 taken to manual

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AND

- (c) PUSH TO RET TO NORMAL AFT ES RESET Push Button on AB-3 must be depressed for each RBCU.

THEN

- (d) RBCUs will return to the operating switch control and present operating switch position called for by the EOP.
- 5) ES 5 AND/OR 6 actuated ALL '3' RBCU control switches are removed from the circuit and the RBCUs stop.
4. A mod was implemented on all three units to improve safety related bus and terminal voltages during accident situations. During a worst case event (LBLOCA and ES channels 1-8 actuate immediately) the greatest number of MOVs and other loads start during the same short period of time; this condition produces the lowest voltages throughout the safety related portion of the Electrical Distribution System. Low voltage may cause motors to stall for up to 5 seconds when initially energized.

Delaying RBCU start improves the voltage at the 600V and 208V levels for the starting loads. **The RBCU control circuits have a 3 minute start delay upon ES actuation.** On ES actuation, the RBCUs operating in High speed will auto trip and then restart in Low speed **after 3 minutes.** Any idle RBCU will start and run in Low speed **after 3 minutes.** **Any RBCU previously running in Low speed will also auto trip and restart after a 3 minute time delay in Low speed.**

If a simultaneous LOCA/LOOP occurs, the 3 minute timer will NOT begin until after power is restored. This time delay allows MOVs to complete their stroke and allows other ES equipment to start before the RBCUs start.

5. Fusible dropout plates (**"B" RBCU ONLY**) will drop if the Reactor Building air temperature heats up to a temperature between 150°F and 165°F.
- C. **(Obj R10)** Returning RBCUs to Normal Following ES Actuation (sequence)
1. Upon ES signal, RBCUs are locked in low speed. Control room switches have no effect.
 2. After the Automatic Logic Channels 5 & 6 have been reset **OR** after ES Channels 5 & 6 have been placed in "MANUAL", a seal in logic must be reset prior to regaining control of the RBCUs.
 3. To regain control of each RBCU, the operator must depress the "PUSH TO RET TO NORMAL AFT ES RESET" button located above each damper position indicator on AB3.
 4. The RBCUs will go to the position presently selected on the RBCU switch.

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2010-301 Q41

Unit 1 plant conditions:

- Time = 03:00
- Reactor power = 100%
- 1B and 1C RBCUs operating in HIGH speed
- 1A RBCU is operable and OFF
- ES channels 1-6 actuate

Based on the above conditions, which ONE of the following describes RBCU status one minute later?

- A. 1B and 1C RBCUs operating in HIGH speed and 1A RBCU OFF
- B. 1B and C RBCUs operating in LOW speed and 1A RBCU OFF
- C. ALL RBCUs operating in LOW speed
- D. ALL RBCUs will be OFF

T2/G1 - okm

022A3.01, Containment Cooling System (CCS) **Ability to monitor automatic operation of the CCS, including:** Initiation of safeguards mode of operation

(4.1/4.3)

K/A MATCH ANALYSIS

Requires the ability to monitor RBCU operation during initiation of safeguards (ES) mode of operation

ANSWER CHOICE ANALYSIS

Answer: D

A. Incorrect: Plausible if the 3-minute time delay is mis-applied. Since this is the pre-Es position of the RBCU's this would be the correct answer if you understand that the 3 minute time delay is when the RBCU's got their signal to re-position to ES position but did not understand that they were all initially stopped at the point of ES actuation.

B. Incorrect: Plausible if you were not aware of the 3 minute time delay and believed that the RBCU in OFF would not actuate on ES.

C. Incorrect: When ES actuates a 3-minute time delay is in effect and once the time delay is finished then all 3 RBCUs will start at LOW speed. This choice is plausible if you are not aware of the 3 minute time delay or believe it is less than 1 minute.

D. CORRECT: When ES actuates all operating RBCU's will stop and a 3-minute time delay is in effect. Once the time delay is finished then all 3 RBCUs will start at LOW speed. Since the 3 minute time delay has not yet timed out all RBCUs would be off.

Technical Reference(s): **PNS-RBC pg 5,6,16,17**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **PNS-RBC R1,R5**

Question Source: **Modified Bank – PNS150501- enclosed**

Question History: Last NRC Exam **N/A**

Question Cognitive Level: **Comprehension or Analysis**

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Question 40

Given the following Unit 1 conditions:

The Reactor trips from 100% power due to a LBLOCA.

Which ONE of the following completes the statement:

__ (1) __ is added to the RB Emergency Sump to allow RBS to __ (2) __.

- A (1) LiOH (Lithium Hydroxide)
(2) minimize hydrogen production from the boric acid reaction with Zircoloy
- B (1) LiOH (Lithium Hydroxide)
(2) aid in keeping Iodine in solution, ultimately reducing offsite dose
- C (1) TSP (Trisodium Phosphate Dodecahydrate)
(2) minimize hydrogen production from the boric acid reaction with Zircoloy
- D (1) TSP (Trisodium Phosphate Dodecahydrate)
(2) aid in keeping Iodine in solution, ultimately reducing offsite dose

Proposed Answer: D

K/A Match Analysis

The question requires the applicant to recall which chemical is added to the sump (the “design feature”) to entrain iodine, thereby lowering the dose in the RB and ultimately (because of some amount of leakage from the RB) reducing offsite dose.

Answer Choice Analysis

- A. INCORRECT. First part is plausible because LiOH is added to the RCS to control PH. Second part is plausible since it does inhibit H₂ production due to the boric acid reaction with Zinc and Aluminum.
- B. INCORRECT. First part is plausible because LiOH is added to the RCS to control PH. Second part is correct.
- C. INCORRECT. First part is correct. Second part is plausible since it does inhibit H₂ production due to the boric acid reaction with Zinc and Aluminum.
- D. CORRECT. TSP is loaded into baskets in the RB basement. When RBS suction is swapped to the RBES TSP will be added to the RBES water. TSP is added to aid in keeping Iodine in solution to minimize dose from iodine in the RB atmosphere.

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Technical Reference(s): Lesson Plan OP-OC-PNS-BS, Rev 15a, page 7
(Attach if not previously provided) Lesson Plan OC-CH-RC, Rev 8, page 8
(including version/revision number) _____

Proposed references to be provided to applicants during examination: None

Learning Objective: Lesson Plan OP-OC-PNS-RB Obj. R1, R5, R10

Question Source: Bank # 2012-301 Q40
Modified Bank # _____ (Note changes or attach parent)
New _____

Original question at the end.

Question History: Last NRC Exam: 2012-301

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

10 CFR Part 55 Content: 55.41 X
55.43 _____

Comments:

026K4.09 Knowledge of CSS design feature(s) and/or interlock(s) which provide for the following: Prevention of path for escape of radioactivity from containment to the outside (May need to change this one)

3.7/4.1

Author: Amanda Toth

Supporting References (excerpted from lesson plans)

OP-OC-PNS-BS, Rev 15a, Page 7

(Obj R1) Purposes of the Reactor Building Spray (RBS) System

The Reactor Bldg. Spray system has no function during normal plant operation.

When actuated by high-high Reactor Building (RB) pressure, the system provides two major functions:

Removes sensible and latent heat from the containment atmosphere.

Operation of the RBS System also serves to entrain fission product iodine (released into the RB during a LOCA) into the spray water, thereby reducing possible iodine leakage to the environment (to meet 10CFR100 criteria concerning offsite dose limits).

(Obj. R5, R10) Recirculation from the Reactor Building Emergency Sump (RBES)

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Passive Caustic Addition System {1}

The generation of hydrogen is a result of a zinc-boric acid reaction. Zinc is contained in galvanized metals in the RB and boric acid is contained in the RB spray water.

NSM ON-13104 has been implemented on all three Oconee units and is part of the Alternate Source Term Licensing Project. This modification corrected the design deficiencies and licensing vulnerabilities of the caustic addition function of the Chemical Addition System. Manual operator actions were eliminated and Control Room doses during certain accidents will be substantially reduced. The modifications also improved the reliability of the system.

The control of pH in re-circulated coolant after a LOCA is important to minimize the re-evolution of radioactive iodine isotopes that are dissolved in the coolant in the RB basement and emergency sump. Maintaining the radioiodine in solution reduces radioactive material releases to the environment.

The TSP (Trisodium Phosphate Dodecahydrate) Addition System performs this function during a LOCA. It has no function during normal operation. TSP is stored in wire mesh baskets in the reactor building basement. Following an accident, the TSP will be dissolved by the containment fluid. This will raise the pH of the water in the containment following a DBA (Design Basis Accident). The quantity of TSP stored in the baskets is sufficient to raise the containment sump fluid pH to at least 7.0 at STP following a DBA.

Following a DBA, the fission product of primary concern is radioiodine. RBS will remove elemental and particulate iodine from containment atmosphere in the injection phase from the BWST. The amount of iodine that comes out solution during sump recirculation is reduced as pH is increased. The predicted sump pH over time, which based on TSP amounts required herein, is input into a Dose Analysis model to ensure that acceptable offsite doses would be expected during a DBA.

Caustic addition has two functions:

Minimize hydrogen production from the boric acid reaction with zinc and aluminum in RB materials

Maintain Iodine in solution to minimize dose from iodine in the RB atmosphere (possible releases)

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2012-301 Q40

SYS026 K4.02 - Containment Spray System (CSS)

Knowledge of CSS design feature(s) and/or interlock(s) which provide for the following: (CFR: 41.7)

Neutralized boric acid to reduce corrosion and remove inorganic fission product iodine from steam (NAOH) in containment spray

Given the following Unit 1 conditions:

Reactor trips from 100% power due to a LBLOCA

__ (1) __ is added to the RB Emergency Sump to allow RBS to __ (2) __.

Which ONE of the following completes the statement above?

- A. 1. LiOH (Lithium Hydroxide), 2. minimize hydrogen production from the boric acid reaction with Zircoloy
- B. 1. LiOH (Lithium Hydroxide), 2. aid in keeping Iodine in solution to minimize dose from iodine in the RB atmosphere
- C. 1. TSP (Trisodium Phosphate Dodecahydrate), 2. minimize hydrogen production from the boric acid reaction with Zircoloy
- D. 1. TSP (Trisodium Phosphate Dodecahydrate), 2. aid in keeping Iodine in solution to minimize dose from iodine in the RB Atmosphere

Answer A Discussion

Incorrect. First part is plausible because LiOH is added to the RCS to control PH. Second part is plausible since it does inhibit H₂ production due to the boric acid reaction with Zinc and Aluminum.

Answer B Discussion

Incorrect. First part is plausible because LiOH is added to the RCS to control PH. Second part is correct.

Answer C Discussion

Incorrect. First part is correct. Second part is plausible since it does inhibit H₂ production due to the boric acid reaction with Zinc and Aluminum.

Answer D Discussion

Correct. TSP is loaded into baskets in the RB basement. When RBS suction is swapped to the RBES TSP will be added to the RBES water.

TSP is added to aid in keeping Iodine in solution to minimize dose from iodine in the RB atmosphere.

Cognitive Level

Memory

Job Level

RO

QuestionType

NEW

Question Source

Development References Student References Provided

PNS-BS R5, 10

CH-PC

401-9 Comments: Remarks/Status

Basis for meeting the KA

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Question requires knowledge of how and why caustic is added to the RB Spray system.

Basis for Hi Cog

Basis for SRO only

SYS026 K4.02 - Containment Spray System (CSS)

Knowledge of CSS design feature(s) and/or interlock(s) which provide for the following: (CFR: 41.7)

Neutralized boric acid to reduce corrosion and remove inorganic fission product iodine from steam (NAOH) in containment spray

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Question 41

Unit 1 Current Conditions:

- A Large Break LOCA has occurred.
- Reactor Building pressure is 10.1 psig and increasing.
- ES Channels 1-8 have actuated.
- "A BS HEADER FLOW HIGH/LOW" (1SA-8, D11) statalarm is alarming.
- "B BS HEADER FLOW HIGH/LOW" (1SA-8, D12) statalarm is NOT alarming.
- Amps on the 'A' RBS Pump are oscillating.
- Amps on the 'B' RBS Pump are normal.
- Flow for 'A' BS Header indicates 660 gpm on VB2.
- Flow for 'B' BS Header indicates 680 gpm on VB2.

Based on the above conditions, which ONE of the following describes actions required by OP/1/A/6101/008, Alarm Response Guide 1SA-08?

- 1) 'A' RBS Pump ____ (1) ____ required to be secured.
- 2) 'B' BS Header flow ____ (2) ____ required to be raised.

- A 1) is
 2) is NOT
- B 1) is
 2) is
- C 1) is NOT
 2) is NOT
- D 1) is NOT
 2) is

Proposed Answer: B

K/A Match Analysis

Requires the applicant to verify that the alarms received on 1SA-1 are consistent with the described plant conditions and to determine the required actions in accordance with the ARP.

Answer Choice Analysis

- A. INCORRECT. First part is correct. Second part is incorrect but plausible because the 'B' header statalarm is not alarming (even though it should be when flow is less than 700 gpm). Design flow for the RBS system is 600 gpm, so the applicant could think that as long as flow is above this, no action is required.
- B. CORRECT. The ARP states that if there are indications of pump cavitation the associated pump must be secured. Additionally, flow in the 'B' header should have caused D-12 to alarm. As such, the ARP states that the suction and discharge valves should be checked open and (barring an RBS piping rupture) the level in the BWST

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should be checked to evaluate swapping to Containment Sump recirc to raise flow in the header.

- C. INCORRECT. First part is incorrect but plausible if the applicant thinks that the ARP states to continue to run the pump while adjusting suction and discharge valves to clear the alarm. Increasing RB pressure lends plausibility to keeping the pump running. Second part is incorrect but plausible because the 'B' header statalarm is not alarming (even though it should be when flow is less than 700 gpm).
- D. INCORRECT. First part is incorrect but plausible if the applicant thinks that the ARP states to continue to run the pump while adjusting suction and discharge valves to clear the alarm. The second part is correct.

Technical Reference(s): Lesson Plan OP-OC-PNS-BS, Rev 15a , pages 8-10
(Attach if not previously provided) EP/1/A/6108/008 (ARG 1SA-08), Rev 28, D-11 and D-12
(including version/revision number) _____

Proposed references to be provided to applicants during examination: None

Learning Objective: Lesson Plan OP-OC-PNS-BS Obj. R2, R6, R7

Question Source: Bank # _____
Modified Bank # 2006-301 Q34 (Note changes or attach parent)
New _____

Modified the second part of the question to ask what is required to be done to 'B' Header flow. Original question at end.

Question History: Last NRC Exam: 2006-301

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

10 CFR Part 55 Content: 55.41 X
55.43 _____

Comments:

026G2.4.46 Ability to verify that the alarms are consistent with the plant conditions.
(Containment Spray)
4.2/4.2
Author: Amanda Toth

2006-301

34. 026A2.04 1

Unit 1 initial conditions:

- A Design Basis LOCA has occurred inside the RB
- RBS actuated as designed

Unit 1 current conditions:

- BS "A" HEADER FLOW HIGH/LOW has annunciated
- Operators note that amps on the 'A' RBS Pump are oscillating
- All indications on the 'B' Train RBS appear normal for the plant conditions

Which one of the following correctly describes the actions required by the ARG and the effect of those actions on the plant's design basis?

A. Allow the '1A' RBS Pump to run, throttle the discharge valve (1BS-1), and vent the pump. RBS is necessary to maintain RB pressure and temperature within their design values.

B. Allow the '1A' RBS Pump to run, throttle the discharge valve (1BS-1), and vent the pump. RBS is necessary to maintain RB pressure within its design values. No design value exists for RB temperature.

C. Secure the '1A' RBS Pump. RBS is necessary to maintain RB pressure and temperature within their design values.

D. Secure the '1A' RBS Pump. RBS is necessary to maintain RB pressure within its design values. No design value exists for RB temperature.

K/A

Containment Spray

Ability to (a) predict the impacts of the following malfunctions or operations on the CSS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Failure of spray pump.

K/A MATCH ANALYSIS

The impacts of losing the spray pump are that the plant is still within the P-T design basis limits. The procedure usage to correct, control, or mitigate is covered by the Alarm Response Guide, which states in a CAUTION that the BS Pump must be secured if showing signs of loss of suction.

SRO-ONLY ANALYSIS

Plant design basis is SRO-only knowledge.

ANSWER CHOICE ANALYSIS

A. Incorrect. Throttling is contrary to the guidance in the ARG. Plausible because throttling the discharge valve would be a method to raise suction pressure, which can be a cause of cavitation.

B. Incorrect. According to TS Basis, a design value for RB Temperature exists. Throttling is contrary to the guidance in the ARG. Plausible because throttling the discharge valve would be a method to raise suction pressure, which can be a cause of cavitation. Also plausible because ES actuation is triggered by containment pressure, not temperature.

C. Correct. CAUTION in the ARG states that RBS is to be secured if loss of suction is causing the low flow. Loss of suction is indicated by the oscillating amps in conjunction with the alarm. Tech Spec Bases states RB spray is required to keep post-accident pressure and temperature within design values.

D. Incorrect. According to TS Basis, a design value for RB Temperature exists. Plausible because ES actuation is triggered by containment pressure, not temperature.

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REFERENCES

1. OP/1/A/6101/008, Alarm Response Guide 1SA-08, Rev. 016.
2. Tech Spec 3.6.5, Reactor Building Spray and Cooling System, Basis

From OP-OC-PNS-BS, Rev 15a:

- D. State the setpoint, statalarms armed, and equipment actuated by ES Channels 7 and 8. (R6)
- E. List the following flow values for the RBS pumps. (R2, R7)
- E.1 Minimum flow requirement
 - E.2 Normal ES flow when taking suction from BWST
 - E.3 Normal flow when taking suction from RB Emergency Sump (RBES)
- A. **(Obj. R2, R7)** RBS Pumps
1. Single stage centrifugal pumps with operating characteristics as indicated by the pump characteristic curve.
 2. Pump runout occurs at 1800 gpm at the maximum assumed NPSH.
 3. Minimum flow required for the RBS pumps is 600 gpm.
 4. Normal ES flow from BWST and flow when taking suction from RBES is 700 – 1200.
 5. The pumps are started from switches on 1UB2.
 6. When ES-7 and 8 actuate, the "BS Header Flow High/Low" statalarm for each header is armed.
 - a) Low flow on a given header alarms at 700 gpm.
 - 1) The operator is directed to check RBS pump operation, BWST level, and suction and discharge valve position by the Alarm Response Guide.
 - b) High flow on a given header alarms at 1200 gpm
 - 1) The operator is referred to the EOP for guidance to throttle RBS flow to within operating limits.
 7. **(Obj. R2, R7)** Nominal RBS pump flow is 700-1200 gpm when suction to the pumps is from the BWST. This eliminates the need for throttling during ES.
 - a) As RB pressure changes occur following a LOCA, flow may have to be readjusted.
 - b) Flow Indication via Recorders is located on VB2.

OP/1/A/6101/008, Rev 28 Alarm Response Guide 1SA-08

BS "A" HEADER FLOW HIGH/LOW (D-11)

1. Alarm Setpoint
 - 1.1 High Alarm - 1200 gpm increasing flow.
 - 1.2 Low Alarm - 700 gpm decreasing flow.

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2. Automatic Action

None

3. Manual Action

3.1 High Flow:

3.1.1 Refer to EP/1/A/1800/001 (Emergency Operating Procedure) for guidance to limit RB Spray flow to within operating limits.

CAUTION: If low flow is due to loss of suction, secure RBS pump until suction is regained and the pump is vented.

