# INITIAL SUBMITTAL OF THE WRITTEN EXAMINATION

# FOR THE QUAD CITIES EXAMINATION - DEC 2002 (Part 2 of 2)

700

D.5. (cont'd)

b. **IF** a Recirc Pump is **NOT** running, **THEN** **monitor** both:

(1) RHR Inlet Temperature to RHR Heat Exchanger (TE-1046A/B on recorder TRS-1040-5, Panel 901(2)-3. ____

**AND**

(2) Reactor Shell and Reactor Flange temperatures on recorder 1(2)-263-105, Panel 901(2)-4. ____

## CAUTION

**IF** both Recirc Pumps are OFF, Reactor vessel water temperature stratification may occur causing Reactor repressurization. An indication of stratification could be:

- An observed increase in Reactor metal temperatures which did **NOT** result from a decrease in Reactor vessel level or an increase in Reactor water temperature.
- An unexpected increase in Reactor Pressure.

**IF**, during performance of this procedure, indications of stratification exist, **see** section D.4.d below.

D.6. **IF** at ANYTIME during this procedure, Reactor water temperature increases to 212°F, **THEN**:

a. **Establish** Primary and Secondary Containment Integrity. (G.1.m) (G.1.n) ____

b. **Evacuate** the Drywell. ____

c. **Consider** tripping RECIRC PMP 1(2)A and B to minimize heat input into vessel **AND** to maintain Reactor temperature < 212°F. ____

D.7. (cont'd)

## CAUTION

While performing the following step, **do NOT** allow Reactor water level to reach 100 inches. 100 inches is the elevation of the Main Steam lines.

d. **IF** both Recirc Pumps are OFF **AND** indications exist of Reactor water temperature stratification, **THEN either start** a Recirc Pump **OR raise** Reactor level **AND perform** a feed and bleed as follows:

(1) **Raise** Reactor level to 90" to 100" per LI-263-101, WR RX WTR LVL, using one of the following methods:

(a) CRD System per QCOP 0300-16. ____

(b) Condensate Booster Pump via Feedwater System per QCOP 0201-02. ____

(c) Condensate Transfer System via RHR System per QCOP 1000-26. ____

(d) Condensate System per QCOP 3200-09. ____

(e) LPCI per QCOP 1000-30. ____

(f) Core Spray per QCOP 1400-02. ____

(2) **Place** Reactor Water Cleanup System in operation per QCOP 1200-11 and **reject** through the RWCU System to the Main Condenser or Radwaste per QCOP 1200-07 while maintaining Reactor level 90" to 100". ____

# 60



E.14. Repeated starts/jogs of induction motors greatly reduces the life of the windings. **WHEN** the amount of decay heat requires SDC to be operated only for short periods of time, **THEN** throttling the 1(2)-1001-17A/B, 1(2)A/B RHR HX OUTLET VLV, is preferred to starting/stopping Shutdown Cooling.

E.15. **WHEN** an RHR Pump is run in the minimum flow mode of operation for more than 10 minutes, **THEN notify** the IST group to perform a vibration analysis to ensure **NO** pump degradation has occurred. Although the pump shall **NOT** be declared inoperable, the vibration analysis should be performed within 72 hours. (H.8.b)

E.16. **WHEN** one Recirc Pump is **off**, **THEN perform** QCOP 1000-17 at least once per hour.

E.17. **WHEN** both Reactor Recirc Pumps are **off**, **THEN**:

a. **Perform** QCOP 1000-17 at least once per hour.

b. **Maintain** Rx vessel water level > 90" to prevent stratification. **IF** the Main Steam lines must be protected from flooding, **THEN maintain** Rx vessel level < 100" as read on computer pt. C104(204) **OR** LI 1(2)-263-101, WR Rx WTR LVL GEMAC UPPER 400 on panel 901(2)-4. LI 1(2)-263-101 is calibrated for cold conditions. (H.8.c, H.8.d)

c. **IF** increasing metal temperature occurs without a corresponding vessel level or water temperature change, **THEN refer** to QCOA 1000-02.

E.18. **IF** A/B Reactor Recirc Pump trips **AND** Shutdown Cooling is discharging to the loop, **THEN close** MO 1(2)-202-5A/B, RECIRC PMP DISCH VLV **OR** MO 1(2)-202-4A/B, RECIRC PMP SUCTION VLV.

E.19. **IF** differential temperature between Recirc Loops approaches 50°F, **THEN** actions should be initiated to prevent exceeding the 50°F differential. This could include briefly opening 1(2)-202-4A/B and/or 1(2)-202-5A/B to equalize loop temperatures.

**61** **ID: SR-0302-K24** **Points: 1.00**

During a plant startup RPV pressure is 900 psig.

A sustained loss of CRD flow will have which one of the following immediate effects on control rod motion and scram times?

Normal rod motion is:

A. lost but scram times will be within acceptable limits.

B. unaffected but scram times will NOT meet acceptable limits.

C. lost and scram times will NOT meet acceptable limits.

D. unaffected and scram times will be within acceptable limits.

Answer: A

**Question 61 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #61 (RO/SRO) |
| System ID: | 9781 |
| User ID: | SR-0302-K24 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 3.25 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | LF-0301, pg. 25 |
| User Text: | 295022AK1.01 |
| User Number 1: | 3.30 |
| User Number 2: | 3.40 |
| Comment: | INPO Exambank # 7624. Higher. Cannot drive control rods if CRD pumps are tripped, but rods can still be scrammed at 900 psig. |

As can be seen on Figure 0301-17, the drive may be scrammed by reactor pressure alone however only at reactor pressures above 400 psig. This is an important consideration since, when operating at less than 400 psig with a CRD accumulator discharged or isolated, a rod would be incapable of scramming.

**Emphasize the importance of the fact that at less than 400 psig the rods cannot be scrammed by reactor pressure. At these low pressures (and temperatures), a loss of CRD pumps is not so much a loss of cooling concern as it is the possible inability to scram the rods if the accumulator pressure starts to bleed down.**

The reason that reactor pressure is unable to effect a scram is that even with the difference in area across the piston sides, enough force cannot be developed to overcome the weight of the rod and drive mechanism and the associated friction.

It can also be seen on Figure 0301-17 that when considering scrams with accumulator pressure only, the scram time increases as reactor pressure increases. This is due to the overpiston pressure increasing as reactor pressure increases, causing a decrease in the net upward force.

**SR-0301-K02d**
**L-0301-K02d**

**Show Figure 0301-11, CRD Internal Operation**

Further evaluation of Figure 0301-17 shows the longest combined (accumulator and reactor pressure) scram times are encountered at 800 psig reactor pressure. For this reason technical specifications require scram timing testing to be conducted at greater than or equal to 800 psig.

C. Operational Problems

****SR-0301-K22b**

1. Scramming Too Rapidly (Figures 0301-11 & 0301-18)

Excessively rapid scram times can result from excessive accumulator gas pressure or worn lower stop piston seals. This can result in index tube damage due to excessive pressures in the buffer region that can cause bulging in the reduced cross sectional area at the bottom of the index tube where the drive piston is threaded to the index tube. In extreme cases this can result in rubbing between the index tube and the inner tube.

**Show Figure 0301-18, Accumulator Precharge Nitrogen Pressure vs. Ambient Temperature**

**There is no minimum scram time per the Technical Specification.**

# EXAMINATION ANSWER KEY

2002 Quad Cities NRC Exam

**62** | **ID: SRLF-805-K15** | **Points: 1.00**

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

consider All SRMs gradually increasing by 20 to 30 counts

longer discription tends to cue the answer - make it shorter

SRM doubles & stabilizes

A. ~~Refuel bridge reve~~rse motion interlock activates. not credible

monitors ~~steady~~ increasing and the High

B. ~~A high~~ refuel floor radiation alarm sounds.

increase

C. ~~A sustained~~ upward ~~trend on the~~ source range monitor nearest the fuel bundle lo~~cation.~~

~~continuous~~

D. Source range monitor spiking repeatedly.

nearest the fuel bundles

Answer: C

**Question 62 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #62 (RO/SRO) |
| System ID: | 6712 |
| User ID: | SRLF-805-K15 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 2.75 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | QCFHP 0110-02, R.3, pg1. |
| User Text: | 295023AK1.03 |
| User Number 1: | 3.70 |
| User Number 2: | 4.00 |
| Comment: | LWQ.00297 (81809) Bank question. Higher Criticality is indicatd by a SUSTAINED increase ~~in cou~~nt rate of the SRM nearest the fuel bundle. |

fundamentals

#62



# INADVERTENT CRITICALITY DURING FUEL MOVES

## A. SYMPTOMS

**NOTE**

True criticality is indicated by a sustained increase in count rate, over 15 to 20 seconds, of the SRM closest to the Fuel Assembly/Bundle **OR** Control Rod being moved. The other SRMs may also begin to increase as neutron population increases throughout the core.

A.1. Control Room Operator announces that Reactor is critical as indicated on Nuclear Instrumentation.

A.2. Unplanned Control Rod withdraw observed in Reactor Core.

A.3. **IF** all Control Rods are **NOT** fully inserted into core, **THEN**:

a. Refueling Bridge interlocks to prevent bridge travel towards Reactor Core.

b. Refueling Bridge Hoists interlocks activated to prevent hoist operation..

A.4. Reactor Building Crane interlocks activated to prevent further operation in upward direction.