3.2 Low Flow:

3.2.1 Verify pump suction and discharge valves are open and pump is in operation.

3.2.2 Check level in BWST.

A. IF required, shift pump suction to RB Emergency Sump per EP/1/A/1800/001 (Emergency Operating Procedure).

3.2.3 Check RBS piping for rupture.

A. IF piping ruptured, secure pump and isolate the break.

4. Alarm Sources and References

4.1 OEE-118-17 & 18.

4.2 1BS FT-3A RB Spray Hdr A Flow.

BS "B" HEADER FLOW HIGH/LOW (D-12)

1. Alarm Setpoint

1.1 High Alarm - 1200 gpm increasing flow.

1.2 Low Alarm - 700 gpm decreasing flow.

2. Automatic Action

None

3. Manual Action

3.1 High Flow:

3.1.1 Refer to EP/1/A/1800/001 (Emergency Operating Procedure) for guidance to limit RB Spray flow to within operating limits.

CAUTION: If low flow is due to loss of suction, secure RBS pump until suction is regained and the pump is vented.

3.2 Low Flow:

3.2.1 Verify pump suction and discharge valves are open and pump is in operation.

3.2.2 Check level in BWST.

A. IF required, shift pump suction to RB Emergency Sump per EP/1/A/1800/001 (Emergency Operating Procedure).

3.2.3 Check RBS piping for rupture.

A. IF piping ruptured, secure the pump and isolate the break.

4. Alarm Sources and References

4.1 OEE-118-17 & 18.

4.2 1BS FT-2A RB Spray HDR B Flow

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Question 42

Unit 1 Initial Conditions:

- Reactor power is at 22%.
- The Turbine is online.
- OP/1/A/1106/014, Enclosure 4.2, Startup of Moisture Separator Reheaters (Turbine Online) is in progress.
- 1HD-92/1HD-645 Moore Controller and 1HD-95/1HD-647 Moore Controller are in AUTOMATIC.
- 1HD-92 and 1HD-95 are OPEN.

Unit 1 Current Conditions:

- 1HD-92/1HD-645 and 1HD-95/1HD-647 Moore controllers have experienced a loss of power.
 - The upstream side of 1HD-92 and 1HD-95 is 5°F warmer than the downstream side.
- 1) When power is restored to 1HD-92 & 1HD-645 and 1HD-95/1HD-647, with no additional operator actions, 1HD-92 and 1HD-95 (SSRH Tank Level Control) will be operating in ___(1)___; and
 - 2) With no additional operator actions, a water hammer event ___(2)___ likely to occur.

Which ONE of the following completes the above statements?

- A 1) Automatic
 2) is NOT
- B 1) Automatic
 2) Is
- C 1) Manual
 2) is NOT
- D 1) Manual
 2) is

Proposed Answer: C

K/A Match Analysis

The Moore controllers for 1HD-92/1HD-645 and 1HD-95/1HD-647 are placed in Automatic during the startup of the MSRs. Operation in automatic helps to ensure a water hammer event will not occur as a result of the introduction of a sub-cooled liquid into a saturated vapor by

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controlling the valve positions of 1HD-92/95. 1HD-92/95 are maintained closed when the required ΔT is not met based on comparing upstream and downstream temperatures while 1HD-645/-647 cycle to control tank level and preheat the line upstream of HD-92/95. When the ΔT is met, HD-645/647 close and HD-92/95 cycle to maintain tank level. OP/1/A/1106/014 states that operation of HD-92/95 in manual can cause water hammer because temperature/level interlocks can be bypassed.

However, on a loss of power, HD-92/95 and HD-645/647 close and all drains discharge to the condenser. When power is restored, HD-92/95 remain closed and in Manual and HD-645/647 start cycling again to reheat the line. The applicant is required to know how and why these valves operate the way they do in reference to water hammer and MSR startup.

Answer Choice Analysis

- A. INCORRECT. First part is plausible because 1HD-645 and -647 (Low Point Drain to Condenser Control Valve) will be cycling automatically after power is restored. Second part is correct.
- B. INCORRECT. First part is plausible because 1HD-645 and -647 (Low Point Drain to Condenser Control Valve) will be cycling automatically after power is restored. Second part is plausible if the applicant thinks that 1HD-92/94 fail as-is. The ΔT information in the stem was added to enhance the plausibility of this option: if the applicant thinks that the valves would be in automatic, the ΔT given means they should be shut. (HD-92/95 will not open unless the upstream pipe is at least 18F warmer than the downstream pipe.)
- C. CORRECT.
- D. INCORRECT. First part is correct. Second part is plausible if the applicant thinks that 1HD-92/94 fail as-is. Additional plausibility because OP/1/A/1106/014 states that operation of HD-92/94 in manual can cause water hammer. However, the water hammer can be caused because being in manual bypasses the temperature/level interlocks, and thus HD-92/94 can be opened allowing potentially sub-cooled liquid into a steam environment. Since HD-92/94 fail closed, a water hammer event should not occur.

Technical Reference(s): Lesson Plan OP-OC-STG-MSR, Rev 13a, page18-20
(Attach if not previously provided) OP/1/A/1106/014, Rev 62, Limitations and Precautions
(including version/revision number) OP/1/A/1106/014, Rev 62, Enclosure 4.10, Valve Logic

Proposed references to be provided to applicants during examination: None

Learning Objective: Lesson Plan OP-OC-STG-MSR Obj. R11

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

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge —
Comprehension or Analysis

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Question 43

Unit 1 initial conditions:

- Reactor power = 50%
- 1A and 1B SG levels ≈30% Operating Range

Current conditions:

- 1A1 and 1B1 RCP's trip

Based on the above conditions, with NO operator actions, which ONE of the following states where SG levels will be controlled?

- A. 240 inches on the Extended Startup Range.
- B. 30 inches on the Extended Startup Range.
- C. 25 inches on the Startup Range.
- D. 50% on the Operating Range.

Proposed Answer: C

K/A Match Analysis

Question requires applicant to know that a reactor trip occurs as a result of the loss of two RCPs. Applicant must also know that at least one RCP remains (3 or 4 could be operating in the IC's). The result is a condition where main feedwater is still available and feeding the steam generators (EFW is not required). Since at least one other RCP is still operating, the SGWLCS will control FDW at 25" on the SUR.

Answer Choice Analysis

- A. INCORRECT. Would be correct if EFW actuated and no RCPs are running.
- B. INCORRECT. Would be correct if EFW actuated.
- C. CORRECT.
- D. INCORRECT. Would be correct if no RCPs running.

Technical Reference(s): Lesson Plan OP-OC-CF-FDW, Rev 17d, page 20-22

(Attach if not previously provided) Lesson Plan OP-OC-CF-EF, Rev 27a, pages 27, 49

(including version/revision number) Lesson Plan OP-OC-IC-RPS, Rev 18b, page 49

Proposed references to be provided to applicants during examination: None

Learning Objective: Lesson Plan OP-OC-CF-FDW, Obj. R28

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

Question History: Last NRC Exam: 2007-301 Question 39 (provided at end for reference)

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

10 CFR Part 55 Content: 55.41 X
55.43 —

Comments:

059K1.04 Knowledge of the physical connections and/or cause-effect relationships between the MFW and the following systems: S/GS water level control system

3.4/3.4

Author: Amanda Toth

Supporting References (excerpted from lesson plans)

OP-OC-CF-FDW, Rev 17d, page 20-22

8. **(Obj. R28)** Startup FDW Control Valves (FDW-35 and 44)
 - a) Interlock with Main FDW Block valves already discussed.
 - b) Controlled by ICS Bailey Station on UB1. MOD OD1,2,301430 (ICS) [See writeup next page].
 - c) In AUTO, and at low unit load, the SU CV's control SG levels a minimum level of 25" SUR (Startup Range) to prevent the SG's from boiling dry.
 - d) As unit load increases during startup, the SG's come off of level control, as the amount of FDW to them increases; the SU CV's begin to ramp open under command of the ICS.
9. Another level control circuit associated with the SU CV's is SG level control on a loss of all RCP's. In this condition, in order to establish natural circulation cooling through the SG's, the SU CV's, if in AUTO, will establish and maintain SG levels at 50% on the OR. ICS limits the valve demand to 40% open demand to prevent over-cooling during the fill to 50% OR level. The Operator can take manual control of the valves during the fill if necessary to prevent overcooling.
10. Startup Line Isolations (FDW-36 and 45) and SG EFDW Header Isolations (FDW-38 and 47)
 - a) Motor operated from Control Room.

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- b) CLOSE, AUTO, OPEN.
- c) Normally in AUTO.
- d) If all four RCP's trip, FDW-36 ("A" SG) and FDW-45 ("B" SG) will CLOSE to isolate SG feed via the Main Feed Ring, while FDW-38 and FDW-47 OPEN to provide flow into the SG's via the Auxiliary Feed Ring. The Aux Feed Ring sprays Feedwater into the SG at a much higher point than the Main Feed Ring, effectively raising the thermal center of the SG, helping to promote natural circulation cooling when all of the RCP's have been lost
- e) Another part of the logic associated with these four valves (FDW-36/38 & FDW-45/47) no longer serves original design purposes because of required post-TMI changes that made EFDW independent of the ICS:
 - 1) The valves are also designed to swap positions (FDW-36 & 45 close and FDW-38 & 47 open) if Main FDW is lost (both FDWP's trip).
 - 2) This design was intended to isolate the Main Feed Ring and establish a path into the SG's via the Aux Feed Ring for EFDW, using the TDEFDWP and the SU CV's to control SG level:
 - (a) At 25 inches on the Startup Range with a loss of MFDW.
 - (b) At 50% on the Operating Range with a loss of RCP's (would control at same level if MFDW is also lost).

OP-OC-CF-EF, Rev 27a, page 27, 49

- E.1 The purpose of the Emergency Feedwater System is to auto start upon loss of MFDW for decay heat removal, until the DHR System is in service.
- A. Automatic Level Control
 - 1. FDW-315 and FDW-316 maintain SG level at setpoint.
 - a) 30" / RCPs on
 - b) 240" / RCPs off
- B. Level Control System
 - 1. (OBJ. R37) Auto level control selected: Level Control System will control SG levels at:
 - a) 30" XSUR – any RCP running
 - b) 240" XSUR - loss of all RCPs (natural circulation)

OP-OC-IC-RPS, Rev 18b, page 49

- C. (Obj. R3, R4) Power Level Based on the Number of RCPs Running (Flux/Pump) Trip
 - 1. Provides redundant trip protection to maintain DNBR by tripping the reactor due to loss of RCPs (diverse signal from flux/flow/imbalance trip signal).
 - 2. This trip signal will also restrict the power level obtainable, based upon the number of operating RCPs.
 - 3. Per TS, if reactor power is above 2% Full Power, RPS will trip the reactor if two RCPs are lost. The actual setpoint is 1.5%.

Parent Question

From 2007-301, Question 39

Unit 1 initial conditions:

- Reactor power = 50%
- 1A and 1B SG levels ≈30% Operating Range

Current conditions:

- 1A1 and 1B1 RCP's trip

Based on the above conditions, which ONE of the following states where SG levels will be controlled?

Assume no operator actions

- A. 240 inches on the Extended Startup Range.
- B. 30 inches on the Extended Startup Range.
- C. 25 inches on the Startup Range.
- D. 50% on the Operating Range.

T2/G1 - gcw

059A3.02, Main Feedwater

Ability to monitor automatic operation of the MFW, including: Programmed levels of the S/G (2.9/3.1)

K/A MATCH ANALYSIS

Knowledge of plant response and automatic SG level control on a loss of 2 RCPS.

ANSWER CHOICE ANALYSIS

Answer: C

A. Incorrect. Still have at least one RCP available. MFDW also available. EFDW should not start.

B. Incorrect. EFDW is not be required.

C. Correct. MFDW is still available. SG level should be maintained at 25" on SUR via Startup Control Valves.

D. Incorrect. Will still have at least one RCP on.

Technical Reference(s): **CF-FDW Page 22**

Proposed references to be provided to applicants during examination: **None**

Learning Objective: **CF-FDW R37, R28**

Question Source: **Bank; CF033701**

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis

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Question 44

Unit 1 conditions:

- A Reactor trip has occurred.
 - Main Feedwater, Condensate Booster Pumps and Emergency Feedwater are unavailable.
 - Rule 3 is initiated.
 - Rule 4 is in progress, but HPI cooling is inadequate.
 - The crew is performing actions in the LOHT tab.
 - A SSF Event is NOT in progress.
 - All SCM > 0F.
- 1) In accordance with the LOHT tab, the preferred method of feeding the SGs is via ___(1)___; and
- 2) After heat transfer is established, the maximum flow rate is ___(2)___ in accordance with Rule 7.

- A (1) SSF-ASW
 (2) 400 gpm
- B (1) SSF-ASW
 (2) 500 gpm
- C (1) Station ASW
 (2) 400 gpm
- D (1) Station ASW
 (2) 500 gpm

Proposed Answer: B

K/A Match Analysis

Question requires applicant to know how to feed SGs when EFW is not available.

Answer Choice Analysis

- A. INCORRECT. First part is correct. Second part is plausible because this is the maximum flow rate allowed when any $SCM \leq 0F$.
- B. CORRECT.
- C. INCORRECT. First part is plausible because this is the method used when SSF-ASW is not available. Second part is plausible because this is the maximum flow rate allowed when any $SCM \leq 0F$.
- D. INCORRECT. First part is plausible because this is the method used when SSF-ASW is not available. Second part is correct.

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Technical Reference(s): Lesson Plan OP-OC-EAP-LOHT, Rev 16, pages 10-11
(Attach if not previously provided) EP/1/A/1800/001 Rule 7, Rev 39, page 3-4 (Tables 1 and 2)
(including version/revision number) Lesson Plan OP-OC-EAP-SA, Rev 16a, page 15

Proposed references to be provided to applicants during examination: None

Learning Objective: Lesson Plan OP-OC-EAP-LOHT, R19

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

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content: 55.41 X
55.43 -

Comments:

061K3.02 Knowledge of the effect that a loss or malfunction of the AFW will have on the following: S/G

4.2/4.4

Author: Amanda Toth

Supporting References (excerpted from lesson plans)

OP-OC-EAP-LOHT, Rev 16, pages 10-11 (obj R19)

(Obj. R19) Steps 7-11 apply if you have some HPI flow, but adequate HPI Forced Cooling has not been established. Additional methods to cool the core must be established including feeding the SG with SSF-ASW or Station-ASW. Steps 7-10 will align SSF-ASW or Station-ASW to feed the SGs and the RCS High Point Vents will be opened to reduce RCS pressure and maximize the amount of HPI injection into the RCS. HPI F/C is preferred over introducing raw water into the SGs; therefore, raw water is only used if HPI F/C is unavailable or inadequate.

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SSF-ASW would be available unless you know some reason it is OOS. i.e. on the turnover sheet. SSF-ASW is aligned by dispatching an operator to perform EOP Enclosure 5.34, Aligning SSF-ASW for SG Feed. This enclosure provides actions to establish SSF-ASW flow to the SGs. Flow will be throttled by the operator in the SSF at the direction of the Control Room not to exceed total SSF-ASW Flow Limits specified in the enclosure.

If SSF-ASW is not available then the RNO will dispatch an operator to perform EOP Enclosure 5.8, Feeding SGs with Station ASW. This enclosure will align and start the Station ASW Pump to feed the SGs concurrently with Control Room operators lowering SG pressure. When SG pressure is <150 psig, valves are locally opened in the Penetration Room to allow SG feeding to commence. Cooldown is controlled using steaming until the TBVs or ADVs are fully open and subsequently by directing throttling of the local flow valves. **It is not desired to establish a level in the SGs as an excessive RCS cooldown will occur.**

Because the shut off head of this pump is approximately 75 psig, the TBVs should be used to reduce steam generator pressure to approximately atmospheric pressure to ensure flow.

The use of raw cooling water should be limited since significant fouling and extensive secondary chemistry problems will develop throughout the plant. It should be used as a last option to prevent core damage.

Steps 9 and 10 open the RCS High Point Vents which will result in lower RCS pressure, which will allow greater injection flow and therefore better core cooling. RCS pressure will be governed by a combination of HPI pump discharge pressure, the PORV relief flow capacity and the decay heat level.

Once you reach step 11, you will transfer to HPI CD tab. You should have some HPI flow as well as either SSF-ASW or SASW feeding the SG's and you can now get cooldown and recovery guidance in HPI CD.

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EP/1/A/1800/001 Rule 7, Rev 39, page 3-4 (Tables 1 and 2)

Table 1 Maximum Feed Rates When All SCMs are > 0°F		
SG Condition	Flow Instrument	Maximum Feed Rate
Dry SG w/o Heat Transfer	EFDW flow indicator	100 gpm to <u>affected</u> SG
	S/U FDW flow indicator	0.05×10^6 lbm/hr to <u>affected</u> SG
	SSF ASW flow indicator	100 gpm <u>total</u> to Unit 1
Non-dry SG OR Dry SG with Heat Transfer	EFDW flow indicator	1000 gpm per header
	S/U FDW flow indicator	0.5×10^6 lbm/hr per header
	SSF ASW flow indicator	500 gpm total to Unit 1
SSF Event* occurred while in MODE 1 <u>or</u> 2 with TD EFDWP OR alternate unit providing SG feed	EFDW flow indicator OR S/U FDW flow indicator	<ul style="list-style-type: none"> • Feed as needed to maintain RCS pressure band of 1950 - 2250 psig (1600 - 2200 psig if notified by SSF of PZR Solid Ops Control Band) • WHEN RCS pressure band is established, THEN feed to establish RCS Narrow Range T_c 550 - 555°F.

Table 2 Feed Rates To Be Established When <u>Any</u> SCM is ≤ 0°F and Rapid Cooldown NOT in Progress			
NOTE			
After initial feed rates are established, flow may be throttled to control cooldown but SG levels must continue to increase until LOSCM setpoint is reached. This note does NOT apply to SSF Event.			
FDW source	Flow Instrument	Initial Feed Rates	
Emergency FDW	EFDW total flow indicator	1 SG	450 gpm
		2 SGs	300 gpm each
	S/U FDW flow indicator	1 SG	0.23×10^6 lbm/hr
		2 SGs	0.15×10^6 lbm/hr each
Main FDW	S/U FDW flow indicator	1 SG	0.33×10^6 lbm/hr
		2 SGs	0.22×10^6 lbm/hr each
SSF ASW AND NO SSF Event*	SSF ASW flow indicator	400 gpm total to Unit 1	
NOTE			
SSF events and LOCAs are NOT postulated at the same time. LOSCM during SSF event is caused by overcooling. Cooldown while on SSF RC Makeup must be controlled due to capability of the SSF RC Makeup Pump. This note only applies to SSF events.			
EFDW AND SSF Event* that occurred while in MODE 1 <u>or</u> 2	EFDW flow indicator OR S/U FDW flow indicator	<ul style="list-style-type: none"> • Feed as needed to maintain RCS pressure band of 1950 - 2250 psig (1600 - 2200 psig if notified by SSF of PZR Solid Ops Control Band) • WHEN RCS pressure band is established, THEN feed to establish RCS Narrow Range T_c 550 - 555°F. 	

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Question 45

Unit 1 initial conditions:

- Reactor power = 50%.
- Loss of main feedwater.

Current plant conditions:

- RCS temperature 546 °F decreasing.
- PZR Level 45" decreasing.
- RCS pressure 2015 psig decreasing.
- A SG pressures = 995 psig decreasing.
- B SG pressures = 1010 psig stable.

Which ONE of the following correctly completes the statements:

- 1) Assuming no operator actions, the malfunction that would result in the above conditions is ___(2)___, and
- 2) In accordance with the EHT tab of EP/1/A/1800/001, ___(2)___ SG/SGs is/are required to be isolated.

- A 1) the CSAE steam supply relief valve failing OPEN
 2) both
- B 1) the CSAE steam supply relief valve failing OPEN
 2) a single
- C 1) 1FDW-315 failing OPEN
 2) both
- D 1) 1FDW-315 failing OPEN
 2) a single

Proposed Answer: D

K/A Match Analysis

Question requires the operator to understand the effect of excessive EFW flow to A SG (1FDW 315 failing open) on RCS parameters, namely that excessive EFW flow causes a large increase in heat transferred from the RCS, resulting in a drop of PZR level and RCS pressure.

Additionally, the applicant is required to determine if one or both SG's are required to be isolated (the operational implication) per the EHT tab.

Answer Choice Analysis

- A. INCORRECT. First part is plausible because the CSAE steam supply relief would cause an overcooling condition; however, the steam supply comes from the "B" Main steam line. Second part is plausible if the applicant has the misconception that when the RCS is overcooled, both SGs are isolated (rather than only the affected SG).
- B. INCORRECT. First part is plausible because the CSAE steam supply relief would cause an overcooling condition; however, the steam supply comes from the "B" Main steam line. Second part is correct.

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- C. INCORRECT. First part is correct. Second part is plausible if the applicant has the misconception that when the RCS is overcooled, both SGs are isolated (rather than the affected SG).
- D. CORRECT.

Technical Reference(s): Lesson Plan OP-OC-EAP-EHT, Rev 18c, page 7-10, 16
(Attach if not previously provided) EP/1/A/1800/001, Rev 39, steps 3 and 4
(including version/revision number) _____

Proposed references to be provided to applicants during examination: None

Learning Objective: Lesson Plan OP-OC-EAP-EHT Obj. R3, R8

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

Question History: Last NRC Exam: 2007-301 Q40

Parent Question at end for comparison. Kept the portion of the question that asked about the failure that caused the indications, but added the part about what is required per procedure.

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

10 CFR Part 55 Content: 55.41 X
55.43 —

Comments:

061K5.01 Knowledge of the operational implications of the following concepts as they apply to the AFW: Relationship between AFW flow and RCS heat transfer

3.6/3.9

Author: Amanda Toth

Supporting References (excerpted from lesson plans)

Supporting References

OP-OC-EAP-EHT, Rev 18c, page 7-10, 16 (R3)

Excessive Heat Transfer (EHT) is entered when excessive primary to secondary heat transfer is occurring or has occurred as indicated by low SG pressure or decreasing RCS temperature and pressure.

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EHT would apply if MSLB occurs or cooldown has been excessive due to reduced SG pressure or excessive FDW / EFDW flow.

(Obj. R1) Entry into EHT can be from:

Parallel Actions pages

Subsequent Actions diagnostics steps transfer

Loss of Subcooling Margin section

(Obj. R3) Overall Mitigation Strategy (The below information is from "AREVA Generic EOP Guidelines Basis for Excessive Heat Transfer")

Terminate the cause of the overcooling transient.

Restore controlled primary to secondary heat transfer.

Stabilize RCS pressure and temperature.

Backup SG heat removal by establishing HPI cooling if required.

Concerns

Excessive primary to secondary heat transfer is always caused by a failure in the control of secondary side parameters. This failure manifests itself as a loss of steam pressure, excessive steam flow, excessive FDW flow, or perhaps a combination of both. An extended overcooling is a severe shock to the plant and requires quick and effective action by the operator to mitigate the transient. There are several concerns as follows:

Loss of Pzr level

An extended overcooling can result in a loss of Pzr level. This, in turn, causes a loss of RC pressure control. An extended overcooling can empty the surge line, which results in the RCS becoming saturated.

Saturated RCS with Extended Overcooling

A large steam line break or extended overcooling (i.e., continued FDW with small steam line break) can result in a saturated RCS. This requires treatment of the higher priority symptom, which is a loss of SCM. After treating the loss of SCM (e.g., tripping RCPs and initiating HPI), the excessive overcooling should be treated.

Possible SG Damage

A rapid overcooling could result in SG damage from tube vibration due to high steam/feed flow as well as thermal shock to the SG and from excessive SG tube tensile loads due to exceeding tube-to-shell ΔT limits. The SG tube-to-shell ΔT limits may be exceeded because the tube temperature will decrease corresponding to the SG saturation temperature while the SG shell will cool relatively slowly due to its large mass. The potential exists for an SGTR to occur from the overcooling. If the steam leak is unisolable it may be necessary to boil the SG dry.

Pressurized Thermal Shock

(Obj. R2) With an extended overcooling, thermal shock becomes a concern. PTS guidance must be invoked if the criteria are exceeded. The RCS must be controlled to ensure that PTS guidance is not violated. This requires action on the part of the

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operator to control RC pressure and temperature. PTS conditions may develop if HPI flow is not appropriately throttled during an overcooling event.

An attempt should be made to identify and isolate the cause of the overcooling transient. If it is apparent that only one SG is at fault, then all isolation actions should be performed on the affected SG only. If it is not clear which SG is causing the overcooling, then both should be isolated.

Identification of the affected SG should be possible by comparing the steam pressure, level, loop T-cold temperatures and MFDW or EFDW flow to each SG. The affected SG will be obvious if any of the parameters listed above deviate significantly in either SG from post-trip values.

(Obj.R5) Whenever an intact SG has boiled dry and feedwater is to be restored, the auxiliary nozzles (EFDW) should be used to avoid a potential thermal shock that the dry tube sheet would experience with flow introduced through the main nozzles. If the Aux FDW Ring is used (MDEFDWP or TDEFDWP), the FDW will not contact the lower tube sheet until the FDW has been heated by being sprayed into the upper regions of the SG.

EP/1/A/1800/001, Rev 39, steps 3 and 4

Place the following in HAND and decrease demand to zero on **all affected** SGs:

✓	1A SG	✓	1B SG
	1FDW-32		1FDW-41
	1FDW-35		1FDW-44

Close the following on **all affected** SGs:

✓	1A SG	✓	1B SG
	1FDW-372		1FDW-382
	1MS-17		1MS-26
	1MS-79		1MS-76
	1MS-35		1MS-36
	1MS-82		1MS-84
	1FDW-368		1FDW-369

Parent Question

2007-301, **Question 40**

Unit 1 initial conditions:

- Reactor power = 50%
- Loss of main feedwater

Current plant conditions:

- RCS temperature 546 °F decreasing

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- PZR Level 45" decreasing
- RCS pressure 2015 psig decreasing
- A SG pressures = 995 psig decreasing
- B SG pressures = 1010 psig stable

Which ONE of the following malfunctions will result in the above conditions?

ASSUME NO OPERATOR ACTIONS

Turbine Control Valve #1 failed OPEN

CSAE steam supply relief valve failed OPEN

SG SUR level indication fails LOW

1FDW-315 failed OPEN

Question 40

T2/G1-kds

061K3.01, Auxiliary / Emergency Feedwater (AFW) System

**Knowledge of the effect that a loss or malfunction of the AFW will have on the following:
RCS (4.4/4.6)**

K/A MATCH ANALYSIS

Question describes how a malfunction of the EFDW control valve (1FDW-315) will affect RCS parameters (RCS temp & pZR level).

ANSWER CHOICE ANALYSIS

Answer: D

A. Incorrect: the control valve leak would be isolated upon the Rx/Turbine Trip. RCS temperature would not still be decreasing. Plausible because RCS temperature would be decreasing if the control valve was still leaking.

B. Incorrect: CSAE steam supply relief valve failed OPEN would not cause this plant response. The steam supply comes off the "B" Main Steam line.

C. Incorrect: SUR indication failing low would cause the SU Control Valves to fully open but would not cause an overfeed condition because the MFPs have tripped. Plausible because if a MFP were still operating, it would cause an overfeed condition.

D. Correct: 1FDW 315 failing open will cause overfeeding of the 1A SG which will cause the RCS to cool to below setpoint (~ 555 °F).

Technical Reference(s):

Proposed references to be provided to applicants during examination: **None**

Learning Objective: **SAEL019 R3**

Question Source: **New**

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis

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10 CFR Part 55 Content: 55.41 X
 55.43

Comments:

062A3.05_Ability to monitor automatic operation of the ac distribution system including safety-related indicators and controls.

Question and proposed answers reformatted. Removed teaching about what the Keowee units' auxiliary supplies were.

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Question # 47

The Operators have been directed to synchronize KHU-2 to the grid in accordance with OP/0/A/1106/019, Keowee Hydro at Oconee.

The operators note the following indications:

- Grid Frequency is 60.1 Hz
- Keowee Frequency is 59.7 Hz
- Keowee Line Voltage is 13.9 KV
- Keowee 2 Output Voltage is 15.0 kV

1. What direction is the synchroscope turning for the noted conditions?

AND

2. Raising on the UNIT 2 AUTO VOLTAGE ADJUSTER will cause the 2 voltage to rise.

- A. 1. Clockwise.
 2. incoming
- B. 1. Clockwise.
 2. running
- C. 1. Counterclockwise.
 2. incoming
- D. 1. Counterclockwise.
 2. running

Proposed Answer: D

Explanation (Optional):

- Incoming frequency is below grid frequency. Therefore the synchroscope will be turning in a counterclockwise direction.
- Adjusting the voltage regulator can only affect the machine. Therefore running voltage changes.

Technical Reference(s): _____
(Attach if not previously provided) _____
(including version/revision number) _____

Proposed references to be provided to applicants during examination: None

Learning Objective: EL-KHG Obj R7 (As available)

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

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Question:# 48

Given the following plant conditions:

- 1CA Battery charger fails - voltage output 0 vdc
- 1CA Battery voltage - 125vdc
- 1DCB Bus voltage - 127vdc
- Unit 2 DC Bus voltage - 123vdc
- Unit 3 DC Bus voltage - 126vdc

Which ONE of the following will supply power to 1DIA panelboard?

- A. 1CA battery
- B. Unit 3 DC bus
- C. 1DCB bus
- D. Unit 2 DC bus

Proposed Answer: A

Explanation (Optional):

- A. Correct. The 1CA battery voltage is higher than the Unit 2 voltage.
- B. Incorrect. Unit 3's DC Bus is not connected to Unit 1. Plausible because unit 3 does backup unit 1 in the SSF power scheme. The student may get this confused.
- C. Incorrect. For the Vital DC system, the 1DCB bus is not aligned to the 1DCA bus. Plausible because 1DCB Bus is aligned to backup the essential inverters
- D. Incorrect. Unit 2 supplies power to the alternate isolating diodes for 1DIA panelboard. However, since the voltage from Unit 2 is lower than the 1CA battery voltage it will NOT supply power. Plausible if the student fails to realize unit 2 backs up unit 1 when its voltage is higher than unit 1 battery bus.

Technical Reference(s): Lesson Plan EL- DCD
(Attach if not previously provided) _____
(including version/revision number) _____

Proposed references to be provided to applicants during examination: None

Learning Objective: EL-DCD R4 (As available)

Question Source: Bank # _____
Modified Bank # 2009 #14 (Note changes or attach parent)

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New _____

Question History: Last NRC Exam 2009; similar in 2012
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 X
55.43 _____

Comments:

063A3.01 Ability to monitor automatic operation of the DC electrical system, including meters, annunciators, dials, recorders, and indicating lights.

Changed stem to make A correct answer.

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Question: # 49

Given the following plant conditions:

- All three units at 50% power
- 1TC is de-energized
- KHU-1 is not running
- KHU-1 is aligned to the underground

At 1245:

- PCB 8 and 9 trip open and lock out
- ES channel 1 and 2 Keowee Emergency Start Signal has been received

With no operator action, which one of the following correctly describes the status of KHU-1 at 1250?

- A. KHU-1 is operating. KHU-1 control power is supplied by Battery #1
- B. KHU-1 is operating. KHU-1 control power is supplied by Battery Charger #1 energized via KHU-1
- C. KHU-1 is NOT operating. KHU-1 control power is not available due to 1TC being deenergized.
- D. KHU-1 is NOT operating. KHU-1 control power is not available due to PCB 8 and 9 being open.

Proposed Answer: A

Explanation (Optional):

The three sources of power to the charger are:

1. The Keowee unit if operating and output ACB closed
2. Switchyard via PCB-8 and 9
3. 1TC (4160 v Switchgear at ONS)

Refer to AP/0/A/2000/002 (Keowee Hydro Station emergency Start) Encl. 6.4 (KHS One-line Diagram)

- A. Correct. KHU will start. Because power to 1X is not available. The only source of control power is the battery.
- B. Incorrect. . Second part palusible because it would be true if the unit was not connected to the underground feeder and a Switchyard Isolation had occured.
- C. Incorrect. Second part is palusible because the 1TC is a source of power to the Battery charger.
- D. Incorrect. Second part is plausible because the Switchyard via PCB-8 and 9 is a source of power to the Battery charger.

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Technical Reference(s): Lesson Plan EL-KHG, Keowee Hydro Generators
Attach if not previously provided) AP/0/A/2000/002 (Keowee Hydro Station emergency Start)
(including version/revision number) Encl. 6.4 (KHS One-line Diagram)

Proposed references to be provided to applicants during examination: None

Learning Objective: _____ (As available)

Question Source: Bank # _____
Modified Bank # 2009Q51 (Note changes or attach parent)
New _____

Question History: Last NRC Exam _____

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 X
55.43 _____

Comments:

064K2.03 Knowledge of bus power supplies to Control power

Reformatted stem, also rewrote answers to make it a two by two response. Changed C answer to balance two by two responses.

Oconee Nuclear Station 2013-301 Written Examination

Question # 50.

An Operator has just completed adjusting the setpoints on 1RIA-57, High Range Containment Monitor, in accordance with PT/0/A/0230/001, Radiation Monitor Check.

Which ONE of the following is an indication of a satisfactory source check?

- A. Alert Alarm Actuation ONLY
- B. Alert AND High Alarm Actuation
- C. Area Monitor Fault Alarm Actuation
- D. Indication remains at .75 R/HR with no alarms

Proposed Answer: D

Explanation (Optional):

- A. Incorrect. Applicant might believe that source check causes indication to rise and cause an ALERT.
- B. Incorrect. If applicant believes counts will rise, it is plausible to get a HIGH alarm.
- C. Incorrect. If applicant does not understand rad monitors, they could expect a FAULT alarm.
- D. Correct. No change in counts expected. Procedure states that indication should remain from 5E-01 to 1.

Technical Reference(s): PT/0/A/0230/001, Radiation Monitor Check, Rev 165
(Attach if not previously provided) _____
(including version/revision number) _____

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 _____
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 x
55.43 _____

Comments:

073A4.03 Ability to manually operate and /or monitor in the control room: Check source for operability demonstration.

Oconee Nuclear Station 2013-301 Written Examination

Question 51

Unit 3 initial conditions:

- Reactor Power = 100%
- 3A LPSW Pump operating
- LPSW Line leak occurs

Current conditions:

- Unit 3 LPSW Pressure = 60 psig, decreasing slowly
- Operating LPSW pump(s) amps slowly increasing

Based on these conditions, which ONE of the following describes the status of the Unit 3 LPSW Pumps and an appropriate action per AP/3/A/1700/024, Loss of LPSW?

- A ONLY 3A LPSW pump is running; secure operating LPSW pump.
- B ONLY 3A LPSW pump is running; reduce LPSW loads as needed.
- C BOTH 3A and 3B LPSW pumps are running; reduce LPSW loads as needed.
- D BOTH 3A and 3B LPSW pumps are running; secure operating LPSW pumps.

Proposed Answer: C

K/A Match Analysis

Requires knowledge of LPSW pump auto-start setpoint and status (i.e. predicting the impact of loss of service water header pressure), and a procedural action to mitigate the malfunction.

Answer Choice Analysis

- A. **INCORRECT.** Pump status is incorrect in that both LPSW pumps are running because 3B LPSW pump auto-started when LPSW pressure was below the setpoint of 70 psig for 10s. With pressure at 60# slowly decreasing, it should be clear that at least 10s have passed. Second part is incorrect but plausible because AP/024 directs this if there are indications of cavitation (AP/24 Step 4.1, IAAT step).
- B. **Incorrect.** Pump status is incorrect as described above. Action in second part is correct (AP/24 Step 4.24).
- C. **Correct.** Both LPSW pumps are running because current LPSW pressure is below the Auto Start signal setpoint of 70 psig. Action directed if LPSW pressure remains below normal is to reduce LPSW loads.
- D. **Incorrect.** Pump status is correct as noted above in that current LPSW pressure is below the Auto Start signal setpoint of 70 psig. Second part is correct as discussed above.

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Technical Reference(s): Lesson Plan SSS-LPW, Rev. 21
(Attach if not previously provided) EAP-APG Lesson Plan, Rev. 7b
(including version/revision number) AP/3/A/1700/024, Rev. 27

Proposed references to be provided to applicants during examination: None

Learning Objective: SSS-LPW R23, EAP-APG R9

Question Source: Bank # _____
Modified Bank # 52 (parent question attached)
New _____

Question History: Last NRC Exam: Oconee 2009-301

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

10 CFR Part 55 Content: 55.41 X
55.43 _____

Comments:

076A2.02 Ability to (a) predict the impacts of the following malfunctions or operations on the SWS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Service water header pressure. 2.7 / 3.1
(CFR 41.5 / 43.5)

Author: Mike Donithan

Parent Question

2009 NRC REACTOR OPERATOR EXAM

Question 52

Unit 3 initial conditions:

- Reactor Power = 100%
- 3A LPSW Pump operating
- LPSW Line leak occurs

Current conditions:

- Unit 3 LPSW Pressure = 60 psig decreasing slowly
- Operating LPSW pump(s) amps slowly increasing

Based on the conditions above, which ONE of the following describes the status of the Unit 3 LPSW Pumps and an appropriate action per AP/24 (Loss of LPSW)?

- A. ONLY 3A LPSW pump is running / secure operating LPSW pump
- B. ONLY 3A LPSW pump is running / reduce LPSW loads as needed
- C. BOTH 3A and 3B LPSW pumps are running / reduce LPSW loads as needed
- D. BOTH 3A and 3B LPSW pumps are running / secure operating LPSW pumps

Oconee Nuclear Station 2013-301 Written Examination

Question 52

Plant conditions:

- ALL off-site power sources have been lost (230KV and 525KV transmission lines)
- Keowee has energized the MFB via the overhead power path
- IA pressure = 85 psig and decreasing
- ALL Diesel air compressors are OFF
- No operator actions have been taken

Which ONE of the following is correct?

- A. ONLY the Back-up Instrument Air Compressors will be operating
- B. ONLY the Auxiliary Instrument Air Compressors will be operating
- C. ALL Auxiliary Instrument Air Compressors and ALL Back-up Instrument Air Compressors will be operating
- D. ALL Auxiliary Instrument Air Compressors and the Primary Instrument Air Compressor will be operating

Proposed Answer: B

Explanation:

See attached lesson plan references.

Answer Choice Analysis:

- A. Incorrect. Back-up Instrument Air Compressors will not be operating because they are powered from a load shed power supply.
- B. Correct. Auxiliary Instrument Air compressors will be operating because they are powered from a non-load shed source.
- C. Incorrect. First part correct. Second part incorrect.
- D. Incorrect. Primary Instrument Air Compressor will not be operating because it is powered from the 230 KV or 525 KV switchyard.

Technical Reference(s):
(Attach if not previously provided)

OP-OC-SSS-IA Lesson Plan, Rev. 20c

Oconee Nuclear Station 2013-301 Written Examination

(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: Lesson Plan FH-SFC Obj. R11

Question Source: Bank # 46
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam: OC-2007-301

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

10 CFR Part 55 Content: 55.41 (4)
55.43

Comments:

K/A: 078K2.01 Knowledge of the electrical power supplies to: Instrument Air Compressor
2.7/2.9

Author: Mike Donithan

Oconee Nuclear Station 2013-301 Written Examination

Question 53

Unit 1 is operating at 100% power, steady state.