A.5. Refuel Floor Area Radiation Monitors alarming.

A.6. Reactor Building Ventilation isolation.

A.7. Standby Gas Treatment System starts.

A.8. Reactor scram from SRM/IRM channels.

**63** **ID: SR-0001-K23** **Points: 1.00**

Why is Torus Spray initiated prior to torus pressure reaching 5 psig? Drywell 4.4.7

A. Prevent catastrophic containment failure of the suppresion pool.

B. Allow the nitrogen flow back into the Drywell.

C. Reduce containment pressure by steam condensation and convective cooling.

D. Prevent steam from bypassing the suppression pool.

Answer: C

**Question 63 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #63 (RO/SRO) |
| System ID: | 9732 |
| User ID: | SR-0001-K23 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 4.00 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | LP QGA 200 R2, pg.17 |
| User Text: | 295024EK3.02 |
| User Number 1: | 3.50 |
| User Number 2: | 3.80 |
| Comment: | New question. Memory. Correct answer is to condense any residual steam not condensed by the downcomers. Preventing steam bypassing the suppression pool is function of downcomers. Vacuum breakers allow nitrogen back to drywell and prevent torus failure. QGA 200 lesson plan rev 2 |

D. Torus sprays

1. Initiate torus sprays "before" torus pressure reaches 5 psig.

    a. Drywell sprays are initiated when torus pressure *exceeds* 5 psig to prevent chugging.

    b. Since drywell spray operation can result in the loss of electrical equipment located in the drywell, torus sprays are initiated first.

    c. Torus spray operation cannot prevent chugging, but may reduce primary containment pressure through steam condensation and convective cooling.

    d. If steam is bypassing the suppression pool and entering the torus airspace directly, operation of torus sprays may be effective in controlling pressure.

    e. If the pressure increase is caused by the transfer of noncondensibles from the drywell to the torus, operation of torus sprays will have little effect.

2. Initiate sprays only if torus water level is below 27 ft.

    a. Corresponds to the elevation of the torus spray nozzles.

    b. If the spray nozzles are submerged, there is no benefit to spray initiation.

3. Do not use pumps needed for core cooling.

    a. A pump may be used to supply containment spray only if continuous operation in the injection mode is not required.

    b. Core cooling takes precedence over sprays since catastrophic containment failure is not expected while containment pressure is below 5 psig.

    c. Alternating pumps between injection and spray modes is permitted as long as core cooling can be maintained.

**64** **ID: SR-2300-K15** **Points: 1.00**

The following plant conditions exist;

Reactor pressure is 1090 psig.
DW Pressure is 3.7 psig.
CCST level is at 1,200 gallons.

Torus level is 14 feet 3 inches.

You are required to run HPCI in the Pressure Control Mode.
Determine the correct suction and discharge path of the pump to establish pressure control under these conditions?

| | Suction | Discharge |
|---|---|---|
| A. | CCST; | Minimum flow line |
| B. | Torus; | Minimum flow line |
| C. | CCST; | Test return line |
| D. | Torus; | Test return line |

Answer: B

**Question 64 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #64 (RO/SRO) |
| System ID: | 6303 |
| User ID: | SR-2300-K15 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 3.75 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | LN-2300, pg. 38 |
| User Text: | 295025EA1.04 |
| User Number 1: | 3.80 |
| User Number 2: | 3.90 |
| Comment: | ILT.11978 (81397) Bank question. Higher. Due to CCST level being < 10K gal, the HPCI suction will auto transfer to the Torus. Must use the minimum flow line due to Drywell pressure > 2.5 psig closing the test return line. |

| | | | |
|---|---|---|---|
| g. Emergency HPCI Turbine Trip Test Pushbutton. | Directs oil to mechanical overspeed device. If the turbine is ≥ to 70-75% speed the turbine will trip. | | |

C. Automatic Functions

4. Interlocks

| Device/Setpoint | Logic | Bypass/Reset | Response |
|---|---|---|---|
| a. Torus Suction Valve and CCST Suction Valve | | Can be bypassed. By lifting a lead in the 90X-39 panel in AUX Electric Room on terminal Board GG point 7. | 1. If Supp. Pool level +5" or if CCST level < 10K gal, MO 35 and 36 open.<br>2. When MO 35 and 36 fully open, MO 6 closes.<br>3. If either MO 35 or 36 opens, MO 10, 15 and 49 closes. |
| b. Turbine Steam Supply | | | MO 3 opens on initiation.<br>MO 8 opens on initiation. |
| c. Injection Isol Valves | | | MO 9 open on initiation if closed. |
| d. Test Return Valves | | | MO 10 and 15 close on initiation if closed. |
| e. Cooling Test Valve | | | MO 49 closes on initiation or if MO 35 or 36 opens. |
| f. Minimum Flow Valve | | | 1. If MO 3 open, MO 14 opens at 600 gpm and shuts at 1200 gpm. MO 3 open > 15 sec.<br>2. If MO 3 closed, MO 14 does not cycle as minimum flow valve. |

**SR-2300-K13a**
**SR-2300-K15q**
****N-2300-K13a**
**SR-2300-K14p(2)**
****N-2300-K14s(3)**

**SR-2300-K13c**
**SR-2300-K13d**

**65** **ID: SR-9900-K05** **Points: 1.00**

A LOCA occured on Unit 2. the QGA entry condition
Torus water temperature was 87 degrees F and has now increased to ~~97 degrees~~ F.

For these conditions, the SPDS indications for the Torus Water Temperature colored bar graph changed from _____ to _____.

A. green; red

B. yellow ~~white~~; red

C. green; yellow

D. white; yellow

Answer: C

**Question 65 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #65 (RO/SRO) |
| System ID: | 9804 |
| User ID: | SR-9900-K05 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 3.25 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | QOP 9900-102, pg. 3 |
| User Text: | 295026EK2.04 |
| User Number 1: | 2.50 |
| User Number 2: | 2.80 |
| Comment: | New question. Higher. Per QOP 9900-102, Torus temp is green for < 95 degrees and yellow for 95 to 110 degrees. |

memory

#65



4. SPDS Parameters.

   a. Reactor vessel water level is displayed by a bar chart. The chart will indicate the present level. The color of the chart reflects the condition of water level. The bar will be green for the normal range of 0 to +48 inches. The bar will be yellow between 0 and -59 inches. The bar will be red above +48 inches and below -59 inches. In addition a digital reading of current level is displayed above the chart. An arrow will show the direction the level is changing (up, down, or no arrow for steady).

   b. Reactor vessel pressure is displayed by a bar chart. The chart will indicate the present reactor pressure. The color of the chart reflects the condition of reactor pressure (green for pressure < 1040 psig, red for pressure > 1060 psig, and yellow for pressure between 1040 psig and 1060 psig).

   c. Drywell pressure is displayed by a bar chart. The chart will indicate the present drywell pressure. The color of the chart reflects the condition of drywell pressure (green for pressure < 2.5 psig, yellow for pressure between 2.5 and 12.5 psig, and red for pressure > 12.5 psig).

   d. Drywell temperature is displayed by a bar chart. The chart will indicate the present drywell temperature. The color of the chart reflects the condition of drywell temperature (green for temperature < 180°F and yellow for temperature between 180°F and 280°F, and red for temperature > 280°F).

   e. Torus level is displayed by a bar chart. The scale on the chart is from 0 to 30 feet. This scale reflects the reading of the wide range torus level instrumentation. The color of the bar reflects the condition of torus level (green for level between -2 and +2 inches, yellow for level up to 18 feet and down to 11 feet, and red outside this range).

   f. Torus temperature is displayed by a bar chart. The chart will indicate the present temperature. The color of the bar reflects the condition of torus temperature (green for temperature < 95°F, yellow for temperature between 95°F and 110°F, and red for temperature > 110°F).

NRC COPY #1

**66** **ID: SR-0001-K23** **Points: 1.00**

QGA 200, PRIMARY CONTAINMENT CONTROL, directs the operator to maintain torus temperature below the Heat Capacity Limit and if you cannot, then reduce reactor pressure to stay inside the Heat Capacity Limit.

Reducing reactor pressure to stay inside the Heat Capacity Limit is to:

A. prevent inadequate steam condensation in the event of a full reactor depressurization, resulting in the torus to drywell vacuum breakers opening.

B. ensure there is adequate margin to the ECCS suction piping design temperature in the event of a full reactor depressurization.

C. ensure the torus has enough capacity to accept a full reactor depressurization without exceeding the design temperature of the torus.

D. allow the operator to depressurize the reactor to a point where Core Spray and RHR can inject prior to the torus temperature exceeding the low pressure ECCS pump NPSH limit.

Answer: C

**Question 66 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #66 (RO/SRO) |
| System ID: | 9805 |
| User ID: | SR-0001-K23 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 3.25 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | QGA 200 LP, pg. 53 |
| User Text: | 295026 ? Specific K/A ? K301 |
| User Number 1: | 3.80 ? |
| User Number 2: | 4.10 |
| Comment: | New question. Lower. Heat Capacity Limit is based on keeping Torus temp after a blowdown below torus design temp. |

#60



D. Hold torus temperature below the Heat Capacity Limit.

1. If torus temperature cannot be held below 95°F, hold temperature below the Heat Capacity Limit. — SR-0001-K22

   a. Continue to control temperature with available torus cooling.

   b. No other action is required as long as temperature can be held below the Heat Capacity Limit.