A complete loss of Instrument Air (IA) and Auxiliary Instrument Air (AIA) occurs.

Which of the following describes RCP seal cooling and Pressurizer level response?

ASSUME NO OPERATOR ACTIONS

RCP Seal cooling will be maintained by (1). Pressurizer level will (2).

- A 1) Component Cooling
 2) Decrease
- B 1) Component Cooling
 2) Increase
- C 1) HPI Seal Injection
 2) Decrease
- D 1) HPI Seal Injection
 2) Increase

Proposed Answer: D

K/A Match Analysis

Requires knowledge of the effects that loss of IA will have on RCP seal injection flow, RCP thermal barrier flows, and RCS letdown flow. These systems contain pneumatic valves.

Answer Choice Analysis

- A. Incorrect. First part incorrect: 1CC-8, CC RETURN OUTSIDE BLOCK, fails closed, isolating the flowpath for RCP thermal barrier flow. Second part incorrect: 1HP-31, RCP SEAL FLOW CONTROL, fails open, providing more seal injection flow than before; 1HP-5, LETDOWN ISOLATION, fails closed. More flow into the RCS plus no flow out makes PZR level increase.
- B. Incorrect. First part incorrect as discussed in A. Second part correct, explained in A.
- C. Incorrect. First part correct: 1HP-31, RCP SEAL FLOW CONTROL, fails open, providing more seal injection flow than before. Second part incorrect, discussed in A.
- D. Correct. Both parts correct as discussed above.

Technical Reference(s): OP-OC-SSS-IA Rev. 20c
(Attach if not previously provided) OP/1/A/1700/022 Rev. 27

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Proposed references to be provided to applicants during examination: None

Learning Objectives: OP-OC-SSS-IA R47, R48

Question Source: Bank # 823 (From Oconee exam bank. Changed order of distractors, making D correct instead of A.)

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content: 55.41 X
55.43 _

Comments:

078K3.02 (Instrument Air System) Knowledge of the effect that a loss or malfunction of the IAS will have on systems having pneumatic valves and controls. 3.4/3.6 (CFR: 41.7/45.6)

Author: Mike Donithan

Supporting References

OP-OC-SSS-IA

2.5 (OBJ. R48) Loss of Instrument Air (AP/22)

F.7.a).2) CC-8 could fail shut at ≈ 80 PSIG decreasing AIA pressure. This will cause CC Pumps to trip and loss of CC flow to: Letdown Coolers, RCPs, QT Coolers, and the CRDM Stators.

F.7.b) (OBJ. R47) Dispatch an operator to throttle HP-31 to ≈ 32 (≈ 40) gpm seal injection flow.

4. This is a reminder that as IA pressure degrades; HP-31 will begin to fail to its "no air" position, which is OPEN as IA pressure decreases below 70 PSIG.
5. As ...
6. General guidance on what to expect if HP-31 fails fully open:
7. Seal injection flow will increase to ≈ 60 GPM.
8. PZR level will increase at ≈ 2.5 inches per minute and LDST level will decrease at ≈ 2 inches per minute.
9. PZR and LDST levels are affected because HP-5, the Letdown Isolation valve has failed shut on loss of air pressure. The additional seal injection flow, coupled with the inability to compensate with increased letdown, causes the mismatch.

Oconee Nuclear Station 2013-301 Written Examination

Question 54

Unit 1 initial conditions:

- Reactor Power = 100%
- Reactor Building average temperature = 120°F stable
- RBCUs 1A, 1B, & 1C running in High Speed

Current conditions:

- Inadvertent ES Channel 5 actuation

Based on these conditions, which ONE of the following describes the response of RB Pressure and the RB high pressure limit per TS 3.6.4, Containment Pressure?

With no operator action, RB Pressure will (1). The RB high pressure TS Limit is (2).

- A 1) slowly increase
 2) ≤ 1.2 psig
- B 1) slowly increase
 2) ≤ 2.45 psig
- C 1) remain the same
 2) ≤ 1.2 psig
- D 1) remain the same
 2) ≤ 2.45 psig

Proposed Answer: A

K/A Match Analysis

Requires knowledge of the cause/effect relationship that changing RBCU configuration has on RB pressure due to the resulting change in temperature.

Answer Choice Analysis

- E. **Correct:** ES channel 5 actuation causes 1A & 1B RBCUs to shift to low speed which reduces cooling air flow. This results in less heat removal and RB pressure will increase due to heatup of containment atmosphere. RB high pressure limit per TS 3.6.4 is 1.2 psig.
- F. Incorrect: First part correct as described above. Second part is incorrect; plausible in that the **low** TS limit is (-) 2.45 psig, not (+) 2.45 psig.
- G. Incorrect: First part incorrect; plausible if candidate fails to realize that two RBCUs will shift to low speed or that low speed will remove less heat. Also plausible if it is misunderstood that increasing RB temperature will not cause a corresponding increase in RB pressure. Second part is correct.

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H. Incorrect: First part incorrect as discussed above. Second part incorrect as discussed above.

Technical Reference(s): OP/1/A/1104/015 Rev. 22b

(Attach if not previously provided) PT/1/A/0600/001 Rev. x

(including version/revision number) TS 3.6.4

Proposed references to be provided to applicants during examination: None

Learning Objectives: OP-OC-PNS-RBC (R20), ADM-TSS (R4)

Question Source: Bank # _____
Modified Bank # 55 (Note changes or attach parent)
New _____

Question History: Last NRC Exam: Oconee NRC 2009

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

10 CFR Part 55 Content: 55.41 X
55.43 _____

Comments:

103K1.01 Knowledge of the physical connections and/or cause-effect relationships between the containment system and the Containment Cooling System.

3.6/3.9 (CFR: 41.2 to 41.9, 45.7 to 45.8)

Author: Mike Donithan

Oconee Nuclear Station 2013-301 Written Examination

Question 55

Proposed Question:

Unit 1 initial conditions:

- LOCA
- RB Pressure = 4.5 psig increasing
- RCS Pressure = 1500 psig decreasing

Current conditions:

- RB Pressure = 2.5 psig stable
- RCS Pressure – 1800 psig stable
- ES reset is desired

Based on current conditions, which ONE of the following describes the minimum set of components that must be reset in order to allow the HPIPs ES Logic to be reset?

- A 1. HPI Trip Bistables in Analog Channels A/B/C
 2. ES Ch 1 & 2 Digital Channels
- B 1. HPI and RB Pressure Trip Bistables in Analog Channels A/B/C
 2. ES Ch 1 & 2 Digital Channels
- C 1. HPI Trip Bistables in Analog Channels A/B/C
 2. ES Ch 3 & 4 Digital Channels
- D 1. HPI and RB Pressure Trip Bistables in Analog Channels A/B/C
 2. ES Ch 3 & 4 Digital Channels

Proposed Answer: B

K/A Match Analysis

Requires knowledge of HPIP reset following ES (Containment pressure) actuation.

Answer Choice Analysis

- E. Incorrect. Analog Reset requires resetting **both** the HPI Bistable and the RB Pressure bistable. Plausible because the 10 psig RB Pressure trip is self-resetting but the 4 psig bistable is not. Digital Channels 1 & 2 correct.
- F. **Correct.** Analog Reset requires resetting **both** the HPI Bistable and the RB Pressure bistable. Digital Channels 1 & 2 correct.
- G. Incorrect. First part incorrect, discussed above. Digital Channels 3 & 4 for LPI, not HPI.
- H. Incorrect. First part correct. Second part incorrect, discussed above.

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Technical Reference(s): EP/1/A/1800/001 Encl. 5.41 (ES Recovery) Rev. 39
(Attach if not previously provided) IC-ES Lesson Plan, Rev. 18a
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: IC-ES (R17)

Question Source: Bank # 55
Modified Bank # _____
New _____

Question History: Last NRC Exam: Oconee 2008-301

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

10 CFR Part 55 Content: 55.41 X
55.43 _____

Comments:

103K1.08 Knowledge of the physical connections and/or cause-effect relationships between the containment system and SIS, including action of safety injection reset. 3.6/3.8 (CFR: 41.2 to 41.9; 45.7 & 45.8)

Author: Mike Donithan

Supporting References

OP-OC-IC-ES

Enabling Objective 15: Describe the actions necessary to properly return HPI pumps, Reactor Building Cooling Units and Keowee Hydro Units to normal operation following ES actuation. (R17)

EP/1/A/1800/001 Enclosure 5.41, ES Recovery

4. Perform the following to reset ES (1UB1):

- Depress RESET for CH 1.
- Depress RESET for CH 2.

Oconee Nuclear Station 2013-301 Written Examination

Question 56

Which one of the following conditions would result in a Control Rod Out Inhibit?

- A. Count rate = 675 cps increasing and Wide range NI-1 startup rate = 1.8 dpm
- B. Count rate = 675 cps increasing and Source range NI-1 startup rate = 1.1dpm
- C. Reactor Power = 62% and Control Rod Group 1 loses its Group Out Limit
- D. Reactor Power = 58% and Group 6 Rod 5 becomes misaligned by > 9 inches

K/A Match Analysis

Question requires knowledge of CRDS circuitry and effect of primary power on rod motion

Answer Choice Analysis

- A. Incorrect, this would be correct if SUR exceeded 2 dpm
- B. Incorrect, plausible since the SUR number would be correct if asking about exceeding the max allowed stable startup rate per the startup procedure (1DPM)
- C. Correct, with power above 60% this will occur
- D. Incorrect, plausible since this would be correct if asking about an Asymmetric Fault or if power was greater than 60 %

Technical Reference(s): IC-CRI

(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: IC-CRI-R32

Question Source: Bank # _____
Modified Bank # 2009B Oconee NRC exam question 56
New

Question History: Last NRC Exam: Modified from 2009B
Changed answer from A to C by changing Reactor power and SUR.

Oconee Nuclear Station 2013-301 Written Examination

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

10 CFR Part 55 Content: 55.41 X
55.43 _

Comments:

001K5.65 Knowledge of the following operational implications as they apply to the CRDS: CRDS Circuitry, including effects of primary/secondary power mismatch on rod motion 3.2/3.6 (CFR: 41.5, 45.7)

Author: G. Laska

Oconee Nuclear Station 2013-301 Written Examination

Question 57

Unit 1 was operating at 100% power when the following trends were observed:

- RCS pressure began to lower
- Pressurizer level began to rise
- Subcooling Margin (SCM) began to lower
- Quench tank level began to rise
- Quench tank pressure began to rise

Which one of the following correctly describes the initial effect on containment when these trends are observed?

- A. Containment pressure rises. Containment radiation levels rise.
- B. Containment pressure rises. Containment radiation levels remain constant.
- C. Containment pressure remains constant. Containment radiation levels rise.
- D. Containment pressure remains constant. Containment radiation levels remain constant.

Proposed Answer: D

Answer Choice Analysis:

- A. Incorrect. Initially no discharge to containment to cause P to rise or radiation levels to rise. QT rupture disk relieves at 100 psig.
- B. Incorrect. Initially no discharge to containment to cause P to rise. Second part correct.
- C. Incorrect. First part correct. Second part incorrect, no discharge to containment to cause radiation levels to rise.
- D. Correct.** There will be no effect on containment until QT rupture disk blows.

Technical Reference(s): Alarm Response Guide 1SA-18/A-1
(Attach if not previously provided) Lesson Plan PNS-RCS
(including version/revision number) Lesson Plan PNS-PZR

Proposed references to be provided to applicants during examination: None

Learning Objective: Lesson Plan PNS-PZR Obj. R19 & R32

Question Source: Bank # 3
Modified Bank # _____ (Note changes or attach parent)

Oconee Nuclear Station 2013-301 Written Examination

New _____

Question History:

Last NRC Exam: OC-2006-301

ES-401-9 Column 7 Explanation for that exam:

"I don't think anyone will miss it. Very little discriminatory value but meets minimum standards. *It's about all you can do with this KA.* RFA 04/25/06" [emphasis added]

Philosophical Ranking was 1, "Bullet Proof". LOD was 2.

Question Cognitive Level:

Memory or Fundamental Knowledge

Comprehension or Analysis

X

10 CFR Part 55 Content:

55.41 (7)

55.43

Comments:

K/A: 002K3.03 Knowledge of the effect that a loss or malfunction of the RCS will have on Containment. 4.2/4.6

Author: Mike Donithan

Oconee Nuclear Station 2013-301 Written Examination

Question 58

The “Rod Misalignment Correction” feature of rod control will cause some rods in a group to not move until all rods in the group are within (1) of the Group Average.

This logic affects only (2) rod motion.

- A
 - 1) 1%
 - 2) inward
- B
 - 1) 1%
 - 2) outward
- C
 - 1) 2%
 - 2) inward
- D
 - 1) 2%
 - 2) outward

Proposed Answer: B

K/A Match Analysis

Requires knowledge of control rod sequencing logic.

Answer Choice Analysis

- I. Incorrect. First part correct. Second part plausible if applicant rationalizes that inward motion of a misaligned bank would be more limiting with regard to radial flux shaping.
- J. **Correct.** Reference OP-OC-IC-CRI 2.8.Q.
- K. Incorrect. First part incorrect. Plausible because the example/discussion in the lesson plan discusses an initial misalignment of 2%. And 2% is a reasonably small misalignment.
- L. Incorrect.

Technical Reference(s):

OP-OC-IC-CRI Lesson Plan, Rev. 13c

(Attach if not previously provided)

Oconee Nuclear Station 2013-301 Written Examination

(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: OP-OC-IC-CRI (R30)

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
55.43 _____

Comments:

014A4.01 (Rod Position Indication System) Ability to manually operate and/or monitor in the control room: Rod selection control. 3.3/3.1 (CFR: 41.7; 45.5 to 45.8)

Author: Mike Donithan

Supporting References

OP-OC-IC-CRI

2.8

Q. (Obj R30) Rod Misalignment Correction

“Rod Misalignment Correction” causes withdraw commands to result in some selected rods in a group not moving to achieve alignment within 1%. For example if Group 7, Rod 1 is low by 2%, it will move alone for the first 1% of withdrawal until it is within 1% of the Group Average, then all Group 7 rods will move out together. If Group 7 Rod 1 is high by 2%, then rods 2-8 will move outward for 1% of travel and then rod 1 will join in outward movement once it is within 1% of Group Average.

- The logic uses RPI indication (so a stuck rod has no effect on commands).
- Rods flagged as “Dropped” are ignored in this logic (excluded from the group & movement commands).
- This logic has no effect on inward motion (all selected rods move on “In” commands).

Oconee Nuclear Station 2013-301 Written Examination

Question 59

Unit 1 initial conditions:

- Reactor power = 100%
- OAC Computer is out of service

Power Range channel NI-5 begins to drift low and the Control Room Supervisor directs you to remove it from service for calibration.

Which one of the following describes the instrumentation used to determine quadrant power tilt in accordance with OP/1/A/1105/014, Control Room Instrumentation Operation and Information?

- A The three operable NIS channels
- B Incore Detectors
- C Backup Incore Detectors
- D Quadrant power tilt cannot be determined

Proposed Answer: C

K/A Match Analysis

Requires knowledge of the hierarchy of QPT measurement instruments (incore detectors via OAC Computer, then excore NIs, then backup incore detectors), and that removing one NI from service while the OAC computer is OOS makes NI QPT unavailable.

Answer Choice Analysis

- M. Incorrect. If any Power Range NI 5 through 8 is inoperable, outcore detectors **shall not** be used to measure QPT.
- N. Incorrect. Plausible since it would be correct if the Computer Reactor Calculation Package was operable, but with the OAC OOS it is not.
- O. **Correct.** Per OP/1105/014, the hierarchy is: Incore Detectors (Computer Reactor Calculation Package), Outcore Detectors (Power Range NIs), Backup Incore Detectors (ref. PT/0/A/1103/019).
- P. Incorrect. See C.

Technical Reference(s):

OP/1/A/1105/014 Encl. 4.13, Reactor Parameter Info.,
Rev. 35

Oconee Nuclear Station 2013-301 Written Examination

(Attach if not previously provided) OP-OC-ADM-PIS Lesson Plan, Rev. 6a
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: OP-OC-ADM-PIS (R6)

Question Source: Bank # _____
Modified Bank # 14 (Note changes or attach parent)
New _____

Question History: Last NRC Exam: Oconee 2009B

Original question was written to APE057 AA1.05, had OAC Computer OOS then deenergized 1KVIC, making one of the PR NIs inoperable. It then asked how to determine axial imbalance. Changed to manually taking a PRNI out of service, then asking about QPTR. Two distractors are new. Parent question attached.

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis

10 CFR Part 55 Content: 55.41
55.43

Comments:

015A1.04 Ability to predict and/or monitor changes in parameters (to prevent exceeding deign limits) associated with operating the NIS controls including: Quadrant power tilt ratio. 3.5/3.7 (CFR: 41.5; 45.5)

Author: Mike Donithan

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2009B ONS RO NRC Examination QUESTION 14 B

Unit 1 initial conditions:

- Reactor power = 100%
- OAC Computer Out of Service
- 1KVIC deenergized

Based on the above conditions, which ONE of the following describes the instrumentation used to determine axial imbalance in accordance with OP/1/A/1105/014 (Control Room Instrumentation Operation and Information)?

- A. Incore Detectors
- B. Backup Incore Detectors
- C. Imbalance readout indications in RPS cabinets
- D. Control Room Power Range Dixon gauge indications

Supporting References

ADM-PIS

Enabling Objectives

F. When given a copy of

G. Concerning Reactor Imbalance.....

H. Concerning Quadrant Power Tilt: (R6)

H.1 Determine the correct indication to use for Reactor Quadrant Power Tilt for any given plant conditions.

I. Recognize that if the Reactor Calculations Package is NOT running, the requirement is to verify minimum incore detector operability requirements are met per PT/0/A/1103/019 (Backup Incore Detector System). (R15)

2.2.B:

6. (OBJ. R6) Quadrant Power Tilt (QPT) (Limits in COLR)

a) Order of preference in determining QPT is as follows

1) Incore detectors

2) Outcore detectors

3) Backup Incore detectors

b) The QPT surveillance in PT/600/01 Enclosure Mode 1 & 2 gives computer points to use. These CP's are for Incore QPT. If Incore QPT is not available, the next choice is to use Outcore detectors.

c) If Outcore detectors are needed, PT/0/A/1103/018, Excore Tilt Calculations, needs to be performed to determine an Excore value for QPT. Computer indications for Excore QPT should not be used to determine actual QPT without utilizing this PT. Each time a NI calibration is performed, Excore QPT indications are "zeroed". This PT will take that into account and determine a correction factor (if needed) to be applied to the available Excore Tilt indications to determine actual QPT. If Excore Tilt can not be determined, Backup Incore detectors should be used.

d) If Backup Incore detectors are needed, we refer to PT/0/A/1103/019, Backup Incore Detector System. That procedure will walk through verifying required detectors are operable and determining QPT from them.

- e) **(OBJ. R15) IF** the Reactor Calculations package is **NOT** running, PT/600/001 will refer the operator to OP/1105/014 for Axial Imbalance and QPT. OP/1105/014 states that if the Reactor Calculations package is **NOT** running, verify minimum incore detector operability requirements are met per PT/0/A/1103/019 (Backup Incore Detector System). PT/0/A/1103/019 states that if the incore system is NOT available on the unit computer and the backup recorder points are NOT operable per this procedure, then Rx power shall be reduced to below 80% of the power allowable for the existing RCP combination within 8 hours unless either of the following:
- 1) The incore system is restored in the unit computer
 - 2) The backup recorder points are restored to meet the minimum requirements for operability (reference SLC 16.7.8)

OP/1/A/1105/014

Enclosure 4.13, **Reactor Parameter Information**, Page 3 of 4

3.2.4 **Order of preference** of measurement systems to determine axial imbalance and quadrant power tilt is as follows:

- A. **Incore Detectors (Computer Reactor Calculation Package).**
- B. **Outcore Detectors (Power Range Outcore Detectors).**
- C. **Backup Incore Detectors. Refer to PT/0/A/1103/0**

Oconee Nuclear Station 2013-301 Written Examination

Question 60

Given the following:

- Unit 1 operating at 100% power, steady state.
- The Operator Aid Computer (OAC) Subcooling Margin (SCM) program indicates that 1 of the 47 Core Exit Thermocouples (CETCs) is reading 65F higher than any of the others.

The input from this CETC to the OAC SCM program is (1). The computer-calculated subcooling margin will be (2).

- A 1) valid
2) higher
- B 1) valid
2) unaffected
- C 1) not valid
2) higher
- D 1) not valid
2) unaffected

Proposed Answer: D

K/A Match Analysis

Requires knowledge of the malfunction of a CETC and what effect it will have on the OAC SCM program.

Answer Choice Analysis

- I. Incorrect. First part incorrect; plausible if candidate fails to remember that the ICCM will kick out any thermocouple reading 50F higher or lower than the average five high CETCs. Second part is incorrect: since the ICCM kicks this CETC out, there is no effect on calculated 5-high. Plausible if applicant misses the first part, then average 5 high *would* be higher.
- J. Incorrect. First part incorrect as discussed above. Second part correct: since the ICCM kicks this CETC out, there is no effect on calculated 5-high.
- K. Incorrect. First part is correct; the ICCM will kick out any thermocouple reading 50F higher or lower than the average five high CETCs. Second part incorrect as discussed above.
- L. **Correct:** Both parts correct as discussed above.

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Technical Reference(s): OP-OC-IC-RCI Rev. 22c
(Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objectives: OP-OC-IC-RCI, one or more of these from Section 2.7 heading: R36, 37, 38, 39, 40, 41, 42, 60 {can't tell which apply specifically to 2.7.E.7.a).1).a)}

Question Source: Modified Bank # 58 The parent question postulated open circuits on 2 CETCs. Changed to have one reading >50F higher than average of five high. Parent question asked about impact on ICCM; modified to ask impact to OAC SCM. Parent question attached.

Question History: Last NRC Exam: Harris NRC 2011

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

10 CFR Part 55 Content: 55.41 X
55.43 _

Comments:

017K6.01 (In-Core Temperature Monitor System) Knowledge of the effect of a loss or malfunction of ITM sensors and detectors. 2.7/3.0 (CFR: 41.7, 45.7)

Author: Mike Donithan

Supporting References

OP-OC-RT-RBC

2.7 (OBJ R36,37,38,39,40,41,42,60) RCS and Core Subcooling Margin Programs
(OAC PROGRAM)

M. Inputs – OAC Calculation

7. Incore Thermocouple temperatures (CETCs)

a) When NI power is >2% (*Based on the ave. of the 4 NI's*), the program takes the average of the “operable” CETCs not being used by the SSF (total = 47).

1) “Operable”

(a) If any of the 47 CETCs are >50°F higher or lower than the average it will not be used in the calculation.

2) If a CETC is in “scan lockout” or contains an “inserted value” it will not be used in the calculation.

b) Using 47 CETCs gives a more accurate/conservative Core Subcooling Margin indication and will not lead to tripping the RCPs unnecessarily following a “normal” reactor trip when the system is not actually saturated.

c) When NI power is < 2% (*Based on the ave. of the 4 NI's*) the program takes the average of the 5 highest qualified CETCs (24) from both ICCM Train A (12 CETCs) and ICCM Train B (12 CETCs).

1) Use of only the qualified (PAM) CETCs is not mandatory unless a hostile environment exists in the Reactor Building.

2) If NI power is < 2% then the OAC assumes a hostile environment may exist in the Reactor Building.

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Parent question – Harris 2011

58. Given the following plant conditions:

- The crew is performing EPP-004, Reactor Trip Response
- Natural circulation verification is in progress.
- Core Exit Thermocouples G2 and K5 have failed due to open circuits.

With these open circuit failures, the input from these thermocouples to the Inadequate Core Cooling Monitor (ICCM) will be (1) and the subcooling margin calculated by ICCM will be (2).

- A. (1) failed low
(2) unaffected
- B. (1) failed low
(2) higher
- C. (1) failed high
(2) unaffected
- D. (1) failed high
(2) lower

Plausibility and Answer Analysis

When a thermocouple fails or is taken out of service, a reading of 50°F will be displayed for that thermocouple on the MCR RVLIS display screen. A saturation temperature is calculated based on PRC9445. This saturation temperature is then compared with the average of the 5 hottest of all 51 operable thermocouples (Point TRC9300) or the average of RCS wide range TH temperatures, TE-413/423/433 (Point TRC9413), used only if TRC9300 is not available.

- A. Correct. A failed low input will not be used on ICCM (Only highest temperature and lowest pressure).*
- B. Incorrect. Plausible if a failed low input is used in the ICCM calculation.*
- C. Incorrect. Plausible because RTDs fail high on an open circuit.*
- D. Incorrect. Plausible because RTDs fail high on an open circuit.*

017 In-core Temperature Monitor

017K6.01 Knowledge of the effect of a loss or malfunction of the following ITM system components: Sensors and detectors. (CFR: 41.7 / 45.7) 2.7/3.0

Technical Reference: ICCM Lesson Plan

References to be provided: None

Learning Objective: ICCM Objective 5.d

Question Origin: Bank OIT Dev. 017 K6.01 1

Tier/Group: T2G2

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Question # 61

Unit 1 is in Mode 5 with Reactor Building Main Purge in operation.

Which ONE of the following will cause the RB Main Purge Fan to trip?

A	Suction pressure = 5 inches of water vacuum
B	1RIA-45, UNIT VENT GAS NORM, reaches its ALERT setpoint (Noun name from LP RAD-RIA p. 43; couldn't find it anywhere else. That LP even calls it something else (similar) on p. 39.)
C	Statalarm 1SA9/B-3, RBV PURGE INLET TEMPERATURE LOW, alarms
D	1PR-3 (RB PURGE CONTROL) demand is reduced to 5% open

Proposed Answer: D

K/A Match Analysis

The containment buildings at Oconee are called Reactor Buildings, so Reactor Building Main Purge is the Containment Purge System. Several interlocks will stop containment purge, and this question through its correct answer and distracters tests applicant knowledge of design features and interlocks.

Answer Choice Analysis

- E. INCORRECT. Plausible because vacuum on the suction piping must be less than 9" of water to start the Purge Fan, and will trip the fan if exceeded (Ref. LP PNS-RBP 2.7.A.4).
- F. INCORRECT. Plausible because 1RIA-45 WILL stop the purge, but only at the HIGH setpoint, not ALERT (Ref. ARG 1SA-08, B-9). Closes PR-2 through PR-5; any of these valves closed will trip the fan (Ref. LP PNS-RBP 2.7.A.1).
- G. INCORRECT. Plausible because OP/1/A/1102/014, RB Purge System, P&L 2.7 tells the operator to stop the RB Purge Fan on this alarm. LP PNS-RBP says the same. ARG 1SA-09/B-3 doesn't indicate any automatic actions or even direct this as a manual action.
- H. CORRECT. The purge fan is interlocked so that it will not start unless PR-3 is open >10% (Ref. LP PNS-RBP 2.2.B.12). Fan will trip if PR-3 <10% open (Ref. LP PNS-RBP 2.7.A).

Technical Reference(s): OP/1/A/1102/014, RB Purge System, Rev. 43.

(Attach if not previously provided) Lesson Plan OP-OC-PNS-RBP, Rev 12a.

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(including version/revision number) OP/1/A/6101/001, Rev 43, Alarm Response Guide 1SA-01 A-10 & B-10

Proposed references to be provided to applicants during examination: None

Learning Objective: Lesson Plan OP-OC-PNS-RBP Obj. R7

Question Source: Bank # _____
Modified Bank # 56 (Changed distracter D to make it correct. Moved distracter C to A and made new distracter C.)
New _____

Question History: Last NRC Exam: Oconee-2004-301

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

10 CFR Part 55 Content: 55.41 X
55.43 —

Comments: 029K4.03 (Containment Purge System) - Knowledge of design feature(s) and/or interlock(s) which provide for automatic purge isolation. (CFR 41.7)

3.2*/3.5

Author: Mike Donithan

Supporting References

OP/1/A/6101/001 Rev. 43
Alarm Response Guide 1SA-01

B-10

ES CHANNEL 2 TRIP

2. Automatic Action

- 2.12 **1PR-2 (RB PURGE OUTLET (PR)) closes.**
- 2.13 **1PR-3 (RB PURGE OUTLET SWITCH) closes.**
- 2.14 **1PR-4 (RB PURGE INLET) closes.**
- 2.15 **1PR-5 (RB PURGE INLET) closes.**

Lesson Plan OP-OC-PNS-RBP Rev. 12a

2.2.B.12 (Obj. R4) Throttle > 60 % open PR-3

- f) The Purge Fan Motor is interlocked so that it will not start unless PR-3 is open >10%.

2.6 Abnormal Operations

A. (Obj. R7) Engineered Safeguards Operation

3. The R.B. Purge Isolation Valves are non-essential containment isolation valves and receive signals to isolate as follows:

<u>Valve</u>	<u>Channel</u>
PR-1 & 6	ES-1
PR-2 through 5	ES-2

B. (Obj. R5) Unit Vent Radiation Effluent - High

1. RIA-45 monitors the unit vent noble gas effluent.
- a. If unit vent activity exceeds allowable release limits, RIA-45 initiates a signal to close PR-2 through PR-5, and trips the Main and Mini Purge Fans.

2.7 (Obj. R4) Reactor Building Purge Interlocks

A. Main Purge Fan

- The following interlocks must be satisfied to allow the Purge Fan to start and they will trip the Purge Fan during operation if they are not satisfied:
 - RIA-45 and RIA-46 must be reading less than HIGH setpoint. This allows PR-2, 3, 4, and 5 to be opened.
 - PR-1 through PR-6 must be open
 - Vacuum on the suction piping must be less than 9 inches of water vacuum.

OP/1/A/1102/014 Rev. 43 Page 2 of 3

2. Limits And Precautions

- 2.7 If **Statalarm 1SA-9/B-3** (RB Purge Inlet Temperature Low) alarms and Purge Inlet Piping is in service, **stop RB Main Purge Fan**.

OP/1/A/6101/009 Rev. 43

Alarm Response Guide 1SA-09

B-3

RBV PURGE INLET TEMPERATURE LOW

1. Alarm Setpoint

- 1.1 40°F

2. Automatic Action

None

3. Manual Action

- 3.1 **IF** RBV Purge is operating or will be operating within 12 hours, align steam to RB Purge Inlet heating coil per OP/0/A/1104/037 (Plant Heating).

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Question 62

Unit 1 is in a refueling outage with the following conditions:

- The Reactor Vessel has been refueled
- The Refueling Cavity is still full
- It is desired to drain the Reactor Cavity and Fuel Transfer Canal with ONLY a SF Cooling Pump and NOT use LPI Pump(s)
- LPI is in service providing decay heat removal
- OP/1/A/1102/015, Filling and Draining FTC, is in progress to drain the Refueling Cavity in preparation for setting the Reactor Head onto the Vessel
- OP/015 Enclosure 4.9, Draining FTC, has just been entered

Enclosure 4.9 will align the (1) SF Cooling Pump to take suction from the FTC deep end and discharge to the U1 BWST.

If FTC level is lowered too far, damage to the (2) could occur.

Which ONE of the following completes the statements above?

- A. 1. 1B
2. aligned SF cooling pump
- B. 1. 1B
2. operating LPI pump(s)
- C. 1. 1C
2. aligned SF cooling pump
- D. 1. 1C
2. operating LPI pump(s)

Proposed Answer: A

Explanation:

9. OP/1/015 **Enclosure 4.9 aligns ONLY 'B' SF Cooling Pump (SFCP)** to pump down FTC. 'A' & 'C' are not options.
10. Lesson Plan FH-SFC at 2.5.G discusses draining the FTC and says **the respective 'B' SFCP is used** for all three units.
11. OP/015 Enclosure 4.9 Step 6.13.4.B warns of **possible damage to operating LPI pumps**, but only applies if LPI is used to aid in draining the FTC. Substep C. warns of **possible damage to 'B' SFCP**; it would apply anytime 'B' SFCP was being used for FTC drain.
 - a. SFCP could cavitate if FTC deep end level were lowered too far.
 - b. LPI pump(s) could cavitate if RV level were lowered too far, but this can't happen if only the "B" SFCP is being used; RV will only drain to the level of the head flange.
12. System Description Drawing OP-OC-FH-FFC-01 shows that 'B' SFCP is the only pump that can be aligned this way without making one or both of the other pumps inoperable for SFP cooling. Also, 'B' is depicted as the bottom of the three pumps, but a typical flow diagram shows pumps in order from top to bottom, A to C.

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13. 'C' SFCP has different operating characteristics than 'A' & 'B' (Ref. OP/1-2/A/1104/0006 (Rev. 95) P&L 2.11, Notes at Enclosure 4.4 Step 2.4 & Enclosure 4.11 Step 2.24, and Lesson Plan FH-FSC Step 2.5.I.2.b.1).

Answer Choice Analysis:

- A. CORRECT.
- B. INCORRECT. First part is correct. Second part is incorrect but plausible as discussed in Explanation 3.b above.
- C. INCORRECT. First part is incorrect because of Explanations 1 & 2 above. Plausibility from Explanations 4 & 5: A candidate will likely remember that only one SFCP can be aligned for FTC draindown, and may associate that function with the "different" pump; might also remember that it was the "bottom pump" in the drawing and pick SFCP 'C' based on convention. Second part is correct.
- D. INCORRECT. Both parts incorrect as discussed in B & C.

Technical Reference(s): OP-OCFH-SFC Lesson Plan, Rev. 18
(Attach if not previously provided) OP/1/A/1102/015, Filling and Draining FTC, Rev. 83
(including version/revision number) System Description Drawing OP-OC-FH-FFC-01, dated 10/17/05

Proposed references to be provided to applicants during examination: None

Learning Objective: Lesson Plan FH-SFC Obj. R11

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

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content: 55.41 (4)
55.43 —

Comments:

033K1.02 Knowledge of the physical connections and/or cause-effect relationships between the Spent Fuel Pool Cooling System and RHRS. 2.5/2.7

Author: Mike Donithan

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Question 63

Given the following:

- Unit 1 tripped earlier this shift and is in Mode 3
- Cooling down on the Turbine Bypass Valves with the controller/s in HAND

Statalarm 1SA-02 B-12, "ICS HAND POWER FAILURE" annunciates.

An operator dispatched per AP/023, Loss of ICS Power, reports that (2) on the Static Inverter Bypass Switch panel is not its normal position, and painters are in the area.

With these conditions, the Turbine Bypass Valves are operable in (1).

- A
 - 1) SW #2
 - 2) AUTO only
- B
 - 1) SW #3
 - 2) AUTO only
- C
 - 1) SW #2
 - 2) AUTO or HAND
- D
 - 1) SW #3
 - 2) AUTO or HAND

Proposed Answer: D

K/A Match Analysis

Requires knowledge of the breaker scheme for the Essential Inverters and the effects on Turbine Bypass Valves due to loss of an ICS inverter.

Answer Choice Analysis

- N. Incorrect. First part incorrect: SW #2 is OPEN in the normal alignment. Plausible because SW #2 CLOSED means the inverter is bypassed, so opening it **would** cause loss of the KU power panel. (ICS Hand power is from KU.) Second part is incorrect: TBVs are operable in AUTO **or** HAND (ref. 1SA-02 B-12, ICS HAND POWER FAILURE).
- O. Incorrect. First part is correct; SW #3 is normally CLOSED, supplying power from the inverter to its power panel. Second part incorrect as discussed above.
- P. Incorrect. First part incorrect as discussed above. Second part is correct as discussed above.
- Q. **Correct.** SW #3 is normally closed, supply the KU bus from the KU inverter. ASA-02 B-12 states that TBVs are operable in AUTO or HAND.

Technical Reference(s):

OP-OC-EL-VPC Rev. 16b

Supporting References

OP-OC-EL-VPC

7.2.B.4.c)

3. (Obj. R6) Inverter Bypass Switches

- A. An enclosure that sits near the static inverter cabinet, and contains three breaker-type switches, SW-1, SW-2, and SW-3.
- B. The switches are arranged so that the inverter can be completely isolated from the AC panelboard and the Regulated Power Supply, while the Regulated Power Supply is connected directly to the AC panelboard.
- C. This arrangement allows for complete isolation of the inverter for maintenance.
- D. From the diagram of the inverter circuit (OC-EL-VPC-6), it can be seen that:
 - 1) Anytime SW-2 is closed, the AC panelboard is connected directly to the Regulated Power Supply. {SW-2 is normally OPEN}
 - 2) In order for the inverter automatic transfer action to function, both SW-1 and SW-3 must be closed.
 - 3) SW-3 must be closed to connect the output of the inverter to the AC panelboard.
 - 4) If both SW-2 and SW-3 are open, the AC panelboard will lose power.

7.2.D Essential Power Panelboards

1. (Obj. R7) KU Power Panelboard

- a) The KU Inverter and isolation transformer network supplies 120/240VAC to the KU Power Panelboard, which is located in the unit cable room.
- b) Separate breakers in the panelboard feed such loads as:
 - 1) ICS Hand power

OP/1/A/6101/002

Alarm Response Guide 1SA-02

B-12, ICS Hand Power Failure

1. Alarm Setpoint

- 1.1 Loss of voltage to ICS HAND Power (1KU Power).

2. Automatic Action

- 2.1 IF ANY ICS station is in HAND, the function of that station fails, which could create a system transient due to that system/component.
- 2.2 1RC-1...
- 2.3 IF ...
- 2.4 Turbine Bypass Valves are operable in AUTO or HAND.

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336

ID: EL070601

Points: 0.25

Given the following conditions on Unit 1:

- Reactor power = 100%
- 1KI Inverter is aligned normally

Which ONE of the following switches (or switch combinations), if opened, will cause a sustained loss of power to 1KI Power Panelboard?

- A. SW #1
- B. SW #2
- C. SW #3
- D. SW #1 and SW #2

Answer: C

Answer Explanation

- A. Incorrect. SW #1 interrupts power from KRA/KRB to the inverter. In the normal power supply alignment, the KI panelboard receives power from the inverter.
- B. Incorrect. SW #2 interrupts power from KRA/KRB to the KI panelboard. In the normal power supply alignment, the KI panelboard receives power from the inverter.
- C. **Correct: SW #3 does interrupt power from the inverter**
- D. Incorrect: SW #1 interrupts power from KRA/KRB to the inverter. SW #2 is normally open in the NORMAL alignment. SW#3 being open will interrupt all power to the panelboard.

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Question 64

Unit 1 plant conditions:

- Reactor power = 100%
- 50 gpd tube leak 1A Steam Generator for approximately 1 week
- An increase in activity is reported in Chemical Treatment Pond (CTP) #3

Which ONE of the following describes an event which would cause this increase and the action(s) required to mitigate this event?

- A 1RIA-31 (LPI Cooler) activity is increasing and this will increase activity levels in CTP #3. Isolate and repair the faulty cooler.
- B 1RIA-33 (LW Release) interlock has failed and a Waste Monitor Tank release continues from the Radwaste Building. Stop the Waste Monitor Tank release.
- C 1RIA-42 (RCW) activity is increasing and this will increase activity levels in CTP #3. Isolate and repair the faulty cooler.
- D 1RIA-54 (TBS) interlock has failed and the Turbine Building Sump is being continually pumped. Open and White Tag 1A and 1B TBS Pump breakers.