2. Heat Capacity Limit

   a. The Heat Capacity Limit is the highest torus temperature from which a blowdown will not raise:

      - Torus temperature above the torus design temperature, or
      - Torus pressure above the Primary Containment Pressure Limit.

   while the rate of energy transfer from the RPV to the containment is greater than the capacity of the containment vent.

   b. The temperature from which a blowdown will not raise pressure above the Primary Containment Pressure Limit is limiting at QCNPS.

   c. The Heat Capacity Limit is a function of RPV pressure and torus water level.

      1) Fig M is a bounding curve valid for torus water levels at or below 17 ft.

   d. The derivation of the Heat Capacity Limit is discussed in the *Calculations* lesson plan.

3. Elevated torus temperature may damage HPCI/RCIC.

   a. The water being pumped provides cooling for lube and control oil.

   b. High torus temperature is only a concern if suction is aligned to the torus. (CCST suction is normally preferred.)

**67** **ID: SR-0001-K23** **Points: 1.00**

Unit 1 scrammed due to a large LOCA.
A Group One isolation has successfully completed.
Drywell Temperature has risen to 350 degrees Fahrenheit.

What are the immediate concerns?

A. The inboard MSIVs are no longer reliable ~~may NOT function.~~

B. Drywell temperature instrumentations is no longer reliable.

C. Core flow instrumentations is no longer reliable.

D. The ADS valves are no longer reliable ~~may NOT function~~.

Answer: D

**Question 67 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #67 (RO/SRO) |
| System ID: | 9782 |
| User ID: | SR-0001-K23 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 3.50 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | QGA 200 LP, pg. 43 |
| User Text: | 295028EK1.02 |
| User Number 1: | 2.90 |
| User Number 2: | 3.10 |
| Comment: | Bank - LaSalle 2000 #25. Higher. |

Core flow instrumentation is not affected by drywell temperature.

Drywell temperature instruments are not INOP at 350 Deg. F

MSIVs are not the limiting component.

The design limit for the Drywell is 280 Deg F. The SRV solenoids are not environmentally qualified above 338 deg F. and may not function.

(#67)



D. Blow down

1. Keep trying to lower drywell temperature below 180°F.

   a. Continue to control temperature with sprays. All restrictions and operating details continue to apply.

   b. No other action is required as long as temperature can be restored and held below 280°F.

   c. The reference to Detail A is repeated to emphasize its continued applicability.

2. If temperature cannot be restored and held below 280°F: **SR-0001-K22**

   a. Further temperature increases could challenge primary containment integrity and equipment operability.

   b. Depressurization minimizes further release of energy from the RPV to the primary containment.

   c. Perform the blowdown in accordance with QGA 500-1.

   d. The "cannot restore and hold" action level allows flexibility.

      1) If temperature is already high when the instruction is reached, the effectiveness of drywell sprays may be evaluated before a decision is made to blow down.

      2) Brief excursions above 280°F are not expected to challenge either containment integrity or ADS valve operability (the ADS qualification temperature is 338 F).

      3) Spray operation is preferable to a blowdown. A blowdown imposes undesirable stresses upon the RPV and containment, increases the possibility of power instabilities during ATWS events, and adds additional heat to the primary containment.

      4) Does <u>not</u> authorize extended pressurized operation beyond 280°F. The intent is only to allow sprays to lower temperature if possible.

#67

# LaSalle County Station
# ILT Class 1999 NRC Written Exam

| Q_ID | System: | K/A: | Importance | Exam: | Objective: | Cognitive Level: |
|---|---|---|---|---|---|---|
| # 25 | **295028** | **2.4.18** | **2.7** | **RO** | **400.00.14** | **2** |

Unit 1 has just scrammed due to a large LOCA.

Drywell Temperature has risen to 350 degrees Fahrenheit.

What are the concerns, if any?

A. Core flow instrumentation is no longer reliable.

B. Drywell temperature instrumentation is no longer reliable.

C. The MSIVs may not function.

D. The SRVs may not function.

| Answer: | Reference(s): | Question Reference(s): |
|---|---|---|
| **D** | **LP LGA-003, Rev 14, pg. 22** | **090.00.24 001 (modified)** |

**Explanation:**

| | |
|---|---|
| A. Incorrect | Core flow instrumentation is not affected by drywell temperature. |
| B. Incorrect | Drywell temperature instruments are not INOP at 340 Deg. F |
| C. Incorrect | MSIVs are not the limiting component. |
| D. Correct | The design limit for the Drywell is 340 Deg F. The SRV solenoids are not environmentally qualified above these temperatures and may not function. |

**68** **ID: SR-0001-K43** **Points: 1.00**

QGA 500-2, "Steam Cooling" specifies actions that use the steam cooling method of heat transfer to _________ that the reactor core remains adequately cooled under conditions when _________ source of injection into the RPV is available.

A. maximize the time; no

B. indefinitely ensure; no

C. maximize the time; a single

D. indefinitely ensure; a single

Answer: A

**Question 68 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #68 (RO/SRO) |
| System ID: | 2643 |
| User ID: | SR-0001-K43 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 3.25 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | QGA 500-2, R. 2, pg 1 |
| User Text: | 295031EK3.04 |
| User Number 1: | 4.00 |
| User Number 2: | 4.30 |
| Comment: | Bank question. Memory. Purpose of Steam Cooling is to Maximize heat transfer when there is no injection. |

# I. INTRODUCTION

A. Purpose

SR-0001-K43

1. QGA 500-2 relies upon steam cooling heat transfer to prolong the time that the reactor core remains adequately cooled when no source of RPV injection is available.

B. Steam cooling strategy

1. QGA 500-2 is entered with no RPV injection source available and RPV water level decreasing. When RPV water level reaches –184 in., a blowdown is performed.

2. When RPV water level drops below the top of the active fuel, steam generated by the covered portion of the core will provide some cooling of the uncovered portion.

   a. The amount of heat removed is a function of fuel temperature, heat capacity of the steam, the heat transfer coefficient, and the steam flow rate.

   b. Fuel temperatures in the uncovered portion of the core will increase to an equilibrium at which the heat transfer to steam is sufficient to remove the decay heat generated.

3. With no RPV injection, the core is defined to be adequately cooled if the steam flow is sufficient to maintain the hottest peak clad temperature (PCT) below 1800°F.

SR-0001-K44

   a. 1800°F is the onset of significant metal-water reaction.

   b. The steam generated by the covered portion of the core is a function of RPV water level. Adequate steam flow will exist as long as RPV water level remains above –184 in., the "Minimum Zero-Injection RPV Water Level" (MZIRWL).

   c. The MZIRWL is the lowest RPV water level at which the covered portion of the reactor core will generate sufficient steam to prevent any temperature in the uncovered part of the core from exceeding 1800°F.

   d. The derivation of the MZIRWL is discussed in the *Calculations* lesson plan.

**69** **ID: SR-0001-K29** **Points: 1.00**

Unit 2 is operating at 100% power and just experienced an invalid FULL Group 2 isolation.
All systems responded as expected.
Which of the following is most likely to cause entry into QGA 300?

A. MSIV Room High Temperature.

B. HPCI Room Area Radiation.

C. Reactor Building Low Differential Pressure.

D. Reactor Building Ventilation Radiation.

Answer: A

**Question 69 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #69 (RO/SRO) |
| System ID: | 9776 |
| User ID: | SR-0001-K29 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 3.25 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | QGA 300, QCOA 5750-07 |
| User Text: | 295032EK2.02 |
| User Number 1: | 3.60 |
| User Number 2: | 3.70 |
| Comment: | New question. Higher. When Rx Bldg vents are isolated, the concern is excessive MSIV Room Temps. SBGT is designed to maintain Rx Bldg Delta-P when RB Vents isolate. Isolation is invalid, so no reason to suspect rads. |

#69

| ENTRY CONDITIONS | | | | |
|---|---|---|---|---|
| Any of the following in the reactor building: | | | | |
| **Area temperature** above alarm setpoint (Table S) | **Differential Pressure** at or above 0 in. | **Area radiation** above alarm setpoint (Table T) | **Vent radiation** above 10 mr/hr | **Area water level** above 1 in. |

Quest #69

QOA 5750-07
Revision 9
**Continuous Use**

# REACTOR BUILDING VENTILATION ISOLATION

A. SYMPTOMS

1. Alarms.

   a. RX BLDG 1 SPLY/EXH FAN TRIP, panel 912-5 A-1.

   b. RX BLDG 2 SPLY/EXH FAN TRIP, panel 912-5 A-4.

   c. RX BLDG 1 LOW DP, panel 912-5 C-1.

   d. RX BLDG 2 LOW DP, panel 912-5 C-4.

2. RX BLDG ISOL DAMPERS indicate closed on panel 912-1.

3. Local indicating lights for the RX BLDG ISOL DAMPERS indicate closed on local panels 2251-24X and 2252-24X.

B. AUTOMATIC ACTIONS

1. All eight Reactor Building isolation dampers close.

2. All Reactor Building supply and exhaust fans trip.

3. The Standby Gas Treatment System auto-starts.

C. IMMEDIATE OPERATOR ACTIONS

1. None.

D. SUBSEQUENT OPERATOR ACTIONS

1. Verify all AUTOMATIC ACTIONS have occurred.

2. Trip the fan control switches, for the fans that tripped, to prevent an auto start when power is restored or ventilation is reset.