Proposed Answer: D

K/A Match Analysis

Requires knowledge that failure of the 1RIA-54 (Turbine Building Sump monitor) automatic isolation function will require manual action to terminate a liquid radwaste release.

Answer Choice Analysis

- A. Incorrect. LPSW goes to the discharge not to #3 CTP. Isolating the cooler would not stop the release.
- B. Incorrect. Waste monitor tanks discharge to the Keowee tailrace, not CTP # 3. Stopping the waste monitor tank release would not stop the release to CTP # 3.
- C. Incorrect. RCW is a closed system. The RCW cooler is cooled by CCW which goes to the discharge, not CTP #3. Isolating the cooler would not stop the release.
- D. Correct. TBS pumps to CTP #3. Due to the S/G tube leak, activity could be high in the sump. If the interlock failed it could pump high activity to CTP #3. AP/18 requires that the 1A and 1B TBS pump breakers be opened and white-tagged.

Technical Reference(s): AP/1/A/1700/18 Rev. 22

(Attach if not previously provided)

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(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: Lesson Plan RAD-RIA, Rev. 13a, R2

Question Source: Bank # 59
Modified Bank # _ (Note changes or attach parent)
New _

Question History: Last NRC Exam: 2007-301

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

10 CFR Part 55 Content: 55.41 X
55.43 _

Comments:

068A3.02 (Liquid Radwaste System) Ability to monitor automatic operation of the Liquid Radwaste System including automatic isolation. 3.6/3.6

Author: Mike Donithan

Oconee Nuclear Station 2013-301 Written Examination

Supporting References

AP/1/A/1700/018

2. Automatic Systems Actions

MONITOR	DESCRIPTION	AUTOMATIC FUNCTIONS
RIA-31	6 Pt LPSW from LPI Cooler Disch	<ul style="list-style-type: none"> • Cams on the point currently sampling when a high radiation alarm is received
1RIA-45	Norm Vent Gas	<ul style="list-style-type: none"> • Stops RB Purge Fan and Mini-Fan
1RIA-46	Vent Gas HR	<ul style="list-style-type: none"> • Isolates RB Purge System
1RIA-49	RB Gas	<ul style="list-style-type: none"> • Sounds RB Evacuation Alarm
1RIA-49A	RB Gas HR	<ul style="list-style-type: none"> • Isolates RB Normal Sump
1RIA-54	TBS	<ul style="list-style-type: none"> • Terminates release from Unit 1 and 2 TBS

Section 4J, p. 1 of 5

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
NOTE	
1RIA-54 High alarm should automatically terminate release from Unit 1 and 2 TBS.	
1. <u> </u> IAAT 1RIA-54 is in High alarm, THEN GO TO Step 2.	<u> </u> GO TO Step 14.
NOTE	
The white tags can be created and hung after the TBS pump breakers are opened.	
2. Dispatch an operator to open and white tag: 1XD-R3C (1A TURB BLDG SUMP PUMP BKR) 1XE-R3D (1B TURB BLDG SUMP PUMP BKR)	

OP-OC-RAD-RIA, Rev. 13a

- L. 1, 3RIA-54 - Monitors turbine building sump activity - (NaI)(CPM)**
2. Sample pump provides continuous flow. (If RIA-54 sample pump is found off, notify Radwaste Chemistry of a possible unmonitored release.)
 3. Interlock to trip TBS Pumps if:
 - a) High Activity in the TBS
 - b) Loss of power to RIA
 - c) Actuates 1SA-18/C5 RM TBS Interlock
 - d) The sample pump stops (power failure or clogged strainer)
 - .
 - .
 - .
 - I. Units 1 & 2 have a common system while Unit 3 has a separate system.

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Question 65

Given the following plant conditions:

- ALL three (3) Oconee units at 100% power
- The OSM has receive notification of Condition A for the Intake Canal
- AP/13, Dam Failure, has been entered
- CCW-8, Emergency CCW Discharge To Tailrace, has been CLOSED

CCW-8 should be deenergized by opening its supply breaker within (1).

This is done to ensure it (2).

Which ONE of the following completes the above statements?

- A 1 1 hour
 2 will not open if submerged
- B 1 4 hours
 2 will not open if submerged
- C 1. 1 hour
 2. is safe to manually operate in subsequent steps
- D 1. 4 hours
 2. is safe to manually operate in subsequent steps

Proposed Answer: A

K/A Match Analysis

Loss of the intake structure could impact the circulating water system by submerging CCW-8. AP/13 states that CCW-8 must be deenergized prior to being submerged; the lesson plan states that this is to prevent it from spuriously opening. If it were to open it would change or defeat the procedural accident mitigation strategy.

Answer Choice Analysis

- A. CORRECT.
- B. INCORRECT. First part is incorrect but plausible because 4 hours is a time limit mentioned in AP/13 (in a Caution discussing cross-connection of CCW intake and discharge piping). Second part correct.
- C. INCORRECT. First part correct. Second part incorrect but plausible because general practice is to deenergize an MOV prior to manual operation, and if it had been wetted this would be of even more concern. AP/13 never directs manual operation of CCW-8 after its breaker is opened.

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D. INCORRECT. Both parts incorrect as discussed above.

Technical Reference(s): Lesson Plan OP-OC-STG-CCW, Rev 19a
(Attach if not previously provided) Lesson Plan OP-OC-EAP-TCA, Rev 05b
(including version/revision number) AP/1/A/1700/013, Rev 30

Proposed references to be provided to applicants during examination: None

Learning Objective: Lesson Plan OP-OC-STG-CCW Obj. R13

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

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content: 55.41 X
55.43 —

Comments:

075K2.01 (Circulating Water System) Ability to (a) predict the impacts of the following malfunctions or operations on the CWS; and (b) based on those predictions use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of intake structure. 3.0/3.2

Author: Mike Donithan

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Supporting References

AP/1/A/1700/013, p. 5 of 35

NOTE

- CCW-8 must be de-energized prior to submersion by lake water. This should be accomplished within 1 hour of initiation of the event.
- CCW Emergency Discharge Siphon Flow may have been established automatically as a result of loss of power.

OP-OC-STG-CCW

I.1 (Obj. R13) Loss of Intake Canal

- a) AP/1,2,3/A/1700/013, Dam Failure, provides direction for placing the Oconee units in a safe operating condition and actions to help mitigate the effects of a loss of the Intake Canal.
- b) The reactors are tripped to place the units in a safe shutdown condition.
- c) RCPs are secured to reduce primary heat load.
- d) The number of operating CCW Pumps is reduced to one in order to conserve inventory loss.
- e) CCW-8 must be de-energized prior to submersion by lake water. This should be done within 1 hour of the initiation of the event.

OP-OC-EAP-TCA (Time Critical Actions)

2.23 TCA #23 – Remove power from closed valve CCW-8 and Cross-connect (un-dewatered) CCW supply and discharge headers on all three units to supply station ASW pump during a dam failure or loss of intake canal.

A. Required Action: During a dam failure or loss of intake canal:

2. Remove power from closed valve CCW-8 during a dam failure or loss of intake canal.

Cross-connect un-dewatered CCW supply and discharge headers on all three units to supply station ASW pump during a dam failure or loss of intake canal.

I.2 Time to Complete Action:

1. Remove power from closed valve CCW-8 within 1 hour of event
2. Cross-connect un-dewatered CCW supply and discharge headers on all three units to supply station ASW pump within 4 hours of reactor trip

I.6 Comments:

- A. Removing power from closed valve CCW-8 within 1 hour of event mitigation will prevent the valve from opening as a result of it being submerged as the water level rises downstream of the Keowee dam.

Oconee Nuclear Station 2013-301 Written Examination

Question # 66

Given the following Unit 1 conditions:

Initial conditions:

- Reactor power = 100%
- BOTH Main Feedwater Pumps trip

Current conditions:

- Reactor power = 57% slowly decreasing

- 1) The correct sequence of activities directed by Rule 1 (ATWS) is to __ (1) __.
- 2) The direction given to the operator opening the CRD breaker is to __ (2) __ Arc Flash PPE.

Which ONE of the following completes the statements above?

- A. (1) align HPI injection from the BWST THEN dispatch an operator to open the CRD breakers
(2) wear
- B. (1) align HPI injection from the BWST THEN dispatch an operator to open the CRD breakers
(2) NOT wear
- C. (1) dispatch an operator to open the CRD breakers THEN align HPI injection from the BWST
(2) wear
- D. (1) dispatch an operator to open the CRD breakers THEN align HPI injection from the BWST
(2) NOT wear

Proposed Answer: B

Oconee Nuclear Station 2013-301 Written Examination

Explanation (Optional):

- A. Incorrect. First part is correct. HPI is aligned prior to dispatching an operator to open the CRD breakers. Second part is incorrect and plausible. The normal expectation is to wear Arc Flash PPE when operating a 600V breaker. Without the specific direction NOT to wear the PPE the outside operator may take unnecessary time to don this PPE.
- B. Correct. First part is correct. HPI is aligned prior to dispatching an operator to open the CRD breakers. Second part is correct. Rule 1 does have the control room operator direct the outside operator NOT to wear Arc Flash PPE.
- C. Incorrect. First part is incorrect and plausible. Opening the CRD breakers is an action directed by Rule 1 with the intent of remotely tripping the reactor. It is reasonable for the candidate to conclude the highest priority is to accomplish the reactor trip. Since opening the CRD breakers is done outside the control room and takes several minutes to accomplish it would be consistent with getting the reactor tipped to go ahead and get someone dispatched to open the breakers prior to aligning HPI injection. Second part is incorrect and plausible. The normal expectation is to wear Arc Flash PPE when operating a 600V breaker. Without the specific direction NOT to wear the PPE the outside operator may take unnecessary time to don this PPE.
- D. Incorrect. First part is incorrect and plausible. Opening the CRD breakers is an action directed by Rule 1 with the intent of remotely tripping the reactor. It is reasonable for the candidate to conclude the highest priority is to accomplish the reactor trip. Since opening the CRD breakers is done outside the control room and takes several minutes to accomplish it would be consistent with getting the reactor tipped to go ahead and get someone dispatched to open the breakers prior to aligning HPI injection. Second part is correct. Rule 1 does have the control room operator direct the outside operator NOT to wear Arc Flash PPE.

Technical Reference(s): EAP-UNPP
 EOP Rule 1

(Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: OP-OC-PNS-HPI Obj. R4

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

Question History: Last NRC Exam 2011-301_Q 67

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
Dispatch <u>one</u> operator without wearing Arc Flash PPE to open 600V CRD breakers: {33} ___ 1X9-5C (U-1 CRD Norm Fdr Bkr) (U1 Equipment Rm) ___ 2X1-5B (U-1 CRD Alternate Fdr Bkr) (T-3/Dd-28)	C. ___ Open 1HP-409.

Oconee Nuclear Station 2013-301 Written Examination

Question 67

Proposed Question:

A normal plant startup is in progress with the following operations scheduled to occur:

- 0800 the Turning Gear Oil Pump (TGOP) is started
- 0900 an HPIP is started in the process of swapping operating pumps.

Per OMP 1-02, RULES OF PRACTICE, __(1)__ is the earliest time that an announcement is made for starting the above pumps and if a safety barrier is in place, confirmation that there are no personnel in the immediate vicinity of the motor __(2)__ required.

- A. (1) 0759
(2) is
- B. (1) 0759
(2) is not
- C. (1) 0859
(2) is
- D. (1) 0859
(2) is not

Proposed Answer: D

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Explanation (Optional):

- A. 1st part is incorrect but plausible because the TGOP is a large electrical load. 2nd part is incorrect but plausible because personnel are allowed within the safety barriers with permission.
- B. 1st part is incorrect but plausible because the TGOP is a large electrical load. 2nd part is correct.
- C. 1st part is correct. 2nd part is incorrect but plausible because personnel are allowed within the safety barriers with permission.
- D. Correct.

Technical Reference(s): OMP1-02 Rules of Practice
OP-OC-ADM-OMP

(Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: OP-OC-ADM-OMP Obj. R57 & 28

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

Question History: Last NRC Exam
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 X
55.43

Comments:

G2.1.14 Knowledge of criteria or conditions that require plant-wide announcements, such as pump starts, reactor trips, mode changes, etc..

(CFR 41.10 / 43.5 / 45.12)

3.1/3.1

Author: Ken Schaaf

5.13 Starting 6900V/4160V Motors or Returning Main Buslines to Service:

5.13.1 A safety barrier and sign to limit access of station personnel may surround the 6900V/4160V electrical switchgear. These areas require WCC SRO permission prior to entry. On-duty NEOs may enter these areas, as needed, without specific WCC SRO permission to perform equipment inspections or other actions as directed. A temporary safety barrier with signs limiting access may be established around 4160V pumps that will have multiple starts during test procedures. The safety barrier and signs shall be removed when the test is complete.

5.13.2 The protocol for starting reactor coolant pumps (RCP), 4160 volt motors, and operating PCBs except for unanticipated automatic starts and emergency situations is as follows:

A. Prior to starting an RCP:

- When the Reactor Building is open for personnel entry, obtain confirmation that the affected SG cavity has been cleared of unnecessary personnel. This confirmation can be provided by RP or by two NEOs dispatched to clear the affected SG cavity.
- If no safety barriers are in place, confirm that the area immediately surrounding the 6900 V breaker has been cleared of personnel.
- Approximately one (1) minute prior to starting, announce the RCP start via the plant page.

B. Prior to starting a 4160V motor:

- Obtain confirmation that the area immediately surrounding the 4160V motor (H 10-foot radius) has been cleared of personnel, or that a temporary safety barrier and signs are in place.
- If no safety barriers are in place, confirm that the area immediately surrounding the 4160 V breaker has been cleared of personnel.
- Approximately one (1) minute prior to starting, announce the motor start via the plant page.

1. Turning Gear Oil Pump (TGOP)
 - a) During the time the MAIN TURBINE is on turning gear, coasting down, or coming up to speed, operating oil pressure from the Main Shaft Oil Pump is inadequate to supply the bearing header. Since the Motor Suction pump does not supply the bearing oil header, the TGOP is used to supply bearing oil during these modes of operation.
 - b) **To supply the MAIN TURBINE lubrication needs and oil lift pumps suction while on turning gear and while the MAIN TURBINE is coming up to speed, the Turning Gear Oil Pump is provided.**
 - c) Like the MSP, the TGOP is a 600 VAC motor-driven pump, powered from MCC XA. The pump casing, too, is submerged in the MTOT and connects to its 50 HP motor via a flexible coupling and extended shaft. The TGOP also takes suction, via a strainer, directly on the MTOT, near the bottom of the tank.
 - d) The discharge of the TGOP is through a discharge check valve, into the in-service MAIN TURBINE oil cooler, and on to the bearing header. The TGOP will produce 15-20 psi (**GE Value**) bearing header pressure at the turbine centerline.
 - e) During MAIN TURBINE startup, the TGOP has already been started since the turbine is on turning gear – the TGOP must be on for turning gear operation, to supply bearing header oil pressure greater than 10 psi, since the turning gear motor is interlocked off at 10 psi and less.

I.7 Power Supplies For HPI System ES Components

A. High Pressure Injection Pumps

1. "A" HPI pump – TC or PSW B6T switchgear
2. "B" HPI pump – TE or PSW B6T switchgear
3. "C" HPI pump – TD

Oconee Nuclear Station 2013-301 Written Examination

Question 68

Unit 1 initial conditions:

- Reactor power = 100%
- LDST level = 75" stable
- Group 7 rod position = 94% withdrawn
- Makeup to LDST initiated from 1A BHUT
- Neutron error = 0 stable

Current conditions:

- 1HP-15 Bailey controller indicates 470 gallons added to LDST
- 1A Bleed Transfer Pump secured

Based on the above conditions, which ONE of the following would describe a diverse indication that 470 gallons of 1A BHUT had been added to the LDST?

LDST level is approximately _____ inches and neutron error will become _____.

- A. 90 / positive
- B. 90 / negative
- C. 95 / positive
- D. 95 / negative

Proposed Answer: A

Explanation (Optional):

K/A MATCH ANALYSIS

Requires analyzing various diverse indications to validate a change in LDST level and neutron error following a batch addition of boric acid from 1A BHUT.

ANSWER CHOICE ANALYSIS

Answer: A

A. CORRECT: LDST is 31.3 gal/inch. If 470 gallons of water had been added then level should have increased 15 inches which would put level at 90 inches. If A bleed had been added then Boron concentration would be increasing which means neutron error would be building positive towards a rod pull to offset the boron addition.

B. Incorrect. First part is correct. Second part is plausible since it would be correct if the addition was from B BHUT OR if the student made the common error of incorrectly determining the direction of rod motion required to offset the boron addition.

C. Incorrect: First part is plausible because this would be correct if you use 24 gal/inch for LDST volume however this is the Pressurizer value. Second part is correct.

D. Incorrect: First part is plausible because this would be correct if you use 24 gal/inch for LDST volume however this is the Pressurizer value. Second part is plausible since it would be correct if the addition was from B BHUT OR if the student made the common error of incorrectly determining the direction of rod motion required to offset the boron addition.

Technical Reference(s): OP-CP-016 Lesson Plan Rev. 08
(Attach if not previously provided) PNS-PZR Lesson Plan Rev. 21
(including version/revision number) PNS-HPI Lesson Plan Rev. 28b
STG-ICS Lesson Plan Rev. 7d

Proposed references to be provided to applicants during examination: None

Learning Objective: OP-CP-016 Lesson Plan Obj. R5, R12
PNS-PZR Lesson Plan Obj. R1, R2, R3

Question Source: Bank # _____
Modified Bank # 2010A Q. #67 (attach parent)
New _____

Question History: Last NRC Exam _____
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 X
55.43 _____

Comments:

2.1.45, Conduct of Operations
Ability to identify and interpret diverse indications to validate the response of another indication.
(4.3/4.3)

1 POINT

Question 67

Unit 1 initial conditions:

- Reactor power = 100%
- LDST level = 75" stable
- Group 7 rod position = 94% withdrawn
- Makeup to LDST initiated from 1B BHUT
- Neutron error = 0 stable

Current conditions:

- 1HP-15 Bailey controller indicates 470 gallons added to LDST
- 1B Bleed Transfer Pump secured

Based on the above conditions, which ONE of the following would describe a diverse indication that 470 gallons of 1B BHUT had been added to the LDST?

LDST level is approximately _____ inches and neutron error will become _____.

- A. 90 / positive
- B. 90 / negative
- C. 95 / positive
- D. 95 / negative

2010A NRC REACTOR OPERATOR EXAM

Question 67

T3 – CPW

2.1.45, Conduct of Operations

Ability to identify and interpret diverse indications to validate the response of another indication.

(4.3/4.3)

K/A MATCH ANALYSIS

Requires analyzing various diverse indications to validate a SG tube leak exists.

ANSWER CHOICE ANALYSIS

Answer: B

A. Incorrect. First part is correct. Second part is plausible since it would be correct if the addition was from A BHUT OR if the student made the common error of incorrectly determining the direction of rod motion required to offset the boron addition.

B. CORRECT: LDST is 31.3 gal/inch. If 470 gallons of water had been added then level should have increased 15 inches which would put level at 90 inches. If B bleed had been added then Boron concentration would be decreasing which means neutron error would be building negative towards a rod push to offset the boron addition.

C. Incorrect: First part is plausible because this would be correct if you use 24 gal/inch for LDST volume however this is the Pressurizer value. Second part is plausible since it would be correct if the addition was from A BHUT OR if the student made the common error of incorrectly determining the direction of rod motion required to offset the boron addition.

D. Incorrect: First part is plausible because this would be correct if you use 24 gal/inch for LDST volume however this is the Pressurizer value. Second part is correct.

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Technical Reference(s): **PNS-PZR, PNS-HPI, CP-016**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **PNS-PZR R1,2,3 CP-016 R5,**

Question Source: **NEW**

Question History: Last NRC Exam **N/A**

Question Cognitive Level: **Comprehension and Analysis**

Oconee Nuclear Station 2013-301 Written Examination

Question 69

Proposed Question:

Unit 1 plant conditions:

- Unit startup in progress
- Turbine Generator startup is in progress using OP/1/A/1106/001 TURBINE GENERATOR, Enclosure 4.1 TURBINE GENERATOR STARTUP

Selecting the "Turbine Load" pushbutton sends a signal to _____.(1)_____.
This evolution is performed _____.(2)_____.

- A. (1) close the Turbine Bypass Valves which in turn, causes the Turbine Control Valves to open to maintain steam pressure constant
(2) prior to electrically loading the Main Generator
- B. (1) close the Turbine Bypass Valves which in turn, causes the Turbine Control Valves to open to maintain steam pressure constant
(2) after electrically loading the Main Generator
- C. (1) open the Turbine Control Valves which in turn, causes the Turbine Bypass Valves to close to maintain steam pressure constant
(2) prior to electrically loading the Main Generator
- D. (1) open the Turbine Control Valves which in turn, causes the Turbine Bypass Valves to close to maintain steam pressure constant
(2) after electrically loading the Main Generator

Proposed Answer: D

Oconee Nuclear Station 2013-301 Written Examination

Explanation (Optional):

- A. 1st part is incorrect but plausible because the TBVs do close in response to the TCVs opening. 2nd part is incorrect but plausible because it is a common error to assume that the turbine "Load" pushbutton is depressed in preparation for electrical loading of the turbine.
- B. 1st part is incorrect but plausible because the TBVs do close in response to the TCVs opening. 2nd part is correct.
- C. 1st part is correct. 2nd part is incorrect but plausible because it is a common error to assume that the turbine "Load" pushbutton is depressed in preparation for electrical loading of the turbine.
- D. Correct.

Technical Reference(s): OP/1/A/1106/001 TURBINE GENERATOR, Enclosure 4.1
 TURBINE GENERATOR STARTUP Rev 123
 OP-OC-STG-ICS Ch 3

(Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: OP-OC-STG-ICS Obj. R11

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

Question History: Last NRC Exam _____
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 X
 55.43

Comments:

G2.2.2 Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels.
(CFR 41.6 / 41.7 / 45.2)
4.6/4.1

Author: Ken Schaaf

Enclosure 4.1
Turbine Generator Startup

2.64 Load Turbine Generator:

2.64.1 Ensure Generator Output H 35 MWe.

2.64.2 Ensure TURBINE AUTO LOAD PERMISSIVE satisfied.

2.64.3 Ensure TURBINE MASTER control station to "AUTO".

NOTE:

- Steps 2.64.4 through 2.64.7 may be repeated as required.
- De-selecting the TURBINE LOAD pushbutton will result in stoppage of opening the Turbine Control Valves any further. Likewise, de-selecting the TURBINE UNLOAD pushbutton will also result in stoppage of closing the Turbine Control Valves any further.

2.64.4 **IF AT ANY TIME** it is desired to stop loading the turbine, de-select the "TURBINE LOAD" Pushbutton.

2.64.5 **IF AT ANY TIME** it is desired to stop unloading the turbine, de-select the "TURBINE UNLOAD" Pushbutton.

2.64.6 **IF AT ANY TIME** it is desired to unload the Turbine Generator, perform the following:

- A. Select TURBINE UNLOAD button.
- B. Ensure TBVs throttle open
- C. Ensure Turbine Generator MWe decrease

2.64.7 Select "TURBINE LOAD" Pushbutton.

_____/_____
Date Time

2.64.8 Verify TURBINE BYPASS VALVES close.

2.64.9 Verify "TURBINE LOAD" pushbutton lamp extinguishes.

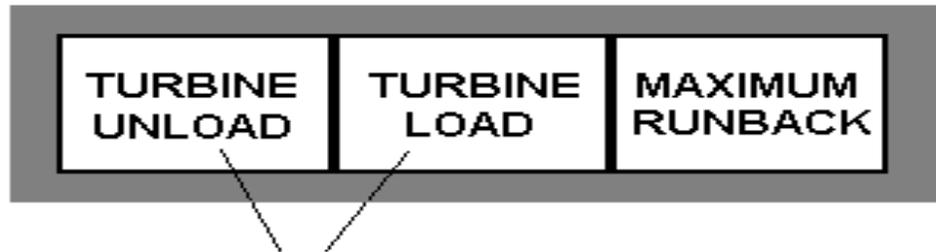
2.64.10 Ensure Generator Output ϵ 90 MWe.

Automatic Turbine Loading

The turbine Load and Unload system enables the operator to smoothly introduce and remove the main turbine into the plant control process.

This feature provides a smooth transition of steam pressure control from the TBVs to the Turbine or vice versa.

Two back-lit buttons are provided on the LCP, TURBINE LOAD and TURBINE UNLOAD.



Each button illuminates when the function is active.

These buttons, along with additional logic, control a separate Turbine Load status flag that controls MWe tracking when the turbine is in manual.

Turbine Load status flag is visible to the operator on the OAC at Point ID X2060, "TURBINE LOADING STATUS" (True/False).

The Turbine Load status flag exists in the ICS to signal to the system that MWe tracking is allowed, and to signal to the bypass valves to add the normal operation 50 psi bias.

The Turbine Loading status is controlled by the operator by the TURBINE LOAD and TURBINE UNLOAD pushbuttons on the LCP.

Certain CTPD demand conditions will also automatically set the state of this flag.

If the turbine is in manual and the Turbine Load status flag is true, the CTP Demand will track MWe.

If the Turbine Load status flag is not true, CTP Demand will be set either by the operator (target load on the LCP) or by one of the other tracking signals.

The Turbine Master can be in automatic OR in manual and the status flag be false.

The Turbine Load status flag is always true above 22% CTPD (unless BOTH Generator breakers are OPEN).

During a load rejection when the Generator breakers open, the status flag immediately turns false to help lessen the THP transient by removing the TBV bias allowing the valves to control THP at setpoint.

With both Generator breakers open, the TRICON system reverts to speed control and therefore the Turbine Master reverts to Manual. NO Tracking would occur because the status flag is false.

The turbine LOAD status flag is always false below 10% CTPD. This ensures that turbine will not force the reactor to severe subcriticality if tracking MWe at low power.

If the Turbine Load status flag is false, the bypass valves and Main Turbine have exactly the same control set point (unless the reactor is tripped).

When the Turbine Load status flag is true, the TBVs have a set point 50 psig greater than the turbine.

When the turbine is in automatic and CTPD is greater than 10%, pressing the TURBINE LOAD button will initiate automatic turbine loading.

This function causes steam flow to be smoothly transferred from the TBVs to the main turbine.

When both TBV demands are less than 0%, the Turbine Load status flag will be set "true", and the normal 50 psi bias is applied to the bypass valves.

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Question 70

Unit 2 plant conditions:

- Reactor startup in progress
- Control Rod Groups 1-4 are fully withdrawn
- Control Rod Group 5 is fully inserted
- A group 2 safety rod drops fully into the core and cannot be moved
- SDM has been determined NOT to be within the limit specified in the COLR

Based on the above conditions:

- (1) Is entry into TS 3.1.5, SAFETY ROD POSITION LIMITS, required?
 - (2) What is the maximum time allowed to initiate boration to restore SDM to within the limit stated in the COLR?
- A. (1) yes
(2) 1 hour
- B. (1) yes
(2) 15 minutes
- C. (1) no
(2) 1 hour
- D. (1) no
(2) 15 minutes

Proposed Answer: D

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Explanation (Optional):

- A. 1st part is incorrect but plausible because as soon as Group 5 rod bottom lights are cleared during withdrawal, the rod withdrawal is stopped and Mode 2 is declared which would make TS 3.1.5 applicable. 2nd part is incorrect but plausible because this is the time requirement in TS 3.1.5 which would be applicable during Modes 1 & 2.
- B. 1st part is incorrect but plausible (see A). 2nd part is correct.
- C. 1st part is correct because the plant is still in mode 3 therefore, TS 3.1.5 is not applicable. 2nd part is incorrect (see A).
- D. Correct.

Technical Reference(s): TS 3.1.5 Safety Rod Position Limits
TS 3.1.1 Shutdown Margin (SDM)

(Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: OP-OC-ADM-ITS Obj. R7

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

Question History: Last NRC Exam _____
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 X
55.43

Comments:

G2.2.39 Knowledge of less than or equal to one hour Technical Specification action statements for systems.

(CFR 41.7 / 41.10 / 43.2 / 45.13)
3.9/4.5

Author: Ken Schaaf

Oconee Nuclear Station 2013-301 Written Examination

3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 SHUTDOWN MARGIN (SDM)

LCO 3.1.1 The SDM shall be within the limit specified in the COLR.

APPLICABILITY: MODES 3, 4, and 5.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limit.	A.1 Initiate boration to restore SDM to within limit.	15 minutes

Oconee Nuclear Station 2013-301 Written Examination

3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Safety Rod Position Limits

LCO 3.1.5 Each safety rod shall be fully withdrawn.

-----NOTE-----
 Not required for any safety rod positioned to perform SR 3.1.4.2.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One safety rod not fully withdrawn.	A.1 Withdraw the rod fully.	1 hour
	<u>OR</u>	
	A.2.1.1 Verify SDM is within the limit specified in the COLR.	1 hour
	<u>OR</u>	
	A.2.1.2 Initiate boration to restore SDM to within limit.	1 hour
	<u>AND</u>	
	A.2.2 Declare the rod inoperable.	1 hour

Enclosure 4.7

OP/1/A/1102/001

Unit Startup From 532°F/2155 psig
To MODE 1

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3.15 IF PT/0/A/0711/001 (ZPPT) is **NOT** in progress, perform the following:

- Ensure SEQUENCE is selected on Diamond. (R.M.)
- Ensure "LO SET" selected for NI Recorder per OP/0/A/1108/001 (Curves and General Information).

3.16 Begin withdrawing Regulating CRDs per OP/1/A/1105/019 (Control Rod Drive System). (R.M.)

3.17 **WHEN** Group 5 rods are off IN LIMIT, stop Group 5 rod withdrawal.

3.18 Perform the following:

- Ensure MODE 2 selected on OAC.
- Ensure MODE 2 selected for Unit 1 in TSAIL.
- Announce on Plant Page "Unit 1 has entered MODE 2".
- Notify Assistant Outage Manager of Unit 1 entry into MODE 2.

_____/_____
Person Notified Date Time

Oconee Nuclear Station 2013-301 Written Examination

Question # 71

Unit 1 plant conditions:
Reactor power = 100%

Based on the above condition, which ONE of the following describes a condition that would require entry into a Tech Spec ACTIONS table?

- A UST level = 6.5 feet
- B BWST level = 47.3 feet
- C CFT-1 pressure = 630 psig
- D BWST temperature = 101.5 F

Proposed Answer: C

K/A Match Analysis

The question requires the applicant to recognize system parameters that are entry-level conditions for Technical Specifications. (CFR: 41.7 / 41.10 / 43.2 / 43.3 / 45.3)

Answer Choice Analysis

Answer A: Incorrect: Plausible since there is a minimum required TS level for UST to support EFDW operability. TS 3.7.6 requires that both the UST and Hotwell be operable and that the UST contain > 30,000 gallons. PT/600/01 (Periodic Instrument Surveillance) verifies this volume by requiring UST level be > 6 feet.

Answer B: Incorrect: Plausible since there is a minimum required TS level for BWST to support operability. TS 3.5.4 requires the BWST to be operable and contain 350,000 gallons of Borated water. PT/600/01 (Periodic Instrument Surveillance) verifies this volume by requiring >47 feet in the BWST.

Answer C: Correct: Tech Spec 3.5.1 requires Core Flood Tank pressure between 575 psig and 625 psig to maintain Operability.

Answer D: Incorrect: Plausible since there is a maximum allowed TS BWST temp to support operability. TS 3.5.4 requires the BWST to be operable with a temperature range of 52.5 to 102.5 F, inclusive. PT/600/01 (Periodic Instrument Surveillance) verifies this temperature.

Technical Reference(s):

Lesson Plan ADM-ITS, Rev 05d, pages 26

Oconee Nuclear Station 2013-301 Written Examination

(Attach if not previously provided) _____
(Including version/revision number) _____

Proposed references to be provided to applicants during examination: None

Learning Objective: Lesson Plan ADM-ITS, Rev 05d Obj. R8

Question Source: Bank # _____
Modified Bank # 2009 ONS RO NRC, Q71 (attached)
New _____

Modified from 2009 ONS RO NRC, Q71. Original question had different distracters for answers C and D. Modified previous correct answer to an incorrect value and created new correct answer.

Question History: Last NRC Exam: 2009 ONS RO NRC, Q71

Question Cognitive Level: Memory or Fundamental Knowledge X_
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X
55.43 _

Comments:

G2.2.42: Ability to recognize system parameters that are entry-level conditions for Technical Specifications. (CFR: 41.7 / 41.10 / 43.2 / 43.3 / 45.3). The original question plus 2 additional questions found searching for G2.2.42 from the combined bank questions provided by ONS are attached below

Author: Newton Lacy

Oconee Nuclear Station 2013-301 Written Examination

Question 72

Which ONE of the following Area Radiation Monitors will sound a LOCAL alarm (do NOT include any associated Statalarms) to indicate increased radiation levels?

- A 1RIA-1 (Control Room Monitor)
- B 1RIA-15 (High Pressure Injection Pump Room Monitor)
- C 1RIA-17 ("B" Main Steam Line Monitor)
- D 1RIA-56 (High Range Stack Monitor)

Proposed Answer: B

K/A Match Analysis

GEN2.3 2.3.5 - GENERIC - Radiation Control; ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc. (CFR: 41.11 / 41.12 / 43.4 / 45.9).

Question requires ability to use a fixed radiation monitor and its alarm. The ability to determine whether or not a local alarm would sound to alert you of increasing radiation levels would be required as part of the ability to use effectively use the radiation monitor.

Answer Choice Analysis

Answer A: Incorrect: Plausible since this is an area RIA monitor and most Area monitors provide a local alarm.

Answer B: Correct. 1RIA-15 has a local alarm.

Answer C: Incorrect. Plausible since this is an area RIA monitor and most Area monitors provide a local alarm.

Answer D: Incorrect. Plausible since this is an area RIA monitor and most Area monitors provide a local alarm.

Technical Reference(s): Lesson Plan RAD-RIA, Rev 13a, pages 26-30
(Attach if not previously provided) _____
(Including version/revision number) _____

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Proposed references to be provided to applicants during examination: None

Learning Objective: Lesson Plan ADM-ITS, Rev 13a, Terminal Obj. R2

Question Source: Bank # 2011B ONS SRO NRC Examination Q72 B

Modified Bank # _____

New _____

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X

55.43 _

Comments:

G 2.3.5 - GENERIC - Radiation Control; ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc. (CFR: 41.11 / 41.12 / 43.4 / 45.9). The original question plus 1 additional question (SRO) found searching for G2.3.5 from the combined bank questions provided by ONS are attached below

Author: Newton Lacy

Oconee Nuclear Station 2013-301 Written Examination

Question # 73

Given the following Unit 3 conditions:

- 3A GWD gas tank release in progress
- Release is at 2/3 Station Limit

1) 1RIA-45 High and Alert setpoints will be set at __ (1) __ the normal 1/3 Station Limit as listed in PT/0/A/230/001 (Radiation Monitor Check).

2) If 1RIA-45 High alarm setpoint is reached, the 3A GWD gas tank release __ (2) __. Which ONE of the following completes the statements above?

- A 1. double
 2. will automatically terminate
- B 1. double
 2. must be manually terminated
- C 1. half
 2. will automatically terminate
- D 1. half
 2. must be manually terminated

Proposed Answer: D

K/A Match Analysis

G2.3.11 - GENERIC - Radiation Control Radiation Control. Ability to control radiation releases. (CFR: 41.11 / 43.4 / 45.10)

Question requires knowledge of the process for a waste gas release at 2/3 the station limit.

Answer Choice Analysis

Answer A: Incorrect.

First part is incorrect and plausible. Per PT/0/A/230/001 the non-releasing unit's RIA-45 setpoint is half that of the releasing unit's.

Second part is incorrect and plausible. The station release limit could be exceeded and the other unit's RIA-45 in high alarm. The release will be automatically terminated if the RIA-37 setpoint is exceeded on the releasing unit. Therefore it is reasonable to conclude a High alarm on the 1RIA-45 would trigger an automatic termination of the release.

Answer B: Incorrect.

First part is incorrect and plausible. Per PT/0/A/230/001 the non-releasing unit's RIA-45 setpoint is half that of the releasing unit's.

Second part is correct. Per OP/3/A/1104/018 (GWD System) if RIA-45 High alarm actuates on a non-releasing unit, the other unit must be notified to manually terminate the release. RIA-37/38 are the process monitors that are interlocked to terminate the release.

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Answer C: Incorrect.

First part is correct. Per PT/0/A/230/001 (Radiation Monitor Check) the setpoint on the non-releasing unit is set at half the value in the PT.

Second part is incorrect and plausible. The station release limit could be exceeded and the other unit's RIA-45 in high alarm. The release will be automatically terminated if the RIA-37 setpoint is exceeded on the releasing unit. Therefore it is reasonable to conclude a High alarm on the 1RIA-45 would trigger an automatic termination of the release.

Answer D: Correct.

First part is correct. Per PT/0/A/230/001 (Radiation Monitor Check) the setpoint on the non-releasing unit is set at half the value in the PT. Second part is correct. Per OP/3/A/1104/018 (GWD System) if RIA-45 High alarm actuates on a non-releasing unit, the other unit must be notified to manually terminate the release. RIA-37/38 are the process monitors that are interlocked to terminate the release.

Technical Reference(s): Lesson Plan WE-GWD, Rev 13a , pages 24-25

(Attach if not previously provided) _____

(Including version/revision number) _____

Proposed references to be provided to applicants during examination: None

Learning Objective: Lesson Plan WE-GWD, Rev 13a, Enabling Objectives R6, R15

Question Source: Bank # ILT39 ONS SRO NRC Examination Q 70

Modified Bank # _____

New _____

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis —

10 CFR Part 55 Content: 55.41 X

55.43 —

Comments:

G 2.3.5 - GENERIC - Radiation Control; ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring

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equipment, etc. (CFR: 41.11 / 41.12 / 43.4 / 45.9). The original question plus 1 additional question (SRO) found searching for G2.3.5 from the combined bank questions provided by ONS are attached below

Author: Newton Lacy

Oconee Nuclear Station 2013-301 Written Examination

Question 74

Given the following Unit 3 conditions:

Initial conditions:

- Reactor power = 100%

Current conditions:

- Chlorine gas is entering the Control Room due to an accidentally dropped cylinder.
- The SRO has implemented AP/3/A/1700/008 (Loss of Control Room).

- 1) The RO will go to the ____ (1) ____
- 2) Bank 2 Groups ____ (2) ____ Pzr heaters will be used to control RCS pressure from this location.

Which ONE of the following completes the statements above?

- A 1. Standby Shutdown Facility
 2. B and D
- B 1. Standby Shutdown Facility
 2. B and C
- C 1. Unit 3 Auxiliary Shutdown Panel
 2. B and D
- D 1. Unit 3 Auxiliary Shutdown Panel
 2. B and C

Proposed Answer: C

K/A Match Analysis

GEN 2.4.34 - GENERIC - Emergency Procedures / Plan. Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects. (CFR: 41.10 / 43.5 /45.13).