3. Notify Radiation Chemistry that the Reactor Building ventilation has isolated.

4. Return the Reactor Building ventilation system to normal as soon as the cause is corrected. For resetting the isolation signal and to restart the fans, refer to QOP 5750-02, Reactor Building Ventilation System.

E. DISCUSSION

1. It is imperative that Reactor Building Ventilation be restored as soon as possible to ensure temperatures in the MSIV Room remain below the MAX NORMAL OPERATING TEMPERATURE limits.

(final)

NRC COPY #1

**70** **ID: SR-5750-K20** **Points: 1.00**

Both Units Reactor Building Ventilation supply and exhaust fans have tripped and the isolation dampers have automatically closed.
NO ECCS systems have initiated on either unit.

This was caused by a 25 GPM leak from the:

A. RWCU filter demineralizer.

B. RBCCW pump discharge header.

C. Fuel Pool filter demineralizer.

D. Reactor Feed pump casing.

Answer: A

**Question 70 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #70 (RO/SRO) |
| System ID: | 9806 |
| User ID: | SR-5750-K20 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 3.50 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | QCAN 901(2)-3 H-3 R. 8 |
| User Text: | 295034EA2.02 |
| User Number 1: | 3.70 |
| User Number 2: | 4.20 |
| Comment: | New question. Higher. RWCU is the only high pressure, high temp system inthe Rx Bldg. RBCCW is low temp. low press. RFPs are in the Turbine Building. Fuel Pool Demins are in Radwaste. |

Q#70



**DESCRIPTION** REACTOR BUILDING VENT RADIATION MONITOR CHANNEL B HIGH HIGH RADIATION

**SETPOINT** **Actual:** 3mR/hr.

**Tech Spec:** ≤ 9 mR/hr.

RX BLDG VENT
CHANNEL B
HI HI RADIATION

**SENSOR** 1(2)-1735B.

**A. AUTOMATIC ACTIONS**

1. Reactor Building Ventilation System isolates.
2. Control Room Ventilation System isolates.
3. Standby Gas Treatment System starts.

**B. OPERATOR ACTIONS**

1. **Verify** automatic actions occur.
2. **Monitor** the following and **confirm** high radiation:
   a. **Trip** Unit 1(2)1705-8B/A RX BLDG VENT CH B and A (located on 901(2)-10 Panel).
   b. Recorder 1(2)-1705-21, RB VENT EXH MONITOR (located on 901(2)-2 Panel).
   c. Area Radiation and Temperature Monitors (located on 901(2)-11 and 901(2)-21 Panels).
   d. Recorder 1/2-1740-203, RX BLDG STACK GAS ACTIVITY (located on 912-1 Panel).
3. **IF** high radiation condition is confirmed, **THEN**:
   a. **Evacuate** Reactor Building.
   b. **IF** indications exist of a Primary System leak outside Primary Containment, **THEN perform** QCOA 0201-05.

NRC COPY #1

**71** **ID: SR-0001-K61** **Points: 1.00**

An ATWS is in progress on Unit 2 with the following parameters.

- Reactor water level -30 inches
- Drywell pressure 8 psig

Plans are to inject boron using the RWCU system.
Simultaneously, reactor water level is being lowered to control reactor power.

Jumpers must be installed to allow opening RWCU : filter d/p 35 psid

A. filter demineralizer isolation valve when ~~Drywell pressure is 8 psig~~.

B. isolation valves when reactor water level is -30 inches.

C. isolation valves when Drywell pressure is 8 psig.

D. filter demineralizer isolation valve when reactor water level is -30 inches.

Answer: B

**Question 71 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #71 (RO/SRO) |
| System ID: | 8053 |
| User ID: | SR-0001-K61 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 2.75 |
| Time to Complete: | 2 |
| Point Value: | 1.00 |
| Cross Reference: | QCAP 0200-10, Att. M |
| User Text: | 295037EA1.11 |
| User Number 1: | 3.50 |
| User Number 2: | 3.60 |
| Comment: | 124667 (revise from LORTB) Bank question. Lower. RWCU isolates on 0 inches, but this does not prevent opening the filter demin bypass. High Drywell pressure is not a RWCU isolation. |

#71



ATTACHMENT M (Page 1 of 1)

AUTOMATIC ACTIONS WHICH OCCUR AT LOW REACTOR LEVEL

1. Full reactor scram.

2. Group 2 automatic isolation.

   The following valves close:

   "TIP ALL CLOSED" Mimic Lit
   Nitrogen Purge AO 1601-55
   $O_2$ Anal Vlvs

| | | | | |
|---|---|---|---|---|
| 8802A | 8802B | 8802C | 8801B | 8801C |
| 8802D | 8804 | 8801A | 8801D | 8803 |

   1601 Vlvs

| | | |
|---|---|---|
| AO-1601-22 | AO-1601-21 | AO-1601-56 |
| MO-1601-57 | AO-1601-59 | AO-1601-58 |
| AO-1601-63 | AO-1601-62 | AO-1601-23 |
| AO-1601-24 | AO-1601-61 | AO-1601-60 |

   RHR Vlvs

| | |
|---|---|
| SDC Supply | MO-1001-47 |
| LPCI Injection | MO-1001-29B (When in SDC mode) |
| Disch to RW | MO-1001-21 |
| SDC Supply | MO-1001-50 |
| LPCI Injection | MO-1001-29A (When in SDC mode) |
| Disch to RW | MO-1001-20 |
| D/W Pneum Supp | AO-4720, 4721 |
| D/W Flrl Drn Disch | AO-2001-3, 4 |
| D/W Equip Drn Disch | AO-2001-15, 16 |
| TIP Purge Vlv | SO-0799-3D |

3. Group 3 automatic isolation.

   The following RWCU system valves close on a Group 3 isolation:

| | |
|---|---|
| Pump Suction Isolation Valve | MO-1201-2 |
| Pump Suction Isolation Valve | MO-1201-5 |
| Return Isolation Valve | MO-1201-80 |

4. Standby Gas Treatment System automatically starts.

5. Rx Bldg Isolation.

| | |
|---|---|
| Rx Bldg Inlt Damper | 5741A |
| Rx Bldg Inlt Damper | 5741B |
| Rx Bldg Outlt Damper | 5742A |
| Rx Bldg Outlt Damper | 5742B |

6. Control Room ventilation goes on recirc.

#71

⑧

CAUTION: Group I isolation will occur at -59 in. RPV water level.

1. Lower RPV water level by preventing all RPV injection except:
   - Boron injection
   - CRD
   - RCIC

   ☛ Ignore any power or level oscillations.
2. Let level drop at least to -35 in.
3. Let level continue to drop until:
   - Power drops below 3%

     OR
   - Level drops to -142 in. (TAF)

     OR
   - All ADS valves stay closed and drywell pressure stays below 2.5 psig
4. Record final level: ____________________



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F-9 Sys B

**72** Alan 912-1-E-9, Rad Mont Sys A High Scale **ID: SR-1701-K05** **Points: 1.00**

The Radwaste Effluent CAN be monitored in the___1___:
The Service Water Effluent CAN be monitored in the___2___:

A. 1. Radwaste Control Room ONLY
2. "B" CR HVAC Room

B. 1. Radwaste Control Room AND Main Control Room
2. Main Control Room

C. 1. Radwaste Control Room ONLY
2. Main Control Room

D. 1. Radwaste Control Room AND Main Control Room
2. "B" CR HVAC Room

Answer: B

ok for monitoring portion of KA for via rad monitor but need the portion pertaining to High OFFSITE Release Rate

**Question 72 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #72 (RO/SRO) |
| System ID: | 9803 |
| User ID: | SR-1701-K05 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 3.00 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | QCOA 1700-02R5 |
| User Text: | 295038EA1.03 |
| User Number 1: | 3.70 |
| User Number 2: | 3.90 |
| Comment: | New question. Memory. Answer is correct due to radwaste effluent monitors via at both the eberline in the control room and the 1/2 1791 in the radwaste control room. Service water effluent only available in the main control room. "B" HVAC room has no monitoring capability. QCOA 1700-02 rev 5 |

Ques # 12



D.3.b. (con't)

(4) **IF** a Primary System leak outside the Primary Containment exists, **THEN perform** QCOA 0201-05.

(5) **Refer** to applicable QGAs.

(6) **Notify** Rad Protection and Chemistry of existing conditions, and **request** access control to the affected areas.

D.4 **IF** alarming channel is Radwaste Effluent, **THEN:**

a. **Monitor** recorder 1/2-1791, RADWASTE EFFLUENT RADIATION, at Radwaste panel 2212-4.

b. **IF** Radwaste Effluent high radiation is confirmed, **THEN:**

(1) **Terminate** Radwaste discharge until cause of high radiation has been identified and corrected.

(2) **Verify** Radwaste discharge was performed per the requirements of QOP 2000-24 or QOP 2000-25.

(3) **Notify** Chemistry and Radwaste Foreman of high radiation.

D.5. **IF** alarming channel is Service Water Effluent, **THEN:**

a. **Monitor** recorder 1(2)-1705-12, PROCESS LIQUID MONITOR, at 901(2)-2.

b. **Determine** if any radioactive material is being moved in the vicinity of monitors.

c. **IF** a high radiation spike has occurred on the Service Water monitor **AND** it was **NOT** due to movement of any radioactive material, **THEN**:

(1) **Notify** Chemistry to respond to Service Water Monitor alarm.