Question requires knowledge of RO action outside of the CR during an emergency and how RCS pressure will be controlled by that RO.

Answer Choice Analysis

Answer A: Incorrect. First part is incorrect and plausible. The Standby Shutdown Facility is used for a fire that results in the loss of the control room. Second part is correct. The ASDP uses PZR heater Bank 2 Groups B and D for RCS pressure control.

Answer B: Incorrect. First part is incorrect and plausible. The Standby Shutdown Facility is used for a fire that results in the loss of the control room. Second part is incorrect and plausible. PZR heater Group 2 Banks B and C are controlled from the SSF which would be used if the evacuation was due to a fire.

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Answer C: Correct. First part is correct. AP/008 directs going to the ASDP when evacuating the control room for any condition other than a fire. Second part is correct. The ASDP uses PZR heater Bank 2 Groups B and D for RCS pressure control.

Answer D: Incorrect. First part is correct. AP/008 directs going to the ASDP when evacuating the control room for any condition other than a fire. Second part is incorrect and plausible. PZR heater Group 2 Banks B and C are controlled from the SSF which would be used if the evacuation was due to a fire.

Technical Reference(s): Lesson Plan IC-ASP, Rev 10

Lesson Plan EAP-APG, Rev 7d (AP-8 enclosure Rev00)

Procedure AP/3/A/1700/008, Rev 14

(Attach if not previously provided) _____

(Including version/revision number) _____

Proposed references to be provided to applicants during examination: None

Learning Objective: Lesson Plan EAP-APG, Rev 7d, Enabling Objective R9

<u>Question Source:</u>	Bank #	<u>ILT39 ONS SRO NRC Examination Q 73</u>
	Modified Bank #	_____
	New	_____

Question History: Last NRC Exam: N/A

<u>Question Cognitive Level:</u>	Memory or Fundamental Knowledge	_
	Comprehension or Analysis	<u>X</u>

<u>10 CFR Part 55 Content:</u>	55.41	<u>X</u>
	55.43	_

Comments:

GEN2.4.34 - Emergency Procedures / Plan. Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects. (CFR: 41.10 / 43.5 /45.13)

Original question included at end of document.

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Author: Newton Lacy

Supporting Reference (from AP/3/A/170000/8, Loss of Control Room)

AP/3/A/1700/008
Page 1 of 15

1. Entry Conditions

Either of the following exists:

- **Any condition exists that requires evacuation of the Control Room while in MODE 1, 2, or 3** with RCS Temperature ≥ 520 °F.
- A fire requires evacuation of the Control Room while above decay heat removal operation.

AP/3/A/1700/008
Page 3 of 15

4. Subsequent Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4.1 <input type="checkbox"/> Verify the Control Room evacuation is due to a fire.	<input type="checkbox"/> GO TO Step 4.10.

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AP/3/A/1700/008
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ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4.10 <input type="checkbox"/> Verify actions can be taken to secure the plant from Control Room before evacuation.	<div style="border: 1px solid black; padding: 5px; text-align: center;">NOTE</div> Other APs, if needed, are available in the TSC. <ol style="list-style-type: none"> 1. <input type="checkbox"/> Proceed to Unit 3 Auxiliary Shutdown Panel with this AP and Unit 3 EOP. 2. <input type="checkbox"/> GO TO Step 4.17.
4.11 Perform the following: A. <input type="checkbox"/> Trip the Rx. B. <input type="checkbox"/> Perform EOP IMAs.	
4.12 <input type="checkbox"/> Emergency start both Keowee units.	
4.13 <input type="checkbox"/> Open 3HP-24.	
4.14 Make the following notifications: <input type="checkbox"/> PA announcement of the required evacuation of CR <input type="checkbox"/> OSM to reference the following: <ul style="list-style-type: none"> • RP/0/B/1000/001 (Emergency Classification) • OMP 1-14 (Notifications) 	
<div style="border: 1px solid black; padding: 5px;">NOTE</div> Other APs, if needed, are available in the TSC.	
4.15 <input type="checkbox"/> Proceed to Unit 3 Auxiliary Shutdown Panel with this AP and Unit 3 EOP.	
4.25 <input type="checkbox"/> Maintain RCS pressure 1800 - 2200 psig by cycling Pzr Heater Bank control switch as required.	

Supporting Reference (from Auxiliary Shutdown Panels (IC-ASP) Lesson Plan)

Enabling Objectives

2. Describe the effect on Control Room controls for Bank No. 2 PZR Heaters for each of the following Bank 2 control positions in the ASP: (R3)
 - a. OFF
 - b. NORMAL
 - c. ON

2.4 Detailed Description of Controls on ASP

Pressurizer Heater Control

Controls Bank No. 2 Groups B & D PZR Heaters

(OBJ R3) OFF - NORM - ON" Selector Switch

In "OFF", Bank 2 heaters Groups B & D are turned off and cannot be operated from the Control Room.

In "NORM", Bank 2 heaters Groups B & D are controlled from the Control Room, either automatically or manually.

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In "ON", Bank 2 heaters Groups B & D are turned on and cannot be turned off from the Control Room. They will still automatically de-energize on low PZR level of ≤ 80 ".

Included below is the original question from the ONS compiled exam reference provided to NRC

ILT39 ONS SRO NRC Examination QUESTION 73

Given the following Unit 3 conditions:

Initial conditions:

- Reactor power = 100%
-

Current conditions:

- Chlorine gas is entering the Control Room due to an accidentally dropped cylinder.
- The SRO has implemented AP/08 (Loss of Control Room).
-

1) The RO will go to the __ (1) __.

2) Bank 2 Groups __ (2) __ PZR heaters will be used to control RCS pressure from this location.

Which ONE of the following completes the statements above?

- A. Standby Shutdown Facility
B and D
- B. Standby Shutdown Facility
B and C
- C. Unit 3 Auxiliary Shutdown Panel
B and D
- D. Unit 3 Auxiliary Shutdown Panel
B and C

GEN2.4 2.4.34 - GENERIC - Emergency Procedures / Plan

Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects. (CFR: 41.10 / 43.5 /45.13)

General Discussion

Chlorine gas cylinders are stored on site per CP/0/B/4002/011 as part of the Chlorine feed system. Per this procedure a chlorine leak ≥ 0.5 ppm is reported to the control room.

Answer A Discussion

Incorrect.

First part is incorrect and plausible. The Standby Shutdown Facility is used for a fire that results in the loss of the control room.

Second part is correct. The ASDP uses PZR heater Bank 2 Groups B and D for RCS pressure control.

Answer B Discussion

Incorrect.

First part is incorrect and plausible. The Standby Shutdown Facility is used for a fire that results in the loss of the control room.

Second part is incorrect and plausible. PZR heater Group 2 Banks B and C are controlled from the SSF which would be used if the evacuation was due to a fire.

Answer C Discussion

Correct.

First part is correct. AP/008 directs going to the ASDP when evacuating the control room for any condition other than a fire.

Second part is correct. The ASDP uses PZR heater Bank 2 Groups B and D for RCS pressure control.

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Answer D Discussion

Incorrect.

First part is correct. AP/008 directs going to the ASDP when evacuating the control room for any condition other than a fire.

Second part is incorrect and plausible. PZR heater Group 2 Banks B and C are controlled from the SSF which would be used if the evacuation was due to a fire.

Cognitive Level

Comprehension

Job Level

RO

Question Type

NEW

Question Source

Development References Student References Provided

IC-ASP R3

EAP-SSF R10

3AP/08

401-9 Comments: Remarks/Status

Basis for meeting the KA

Question requires knowledge of RO action outside of the CR during an emergency and how RCS pressure will be controlled by that RO.

Basis for Hi Cog

Basis for SRO only

GEN2.4 2.4.34 - GENERIC - Emergency Procedures / Plan

Emergency Procedures / Plan

Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects. (CFR: 41.10 / 43.5 /45.13)

Oconee Nuclear Station 2013-301 Written Examination

Question 75

Unit 3 initial conditions:

- Reactor power = 100%
- LPSW Pump Auto Start Circuitry is enabled

At 0400:

- A rupture in the Unit 3 LPSW discharge header occurs.

At 0402, the following current conditions exist. LPSW header pressure has decreased to 20 psig with the following alarms actuated:

- 3SA-9/B-9 LPSW RBCU A Cooler Rupture
- 3SA-9/C-9 LPSW RBCU B Cooler Rupture
- 3SA-9/D-9 LPSW RBCU C Cooler Rupture
- 3SA-9/A-9 LPSW Header A/B Pressure Low
- OAC computer point D2100, RCP MTR CLR INL HDR FLOW LO/NORMAL

Based on the conditions above and taking no manual operator actions, which one of the choices below correctly completes the following statement:

RB Aux Coolers LPSW supply and return valves, 3LPSW-1054, 1055, 1061, and 1062, are (1) and the LPSW 3A and 3B Hdr From RX BLDG Auto Isolation valves, 3LPSW-1121, 1122, 1123, and 1124, are (2).

- A 1. open
 2. open
- B 1. open
 2. closed
- C 1. closed
 2. open
- D 1. closed
 2. closed

Proposed Answer: A

K/A Match Analysis

G 2.4.46: Ability to verify that the alarms are consistent with the plant conditions.

Oconee Nuclear Station 2013-301 Written Examination

Requires ability to evaluate alarm status based on plant conditions and determine the status of LPSW alignment to the RB loads.

Answer Choice Analysis

Answer A: Correct. Per reference OP/3/A/6103/008, alarm C-9 (RB LPSW WATER HAMMER ISOLATION), all of the listed valves for RBAC will not auto close until 18 psig based on 2 of 4 analog channels or 2 of 4 digital channels.

Answer B: Incorrect. First part is correct. Per reference OP/3/A/6103/008, alarm C-9 (RB LPSW WATER HAMMER ISOLATION), all of the listed valves for RBAC will not auto close until 18 psig based on 2 of 4 analog channels or 2 of 4 digital channels. Second part is incorrect and plausible. Prior to 2009, the setpoint for LPSW to RBCU was 23 psig and an erroneous assumption that this isolation has occurred by the information given exists. Additionally an OAC computer point alarm was provided which an individual may erroneously assume was due to a flowpath isolation; however it is due to a low flow sensed at the inlet of the cooler; OAC computer point D2100, RCP MTR CLR INL HDR FLOW LO/NORMAL actuates if flow through 1 and 3 LPSW-6 drops to ≈ 1440 gpm.

Answer C: Incorrect. First part is incorrect and plausible. The annunciator response for 3SA-9/B-9, C-9 or D-9, LPSW RBCU A, B or C Cooler Rupture requires the coolers to be isolated for an actual leak or rupture; an erroneous assumption that this action occurred automatically is possible. Second part is correct. Per reference OP/3/A/6103/008, alarm C-9 (RB LPSW WATER HAMMER ISOLATION), all of the listed valves for RBAC will not auto close until 18 psig based on 2 of 4 analog channels or 2 of 4 digital channels.

Answer D: Incorrect. First part is incorrect and plausible. The annunciator response for 3SA-9/B-9, C-9 or D-9, LPSW RBCU A, B or C Cooler Rupture requires the coolers to be isolated for an actual leak or rupture; an erroneous assumption that this action occurred automatically is possible. Second part is incorrect and plausible. Prior to 2009, the setpoint for LPSW to RBCU was 23 psig and an erroneous assumption that this isolation has occurred by the information given exists. Additionally an OAC computer point alarm was provided which an individual may erroneously assume was due to a flowpath isolation; however it is due to a low flow sensed at the inlet of the cooler; OAC computer point D2100, RCP MTR CLR INL HDR FLOW LO/NORMAL actuates if flow through 1 and 3 LPSW-6 drops to ≈ 1440 gpm.

Technical Reference(s):

Lesson Plan SSS-LPW, Rev 21, pages 20-22, 39

OP/3/A/6103/008, Alarm Response Guide 3SA-08

OP/3/A/6103/009, Alarm Response Guide 3SA-09

(Attach if not previously provided) _____

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A. State how the modification will respond to design bases events and conditions beyond design basis. **(OBJ. R13)** Selected Valves' Function/operation

1. **1, 2, 3 LPSW-251 & 252**: (LPI Cooler outlet controllers) these are Moore controllers on all three units. Their setpoint is 3000 gpm. This is set by the operator per OP/A/1104/04, LPI procedure, on unit startup. When no flow is present, there is an internal limiter in the Moore controller that limits valve position to 50% open. Once any flow is detected, the 50% limiter is automatically removed.

There is a switch for each valve that will fail the valve open if necessary. These are two position switches - NORMAL and FAIL OPEN. They are positioned by the operator per procedure. If in "FAIL OPEN", the high flow interlock (see below) will not throttle the valve as air is bled off the pneumatic operator.

LPSW flow to the LPI coolers will automatically runback at ≥ 5900 gpm to a flow rate of 5200 gpm. Automatic runback prevents exceeding the LPI cooler shell design flow of 6000 gpm. A manual Reset button is provided above the Moore controller. Depressing the Reset button will return control of the valve to the operator via the Moore controller.

Mechanical travel stops have been added to the control valves that limit valve travel to $\approx 85\%$ open. Due to high pipe resistance on Unit 2, the travel stops for 2LPSW-251 and 2LPSW-252 are fully retracted (i.e. 100% open). Travel stops ensure that flow will not damage the LPI Cooler.

2. **1, 2, 3 LPSW-4 & 5**: (LPI cooler outlet valves) these valves have been modified to give them throttle capability. If LPSW-251, 252 were to fail open, LPSW-4 and/or 5 may be throttled to prevent robbing flow from the RBCUs. This will also protect the LPSW pumps from pump run-out. If the Moore controller were to fail during LPI decay heat removal, LPSW-4 and/or 5 may be throttled to prevent overcooling the RCS.
3. **1, 2, 3 LPSW-7&8, 9&10, 11&12, 13&14**: (RCP Motor bearing and air cooler inlet and outlet valves) one switch controls both the inlet and outlet valve for that RCP. OAC/Graphic Display should be checked to verify that both valves have traveled to the desired position.
4. **1, 2, 3 LPSW-516 & 525**: (Motor Driven EFWDs cooling water outlet valves) these valves open automatically on pump start. They fail open on loss of instrument air or loss of power.
5. **1, 2, 3 HPSW-184**: (Oil Cooler Bypass valves) pneumatic valve fails open on loss of AC power to supply cooling from HPSW. The switch is located on the SGLC Panel, and is labeled as follows:
1, 2, 3 HPSW-184
6. **1, 2, 3 LPSW-139**: (Non essential header isolations) remotely operated, seismically qualified valves which can isolate the non-essential (non-seismic) header from the safety related portions of the system.

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7. **1, 2, 3 LPSW-1054, 1055, 1061, 1062:** (RBAC LPSW Supply & Return) Containment isolation valves (CIVs) for the RBACs piping that penetrate the reactor building. Close on ES 5 & 6, loss of IA or if LPSW header pressure decreases to 18 psig (Water Hammer Concern). The LPSW LOW PRESS DIG CH 1 & Ch 2 PBs on VB3 must be used to reset the logic to allow opening of the valves.

J.8 (OBJ. R6) LPSW Water Hammer MOD

- A. Generic Letter 96-06 required utilities to evaluate the potential for water hammers in cooling water systems serving containment following a Loss of Offsite Power (LOOP) concurrent with a Loss of Coolant Accident (LOCA) or Main Steam Line Break (MSLB).
1. Analysis and system testing concluded that water hammers occur in the LPSW system during all LOOP events.
 2. Column Closure water hammers (CCWH) occur when the LPSW pumps restart following a LOOP and rapidly close vapor voids within the system, specifically, in the RBCU and RCP motor piping.
 3. Condensation-Induced Water Hammers (CIWH) occur when heated steam voids interact with sub-cooled water in long horizontal piping sections, specifically in the RBCU and RB Aux. cooler piping.
 4. The operability evaluations in response to GL 96-06 concluded that the LPSW piping would not fail, however piping code allowable stress would be exceeded.
 5. LPSW piping code compliance is achieved by the installed LPSW Water Hammer MOD, which will prevent all GL 96-06 related water hammers.

B. Response during a LOOP

1. Loss of Power;
 - LPSW pumps will stop and LPSW header pressure will decrease.
 - At 18 psig the LPSW RB Isolation Circuitry will close the pneumatic discharge isolation valves, LPSW-1121, 1122, 1123, and 1124.
 - Controllable vacuum valves LPSW-1150 and 1151 open.
 - Check valves LPSW-1111 and 1116 will close as soon as a flow reversal in the LPSW supply header begins.
 - As the water in the LPSW piping warms up and expands, thermal relief valves LPSW-1127 and 1135 prevent over pressurization of the LPSW piping by relieving downstream of the pneumatic discharge isolation valves.

10/14/09	Rev. 20	<ul style="list-style-type: none"> • Corrected AP/24 title in the References section. • Modified step #s 2.2.A.9 and 2.2.A.15 to match procedure L&P. • Changed LPSW pressure in step 2.2.C.7 from 23 psig to 18 psig. Modified step 2.4 and the summary info on Water Hammer MOD to apply to all units. Water hammer MOD now applicable to all units. 	SSL
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Oconee Nuclear Station 2013-301 Written Examination

2009 NRC REACTOR OPERATOR EXAM

1 POINT

Question 15

Unit 3 initial conditions:

- Reactor power = 100%

Current conditions:

- LPSW header pressure = 20 psig increasing

Based on the above conditions, which ONE of the following describes the status of statalarm 3SA-9/C3 (LPSW Low Press RB Aux Cooler Isolation) and if the RBACs isolate, how is LPSW flow restored?

- A. Actuated / Automatically when LPSW pressure returns above setpoint
- B. Actuated / Manually after depressing LPSW LOW PRESS DIG CH 1 AND 2 pushbuttons
- C. Not Actuated / Automatically when LPSW pressure returns above setpoint
- D. Not Actuated / Manually after depressing LPSW LOW PRESS DIG CH 1 AND 2 pushbuttons

2009 NRC REACTOR OPERATOR EXAM

Question 15

T1/G1 jmb

062AG2.4.46, Loss of Nuclear Service Water

Ability to verify that the alarms are consistent with the plant conditions.

(4.2/4.2)

K/A MATCH ANALYSIS

Requires ability to evaluate alarm status based on plant conditions and determine status of LPSW to the RB Aux Coolers.

ANSWER CHOICE ANALYSIS

Answer: B

A. Incorrect: First part is correct. The alarm setpoint was reached at 23 psig decreasing and pressure must rise to > 23 psig in order for the S/A to automatically clear. Second part is incorrect. RBACs did isolate; however rising LPSW header pressure above 23 psig will not automatically restore flow to the RBACs. Plausible if the student fails to realize the RBCUs are not automatically restored.

B. CORRECT: Alarm status is correct; alarm setpoint was reached at 23 psig decreasing and pressure must rise > 23 psig in order for the S/A to automatically clear. RBACs did isolate and they must be manually restored to their normal OPEN/AUTO status after depressing both LPSW LOW PRESS DIG CH 1 AND 2 pushbuttons.

C. Incorrect: First part is incorrect. Alarm setpoint was reached at 23 psig decreasing and pressure must rise to > 23 psig in order for the S/A to automatically clear. Plausible if the student does not remember the alarm setpoint. Second part is incorrect as described above.

D. Incorrect: First part is incorrect as described above. Second part is correct as described above.

Technical Reference(s): **Statalarm 3SA-9/C3 (LPSW Low Press RB Aux. Cooler Isolation)**

Proposed references to be provided to applicants during examination: **None**

Learning Objective: **SSS-LPW Obj R13, PNS-RBC Obj. R14**

Question Source: **N**

Question History: Last NRC Exam _____

Question Cognitive Level: **Memory or Fundamental Knowledge**

Comprehension or Analysis

No changes // MB OK /// Enhanced plausibility statements.