(2) **Monitor** recorder 1(2)-1705-12, PROCESS LIQUID MONITOR, for trends related to change in plant equipment status.

#72

## E. DISCUSSION

The following process channels are monitored by the Eberline Radiation Monitoring System:

| PROCESS CHANNEL | CHANNEL UNIT NUMBER | UNITS |
|---|---|---|
| Rx Vent SPING | | |
| Rx Vent Low Range Noble Gas | 01-05 | $\mu$Ci/cc |
| Rx Vent Area Monitor (not used) | 01-06 | mR/hr |
| Rx Vent Mid Range Noble Gas | 01-07 | $\mu$Ci/cc |
| Rx Vent High Range Noble Gas | 01-09 | $\mu$Ci/cc |
| Chimney SPING | | |
| Chimney Low Range Noble Gas SPING | 03-05 | $\mu$Ci/cc |
| Chimney Area Monitor (not used) | 03-06 | mR/hr |
| Chimney Mid Range Noble Gas SPING | 03-07 | $\mu$Ci/cc |
| ~~Chimney High Range Noble Gas~~ | ~~03-09~~ | ~~$\mu$Ci/cc~~ |
| Radwaste Effluent | 10-01 | cpm |
| Unit 1 Service Water Effluent | 11-01 | cpm |
| ~~Unit 2~~ Service Water Effluent | 12-01 | cpm |

## F. ATTACHMENTS

None.

## G. REFERENCES

G.1. **Technical Specifications:**

None.

G.2. **P&IDs:**

None.

G.3. **Drawings:**

a. 4E-1489, Schematic Diagram Process Radiation Mon System.

G.4. **Manuals:**

a. Offsite Dose Calculation Manual Chapter 12.

# EXAMINATION ANSWER KEY

2002 Quad Cities NRC Exam

**73** — **ID: SR-4700-K22** — **Points: 1.00**

The 1A instrument air compressor is running when the unloader valve fails in the OPEN position.

What effect would this have on compressor / plant operation and what operator action is required?

It appears to be obvious if unloader fails open how could sys press get high? ? B & D dist credible?

A. The compressor would NOT develop any discharge pressure possibly resulting in low system pressure.
Start a standby Instrument Air Compressor.

B. High system air flow ~~pressures~~ ~~could result due to an inability of the~~ compressor ~~to~~ damage ~~relieve excess pressure to the compressor intake.~~
Open the manual dryer bypass valve.

C. The compressor would NOT develop any discharge pressure possibly resulting in low system pressure.
Open the manual dryer bypass valve.

D. High system air pressures could result due to ~~an inability of the compressor to relieve excess pressure to the compressor intake.~~ Same
Start a standby Instrument Air Compressor.

Answer: A

**Question 73 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #73 (RO/SRO) |
| System ID: | 9755 |
| User ID: | SR-4700-K22 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 3.25 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | LP46/4700, QOA 4700-06 |
| User Text: | 300000A2.01 |
| User Number 1: | 2.90 |
| User Number 2: | 2.80 |
| Comment: | New question. Higher. The unloader opening will cause the compressor discharge pressure to decrease, requiring starting a standby compressor. |

area. An oil seal ring is located by the bearing end of the shaft and an air seal is located by the compressor end, any oil entering the space between the seals is drained off. This arrangement makes these compressors "oil free."

The compressors are controlled from panel 912-1, in the Control Room. The 1A and U-2 compressors have a feeder breaker local control switch. The 1A,1/2, and the U-2 compressors have a local start and shut down capability. The 1A, 1/2, and 2 compressor/dryer arrangement is shown in Figure 4600/4700-11.

**Show Figure 4600/4700-11.**

**Q: Screw type compressors are classified as positive displacement. What does that mean?**

**A: It means that the compressors are designed to displace a specified amount of air regardless of discharge pressure. If a positive displacement compressor is started with no discharge path, the compressor or the system will rupture.**

The 1A and U2 compressor breakers can only be closed in from the local control switch. There is no remote control of the breaker. The 1/2 compressor control switch controls both the breaker and the compressor motor starter.

The Unit 1/2 air compressor should not be stopped from the Control Room except in an emergency due to the compressor not going through its' unload cycle, possibly resulting in damage.

B. Unloader

**S/R-4700-EK014**
****S/R-4700-EK015**
****S/R-4700-EK022**
****S/R-4700-EK026**

Pressure control is accomplished by a solenoid operated unloader valve on each compressor. When receiver pressure reaches a high setpoint, a pressure switch sends a signal to the unloader valve to open, equalizing pressure between the suction and discharge of the compressor. When receiver pressure drops to the low setpoint, the unloader valve closes. If the unloader were to fail in the shut position (solenoid de-energized), compressor discharge pressure would increase (depending on system load) and various system relief valves may open to relieve the pressure. If it should fail/stick in the open position system pressure would start to decrease and the other compressors and/or the service air system would supply the instrument air system loads.

1. The following are the individual unloader valve setpoints:

   -1A cycles between 120 and 102 psig discharge pressure and/or 105 and 95 psig receiver pressure
   -1B cycles between 105 and 95 psig receiver pressure
   -1/2 cycles between 116 and 95 psig discharge pressure
   -U-2 cycles between 116 and 102 psig discharge pressure and/or 105 and 95 psig receiver pressure

B. Automatic Functions

1. Initiation

**S/R-4700-EK020

| Device/Setpoint | Logic | Bypass/Reset | Responses |
|---|---|---|---|
| Solenoid operated unloader valve for the 1B compressor opens at 105 psig receiver pressure or when discharge/casing pressure lowers to 60 psig. | Pressure switch (PS-1-4741-34B) on receiver 1-4701B or (PS-1-4741-33B) on the compressor suction send signals to open/close the unloader valve. | The pressure switches automatically send a signals to shut the valve when receiver pressure lowers to 95 psig or when discharge/casing pressure increases to 90 psig. | The unloader valve will cycle open/close to maintain pressure between receiver or suction pressure setpoints. This will prevent damage to system components from higher pressures and prevent the compressor from overheating due to lower water levels. |
| Solenoid operated unloader valve for the 1A compressor opens at 105 psig receiver pressure or when compressor discharge pressure reaches 120 psig. | Pressure switch (PS-1-4741-79A) on receiver 1-4701A or (PS-1-4741-70A) on the compressor discharge send signals to open/close the unloader valve. | The pressure switches automatically send signals to shut the valve when receiver pressure lowers to 95 psig or discharge pressure lowers to 102 psig. | The unloader valve will cycle open/close to maintain pressure between receiver or discharge pressure setpoints. This will prevent damage to system components from higher pressures. If the compressor remains in an unloaded condition for greater than 15 minutes, it will shut down. |
| Solenoid operated unloader valve for compressor 1/2 opens at 116 psig discharge pressure. | Pressure switch (PS-1/2-4741-70) sends signals to open/close the unloader valve. | The pressure switch automatically sends a signal to shut the valve when discharge pressure lowers to 95 psig. | The unloader valve will cycle open/close to maintain pressure between the discharge pressure setpoints. This will prevent damage to the system components from higher pressures. If the compressor remains in an unloaded condition for greater than 15 minutes, it will shut down. |



NRC COPY #1

#73



# LOSS OF INSTRUMENT AIR

A. SYMPTOMS

1. Alarms

   a. UNIT 1A (UNIT 2) (UNIT 1B) INST AIR LOW PRESSURE.

   b. UNIT 1A (UNIT 1B) (UNIT 2) (UNIT 1/2) AIR DRYER BYPASS VALVE OPEN.

   c. UNIT 1 (UNIT 2) SERVICE AIR BACKUP VALVE OPEN.

   d. SCRAM VALVE AIR SUPPLY LOW PRESSURE.

   e. TORUS VACUUM RELIEF VLV 20A (20B) NOT CLOSED.

2. Decreasing instrument air pressure.

B. AUTOMATIC ACTIONS

1. Service air backup valve, AO-4799-221, auto open.
2. Low flow feedwater regulator valve, AO-643, lockup.
3. Control rods drift in.
4. SDV vents and drains close.
5. Reactor Building to torus vacuum breakers, 1601-20A and B, fail open.
6. RWCU pumps trip on low flow.
7. Condensate make-up valves fail closed.
8. Condensate reject valves fail open.
9. **For Unit 1**, Condensate pump minimum flow valve fails open.
10. **For Unit 2**, Condensate pump minimum flow valve fails closed.

C. IMMEDIATE OPERATOR ACTIONS

1. Start any available instrument and service air compressors.

**74** **ID: SR-4700-K09** **Points: 1.00**

Unit 1 has experienced a total loss of TBCCW.

The Instrument Air compressors are protected against this failure by a trip on:

A. cooling water LOW flow.

B. high pressure outlet HIGH air temperature.

C. cooling water HIGH temperature.

D. cooling water LOW pressure.

Answer: B

**Question 74 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #74 (RO/SRO) |
| System ID: | 9807 |
| User ID: | SR-4700-K09 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 3.25 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | QOA 4700-02 |
| User Text: | 300000K4.03 |
| User Number 1: | 2.80 |
| User Number 2: | 2.80 |
| Comment: | New question. Lower. High outlet air temp outlet is the only one that is a trip. |

#74



UNIT 1A AND 1/2 INSTRUMENT AIR COMPRESSOR TRIPS

| Trip/Cause/Indication | Action |
|---|---|
| 1. Motor overload/excessive current/control room alarm | 1. Verify if trip is at Bus Feeder Breaker or at local compressor motor starters.<br>2. Determine cause of trip and reset overloads. |
| 2. Low oil pressure/20 psig/control room alarm, local alarm light at compressor | 1. Check compressor lube oil level.<br>2. Determine cause of trip and reset at compressor. |
| 3. High oil temperature/U-2 and 1/2 160°F/1A 175°F/control room alarm, local alarm light at compressor | 1. Check cooling water (TBCCW) available to compressor motor starters.<br>2. Determine cause of trip and reset at compressor. |
| 4. L.P. outlet air high temperature/425°F/ control room alarm, local alarm light at compressor | 1. Check cooling water (TBCCW) available to compressor.<br>2. Determine cause of trip and reset at compressor. |
| 5. H.P. inlet air high temperature/145°F/ control room alarm, local alarm light at compressor. | 1. Check cooling water (TBCCW) available to compressor.<br>2. Determine cause of trip and reset at compressor. |
| 6. H.P. outlet air high temperature/425°F/ control room alarm, local alarm light at compressor | 1. Check cooling water (TBCCW) available to compressor.<br>2. Determine cause of trip and reset at compressor. |

APPROVED
SEP 0 9 1997



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**75** **ID: SR-0001-K23** **Points: 1.00**

QGA 200-5, "HYDROGEN CONTROL," primary containment pressure control path, directs the primary containment to be vented.

The procedure directs the operator to vent via the torus as the preferred method vice via the drywell.

Venting the primary containment via the torus will:

A. Allow a more rapid reduction in primary containment pressure than venting from the drywell

B. Allow better control of the release rate due to the sizing of the path's piping and valves.

C. Minimize chugging due to loss of non-condensibles from the drywell atmosphere.

D. Reduce the levels of radioactivity released as it passes through the water in the torus.

Answer: D

**Question 75 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #75 (RO/SRO) |
| System ID: | 5405 |
| User ID: | SR-0001-K23 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 2.75 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | QGA 200-5, pg. 7 |
| User Text: | 500000EK3.06 |
| User Number 1: | 3.10 |
| User Number 2: | 3.70 |
| Comment: | STA ILT.10035: 80495 Bank question. Memory. Water in the Torus will "scrub" the air as it passes through, reducing release rates. Piping is the same size for the DW and Torus, so no affect. Chugging was a concern for sprays in the old rev of the QGAs. |

H. Vent and purge instructions

1. Each of the six blocks contains vent and purge instructions. The instructions specify:

   a. Allowable release rates.

   b. Isolations that may be bypassed.

   c. Vent location.

   d. Purge location.

   e. Purge method (air or nitrogen).

2. Directions to "vent/purge the drywell" or "vent/purge the torus" refer to the volume from which hydrogen or oxygen is to be removed, not the actual vent lineup.

   a. The drywell can be vented directly, or through the torus. The preferred lineup is normally through the torus to scrub the drywell atmosphere.

   b. The torus can be vented only if torus water level is below 30 ft. Above this elevation, all vent paths are submerged.

3. QCOP 1600-13 provides guidance on selecting the appropriate lineup. Factors considered include:

   a. Torus water level.

      1) At 30 ft., the torus vent is submerged.

      2) Higher torus water levels require higher drywell pressures to clear the downcomers.

   b. Containment pressure

      1) Venting the drywell through the torus may not be possible at low containment pressures or high torus water levels, since the pressure may be insufficient to clear the downcomers.

**76** **ID: SR-0203-K15** **Points: 1.00**

Given the following plant conditions:

- The reactor has just scrammed from 100% power caused by a loss of off-site power and a Loss of
 Coolant Accident.
- Both Emergency Diesel Generators started but did NOT close on to their respective busses.
- Reactor pressure is being controlled automatically by relief valves.
- Reactor power is 0%.
- Reactor water level is -49 inches and decreasing at 10 inches per minute.
- RCIC is injecting at 400 gpm.
- HPCI started and then tripped and is unavailable.
- Drywell pressure is 2.0 psig and slowly increasing at 0.5 psi per minute.

Which one of the following actions describes the Automatic Depressurization System (ADS) response, assuming NO operator action is taken?

A. Will NOT automatically initiate.

B. Automatically initiates in 60 seconds.

C. Automatically initiates in 110 seconds.

D. Automatically initiates in 570 seconds.

Answer: A

**Question 76 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #76 (RO) |
| System ID: | 8169 |
| User ID: | SR-0203-K15 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 3.00 |
| Time to Complete: | 4 |
| Point Value: | 1.00 |
| Cross Reference: | LIC-0203, pg. 3, 4 |
| User Text: | 203000K3.03 |
| User Number 1: | 4.20 |
| User Number 2: | 4.30 |
| Comment: | LORTB 124784 Bank question. Higher. With a loss of off-site power and a failure of the EDGs to load on their busses, the ECCS pumps do not have a power supply, so they will not start. One of the requirements for ADS to auto blowdown is a Low pressure ECCS pump running. |

#76



## II. COMPONENT DESCRIPTION

A. Electromatic Relief/Power Operated Relief Valves (PORVs on U-2)

1. The electromatic relief valves/PORVs are designed to prevent over-pressurizing the vessel or lifting the safety valves. They are also designed to relieve pressure rapidly to the pressure reset value or to allow the Low Pressure Coolant Injection (LPCI) System and the Core Spray System to function.

2. The relief valves are sized to prevent lifting the safety valves during a specific transient. The transient the relief valves are designed to protect against is:

   a. The turbine trips from full power, and

   b. The bypass valves fail to operate, and

   c. The reactor scrams from a closure of the turbine stop valves.

**Q: The ERV's/PORVs are si[z] to prevent lifting the safety valves during a specific transient. What transient is this?**

**A: The turbine trips from full power, and the bypass valves [f] to operate, and the reactor scrams from the closure of the turbine stop valves.**

3. Four electromatic relief valves/PORVs are located in the drywell, one each on Main Steam Lines C and D, and two on Main Steam Line B, upstream of the flow restrictors. The valves are actuated by energizing a 125 vdc solenoid assembly. Three methods of actuation are used:

**S/R-0203-EK014**
**Q: Where are the relief valve[s] physically located?**

**A: The 3B and 3E reliefs are located on MSL B, the 3C relie[f] is located on MSL C, and the 3[D] relief is located on MSL D.**

   a. Pressure switches (2201(2)-5 rack):

| Valve | B | C | D | E |
|---|---|---|---|---|
| Opening Setpoint | 1115 | 1115 | 1135 | 1135 |
| Closing Setpoint | 1070 | 1070 | 1090 | 1090 |

****S/R-0203-EK007a**

   b. A manual demand (keylock switch).

   c. An ADS initiation signal.

      1) High Drywell Pressure (2.5 psig), and

      2) Low-Low RWL (-59"), and

****S/R-0203-EK007b**

3) 110 second timer timed out, and

4) CS or RHR pump running > 100 psig

- OR -

1) Low-Low RWL (-59"), and

2) CS or RHR pump running > 100 psig, and

3) 8.5 minutes timer timed out.

4. Electromatic Relief Valve Control (Figure 0203-1)

**Show Figure 0203-1**

a. Each relief valve has a control switch with 3 positions:

****S/R-0203-EK007a**
****S/R-0203-EK021**

<u>Auto</u>: An ADS signal or exceeding high pressure setpoint will actuate the valve.

<u>Off</u>: The valve will actuate on an ADS signal only.

<u>Man</u>: This position opens the relief by directly energizing the valve solenoid.

**Q: What signal will cause the ERV's to open with their cont switches in OFF?**
**A: ADS signal only.**

b. When the 203-3B and 203-3C open automatically due to an ADS signal or high pressure, a 14.5 seconds delay is activated which prevents the valves from automatically reopening for 14.5 seconds after closing.

**S/R-0203-EK013**
**S/R-0203-EK028**
****S/R-0203-EK020**

This time delay allows the tailpipe vacuum breakers to cycle, ensuring water hasn't been drawn up into the line as a vacuum is being formed when the steam in the line condenses. The subsequent reopening of a relief valve when the tailpipe is partially filled with water could over-pressurize the relief line and/or result in structural damage to the suppression pool when this slug of water is blown into the suppression pool.

**Q: The 3C ERV is cycled ope then closed and the green and amber lights come on. What does the amber light indicate?**
**A: 10 second inhibit timer actuated.**

During these 14.5 seconds, manual actuation is physically possible; however, procedure cautions direct the operator not to actuate the valves for 14.5 seconds after closing. A light labeled INHIBIT is illuminated during these 14.5 seconds to warn the operator of the 14.5 second limitation.

**Q: What is the purpose of this interlock?**
**A: Prevent possible damage to the tail pipe/suppression pool b warning operator not to open RV for 14.5 seconds.**

The inhibit time delay was changed to 14.5 seconds, from 10 seconds, to account for valve stroke time from full open to full closed.

**Q: Can the relief valve be opened manually?**
**A: Yes.**

**REF. ISC 96-001E**

**77** **ID: SR-2300-K28** **Points: 1.00**

Operation of HPCI below 2200 rpm should be minimized because:

A. the introduction of water into the turbine is very likely at low speed.

B. it may result in unstable system operation.

C. the pump will be in Run Out flow conditions.

D. the min. flow valve will NOT receive an open signal with the turbine below 2200 rpm.

Answer: B

**Question 77 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #77 (RO) |
| System ID: | 2011 |
| User ID: | SR-2300-K28 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 3.00 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | LN-2300,pg 8, QCOP 2300-6 |
| User Text: | 206000G.2.1.32 |
| User Number 1: | 3.40 |
| User Number 2: | 3.80 |
| Comment: | ILT.04272 (77090) Bank question. Memory. Operation below 2200 RPM will cause unstable system operation. Run out condition requires high flow. Min flow is not the concern at this point. Intro of water is dependent on reactor water level. |

C. Lubricating and Control Oil System (Figure 4)

SR-2300-K14d
SR-2300-K15c
**N-2300-K14
**N-2300-K15

1. Purpose

The control and lubricating oil system provides oil for the lubrication of the speed reducer, turbine bearings, and main pump bearings. It also supplies oil to the thrust bearing wear detector, positioning units for the stop valve and control valve, and the hydraulic speed and trip control. Operation of the turbine below 2200 rpm should be avoided due to unstable operation and equipment damage.

**Show Figure 4, HPCI Turbine Oil System,**
**SR-2300-K28a**

2. Oil Tank

**SR-2300-K14d(1)**
**SR-2300-K14d(6)**
**SR-2300-K15c(6), (7)**
****N-2300-K14d(1), (6), (8)**
****N-2300-K15c(6), (7)**
**SR-2300-K14d(8)**

An oil tank is located in front of the HPCI turbine and extends under the front standard. The top of the tank serves as the operating floor for the turbine. A flexible connection between the front standard and the tank allows oil to return to the tank. The oil tank has a capacity of approximately 1000 gallons. An internal tank heater maintains tank temperature between 104°F and 110°F. Temperature is controlled by a thermostat (TS2) located in the panel to the right of the front standard. Oil temperature indicating switches are available throughout the system to provide temperature indication locally and in the control room.

3. Oil Filter

**SR-2300-K14d(7)**
**SR-2300-K15c(5)**
****N-2300-K14d(7)**
****N-2300-K15c(5)**

A duplex strainer is located in the low pressure discharge after the oil cooler. This filter is physically located in the pit by the front standard. The function of the filter is to remove impurities form the oil before reaching the bearings.

4. Main Oil Pump (MOP)

**SR-2300-K14d(4)**
**SR-2300-K15c(3)**
**SR-2300-K14d(5)**
**SR-2300-K15c(4)**
****N-2300-K14d(4), (5)**

The MOP provides the total oil requirements of the turbine hydraulic and lubricating system over the normal operating speed range (2000-4000 rpm). The MOP is a dual, positive-displacement gear pump driven by the turbine shaft. The MOP has a low pressure and a high pressure section.

**Q: Since the main oil pump is an attached pump, how is lubrication supplied to the turbine during startup and shutdown.**

The low pressure section provides oil at 55 psig to the turbine speed and trip controls, the thrust bearing wear detector, the turbine bearing lube oil (at a pressure of 10 psig, maintained by a pressure reducing valve), and the HPCI main pump bearings and speed reducer (at 20 psig, maintained by a pressure reducing valve).

**A: AOP is provided to auto start when a HPCI initiation signal is present and oil pressure is low. The pump may also be locally or remotely started during a manual start of the HPCI turbine.**
**N-2300-K15c(4)

#77

F.4. (cont'd)

## CAUTION

System operation below 5000 gpm should be minimized to limit cycling of the Turbine Exhaust check valve.

Operation of HPCI turbine with torus temperature above 140°F should be avoided because it may result in equipment damage due to inadequate lube oil cooling. (H.4.a.)

Operation of HPCI turbine below 2200 rpm should be avoided because it may result in unstable system operation and equipment damage.

ab. **IF** HPCI discharge flow adjustment is required, **THEN adjust** flow using one of the following methods:

(1) **Adjust** FIC 1(2)-2340-1, HPCI FLOW CONTROLLER setpoint with FIC 1(2)-2340-1, HPCI FLOW CONTROLLER in AUTO **AND** MOTOR SPEED CHANGER at HSS (High Speed Stop) position. ______

(2) **Place** FIC 1(2)-2340-1, HPCI FLOW CONTROLLER to MANUAL position. ______

(a) **Adjust** manual adjustment lever to desired flow **AND maintain** MOTOR SPEED CHANGER at HSS (High Speed Stop) position. ______

(3) **IF** FIC 1(2)-2340-1, HPCI FLOW CONTROLLER appears to have failed, **THEN adjust** HPCI flow using the MOTOR SPEED CHANGER control switch. ______

(4) **IF** HPCI suction is from CCST **AND** an initiation signal is **NOT** present, **THEN open** MO 1(2)-2301-15, TEST RTN VLV. ______

(a) **Throttle open** MO 1(2)-2301-10, TEST RTN VLV to maintain desired reactor water level. ______

**78** **ID: SR-1400-K26** **Points: 1.00**

A LOCA on unit 2 resulted in the following:

| | |
|---|---|
| Drywell pressure | 8.0 psig and steady |
| Reactor water level | -120 inches and lowering |
| Reactor pressure | 400 psig and lowering |

The "A" Loop of Core Spray is NOT running.

Concerning the "A" Loop of Core Spray, the ANSO should:

A. manually start the "A" Core Spray pump immediately and open the MO 1-1401-25A valve when reactor pressure reaches 325 psig.

B. place the 1A Core Spray pump in pull to lock.

C. wait for reactor pressure to drop below 325 psig and verify Core Spray auto initiates and manually open the MO 1-1401-25A valve.

D. wait for reactor pressure to drop below 325 psig and verify Core Spray auto initiates and injects.

Answer: A

**Question 78 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #78 (RO) |
| System ID: | 9814 |
| User ID: | SR-1400-K26 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 2.50 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | LIC-1400, pg. 13 |
| User Text: | 209001A4.05 |
| User Number 1: | 3.80 |
| User Number 2: | 3.60 |
| Comment: | New question. Higher. Core Spray pumps should have auto initiated at > 2.5 psig in the Drywell. When reactor pressure < 325 psig, they will auto inject. With CS pumps not running, you would not expect the injection to happen automatically either. Expectations are to take an auto action that does not happen. OP-AA-101-111, R. 0, pg.6; QCOA 1400-01, R.9 |

B. Automatic Functions

1. Initiation

SR-1400-K07

| Device/Setpoint | Logic | Bypass/Reset | Response |
|---|---|---|---|
| The Core Spray systems A & B will initiate upon receiving the following conditions:<br>- High DW pressure @ 2.5 psig. PS-1(2)-1001-90A/B/C/D<br>- Low RPV level @ -59" LIS 1-0263-72A/B/C/D* for 8.5 minutes.<br>- Low RPV level and low reactor pressure @ 325 psig. PS-1(2)-0263-52A/B.<br>-NOTE-LO RX Press by itself WILL NOT cause an initiate signal. | The initiation logic is arranged in the following manner.<br>HI DW press and LO RPV level are 2/2 once OR 1/2 twice. (one signal will not cause initiation. two signals may cause initiation. three will always cause initiation.) LO RPV pressure operates in conjunction with LO RPV level such that either A OR B contact closing coupled with the necessary LO RPV level will give an initiation signal. | The initiation signal (111A/112A) is reset by the initiating condition signal being reset. There is no bypass for the logic itself. It can be defeated by taking the individual components out of automatic operation. | Upon initiation, both core spray pumps start and its corresponding minimum flow bypass valve opens. The pump suction valve opens and the test bypass valve closes. The CS system is now running with the minimum flow valve open, recirculating to the torus. |
| The Core Spray systems A & B will inject upon receiving the <325 psig reactor pressure signal PS-1(2)-0263-52A/B. | The logic is arranged such that either contact closing will cause both CS systems to inject (provided the initiate signal is already present). | The signal will reset when the initiating condition is clear and reset. | Upon reactor pressure lowering to below 325 psig, MO-24(NO) and 25(NC) receive open signals. As MO-25 opens, the CS pump injects into the reactor vessel. As pump flow rate increases, the minimum flow bypass valve closes to direct full flow to the vessel. |

2. * On U2, the signals come from instruments LT-2-263-23 A/B/C/D via relays 2-260-K3A/B/C/D

D.4. **Notify** Shift Manager to classify event as a possible E-Plan condition and **initiate** E-Plan as necessary.

## E. DISCUSSION

E.1. Automatic initiation signals for Core Spray are:

a. Drywell pressure ≤ 2.43 psig.

**OR**

b. Reactor water level for Unit 1 below ≥ -56.78 inches and for Unit 2 below ≥ -55.2 inches **AND** reactor pressure ≥ 306 and ≤ 342 psig.

**OR**

c. Reactor water level for Unit 1 remains below ≥ -56.78 inches and for Unit 2 remains below ≥ -55.2 inches for more than 530 seconds.

E.2. **WHEN** an injection signal is present, **THEN:**

a. MO 1(2)-1402-24A/B, CS PMP INBD DISCH VLV, is interlocked open and can **NOT** be closed.

b. MO 1(2)-1402-25A/B, CS PMP OUTBD DISCH VLV, should be allowed to reach the full open position before an attempt is made to throttle the valve, to minimize the potential for a breaker trip.

## F. ATTACHMENTS

None.

## G. REFERENCES

### G.1. Technical Specifications:

a. TS 3.3.5.1, Emergency Core Cooling System (ECCS) Instrumentation.

b. TS 3.5.1, ECCS - Operating.

c. TS 3.5.2, ECCS - Shutdown.

### G.2. P&IDs:

a. M-36 (M-78), Diagram of Core Spray Piping.

#71



4.6.1. **REPORT** to the Unit Supervisor.

4.6.2. **OPERATE** the plant in accordance with approved procedures, and within the Limiting Conditions for Operation of the Technical Specifications to ensure the reactor is operated in a safe, conservative, and efficient manner at all times.

NOTE: The RO's immediate actions to stabilize the plant during transient conditions take priority over verbalization to the Unit Supervisor. If possible, verbalization should be accomplished to inform the Unit Supervisor of actions being taken.

1. During transient conditions, the RO may perform immediate operator actions of abnormal procedures from memory, while verbalizing actions being taken to the Unit Supervisor.

2. Subsequent actions taken during transient conditions will be based on direction of the Unit Supervisor per the applicable procedure(s).

4.6.3. **MAINTAIN** an active Reactor Operator's license.

4.6.4. One RO on each unit <u>SHALL</u> be designated the Unit RO and <u>SHALL</u> be "at the controls" (as defined by each station).

1. **ENSURE** applicable Technical Specification time clocks are entered and exited and associated action requirement completed as appropriate based on the scope of the work.

2. **MONITOR** the reactor and **ENSURE** reactor operation remains within established bands.

3. **MONITOR** all assigned control room panels, and **NOTIFY** the Unit Supervisor regarding unusual or unexpected conditions.

4. **MAINTAIN** cognizance of the activities and work impacting the unit, and the work of the assist RO(s) assigned to the unit.

5. **COORDINATE** and/or **PERFORM** necessary reactivity changes on the unit during the shift.

6. **SHUTDOWN** the reactor when the RO determines the safety of the reactor is in jeopardy or when operating parameters exceed any of the reactor protection circuit setpoints and automatic shutdown does **<u>not</u>** occur.

7. Manually **INITIATE** safety systems' automatic actions when operating parameters exceed the systems' automatic initiation setpoints and automatic initiation does **<u>not</u>** occur.

#79



# CORE SPRAY SYSTEM AUTOMATIC INITIATION

## A. SYMPTOMS

A.1. Possible alarms:

a. 901(2)-3 Panel.

(1) A-4 CORE SPRAY PUMP RUNNING

(2) G-15 REACTOR VESSEL LOW LOW LEVEL

(3) H-15 REACTOR VESSEL LOW PRESSURE

(4) G-4 DRYWELL HIGH PRESSURE

b. 901(2)-5 Panel.

(1) B-10 CHANNEL A REACTOR LOW LOW LEVEL

(2) B-15 CHANNEL B REACTOR LOW LOW LEVEL

(3) D-11 PRIMARY CONTAINMENT HIGH PRESSURE

A.2. 1(2)A/B CS PMP breaker closed indicating lights lit.

## B. AUTOMATIC ACTIONS

B.1. Both CS PMP 1(2)A **AND** 1(2)B start.

B.2. **IF** reactor pressure is less than 325 psig, **THEN** MO 1(2)-1402-25A/B, CS PMP OUTBD DISCH VLV, opens.

## C. IMMEDIATE OPERATOR ACTIONS

### CAUTION

**IF** Core Spray pumps are running without an injection, test or minimum flow path, **THEN** RV 1(2)-1402-28A/B, 1(2) CORE SPRAY DSCH HDR RV, will lift diverting torus water to Reactor Building Equipment Drain Tank.

C.1. **IF** initiation signal is **NOT** valid, **THEN** **pull to lock** both Core Spray pumps.

NRC COPY #1

**79** **ID: SR-1100-K26** **Points: 1.00**

Unit 2 has experienced an ATWS. ~~SYS A~~ 2A SBLC is electrically DOS,
Reactor power is ~ 20%.
The Unit Supervisor has directed SBLC injection into the RPV.
The NSO has positioned the SBLC initiation switch to the SYS 1 & 2 position.

What is the expected response and what should be done if the expected response does NOT occur? Per Hard Card.

A. Both squib valves should fire; SYS 2
Place the initiation switch to th~~e OFF position~~.

B. Both squib valves should fire;
Place the initiation switch to the SYS 2 & 1 position.

C. One squib valve should fire; SYS 1
Place the initiation switch to th~~e OFF position~~.

D. One squib valve should fire;
Place the initiation switch to the SYS 2 & 1 position.

how is this discriminating easily eliminated due to stem info SYS 1 & 2 position

credible?

Answer: ~~B~~ D

## Question 79 Details

why (?) high level

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #79 (RO) |
| System ID: | 1879 |
| User ID: | SR-1100-K26 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 2.00 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | QCOP 1100-02, R. 8 |
| User Text: | 211000A2.08 |
| User Number 1: | 4.10 |
| User Number 2: | 4.20 |
| Comment: | 76958 Modified question. Higher. Answer is correct due to both squibs firing should extinquish the continuity lights for both squibs and selection of 2 & 1 activates both squib valves again. Off would not activate any firing of squib valves. In 1 & 2 both squib valves are energized to fire. QCOP 1100-08, R. 9 |

ok

#79



# INJECTION OF STANDBY LIQUID CONTROL

## A. PURPOSE

The purpose of this procedure is to provide the steps necessary to inject the Standby Liquid Control (SLC) System into the Reactor.

## B. DISCUSSION

B.1. The QGA Procedure will direct the injection of a specific percentage of the Storage Tank and then direct securing the SLC pumps. **IF** a LOCA is in progress, **THEN NOT** all Sodium Pentaborate injected will remain in the Reactor and the injection of SLC should continue until the SLC tank reaches 0%.

B.2. The attachment to this procedure can be prepared and used as a Hard Card in accordance with CWPI-NSP-OP-1-5.

## C. PREREQUISITES

C.1. SLC system in standby lineup per QCOP 1100-01. ____

C.2. Shift Manager or the Unit Supervisor has determined that SLC injection is required **OR** SLC injection is required in accordance with procedures **OR** SAMG. ____

C.3. **IF** the Reactor cavity is flooded, **THEN** direct installation of the Fuel Pool to Canal Gate to prevent diffusion of Boron concentration to fuel pool volume. ____

## D. PRECAUTIONS

D.1. Do **NOT** allow SLC Storage Tank level to decrease below 0% to prevent damage to SLC pumps.

D.2. The Reactor Water Cleanup System will be isolated upon initiation of the SLC pumps.

# E. LIMITATION AND ACTIONS

E.1. The SLC storage tank is considered a low quality water injection source. **IF NOT** required by QGA procedures to inject low quality water, **THEN** prior to implementing this procedure as a means of RPV level control, **attempt** to inject with high quality water.

E.2. The neutron absorber solution (Boron) should be retained in the core until control rods have been repaired and inserted.

E.3. Avoid diluting the Reactor water containing the Boron as long as the control rods are withdrawn.

# F. PROCEDURE

F.1. **Inject** SLC by selecting either SYS 1 & 2 **OR** SYS 2 & 1 with keylock switch A AND B PUMP SELECT. ____

## CAUTION

Do **NOT** allow Storage Tank level to go below 0% to prevent damage to SLC pumps due to loss of suction.

F.2. **Verify** the following for indication of SLC system injection:

## NOTE

SQUIB valve continuity lights may still be lit following firing due to fragmenting of the firing mechanism internals and the possibility of a continuity path still existing. The combination of the following indications below will verify system injection into the Reactor vessel.

a. SQUIB A **AND** SQUIB B continuity lights are OFF. ____

b. FLOW light is ON indicating flow to Reactor. ____

c. Reactor Water Cleanup System isolates. ____

#79

F.2. (cont'd)

d. LI 1(2)-1140-2, TANK LEVEL is decreasing. ____

e. PI 1(2)-1140-1, PMP DISCH PRESS reads ≥ Reactor pressure. ____

f. Annunciator H-6, STANDBY LIQ SQUIB VALVE CIRCUIT FAIL is ON. ____

g. Neutron flux level decreasing. ____

F.3. **IF** indications do **NOT** show system injection, **THEN place** keylock switch A AND B PUMP SELECT to the position opposite the one initially selected, either SYS 1 & 2 **OR** SYS 2 & 1 **AND verify** injection indications in above step. ____

F.4. **IF** reactor recirculation pumps have been tripped per QGA procedure, **THEN leave** recirculation pumps off. ____

F.5. **IF** reactor recirculation pumps can be operated, **THEN operate** at least one recirculation pump to provide better mixing. ____

F.6. **IF** a LOCA is in progress, **THEN** the injection of SLC should continue until the SLC tank reaches 0%. ____

F.7. **IF** a LOCA is **NOT** in progress, **THEN continue** SLC injection until otherwise directed by the QGA procedures. ____

F.8. **WHEN** SLC injection is complete, **THEN:**

a. **Place** keylock switch A AND B PUMP SELECT to OFF. ____

b. **Return** SLC system to standby per QCOP 1100-01. ____

## G. ATTACHMENTS

G.1. Attachment A: Standby Liquid Control.

**1** **ID: 81117** **Points: 1.00**

Given the following information:

The unit was operating at full power when an instrument air line break caused the outboard MSIVs to go closed.
The reactor failed to scram and attempts to drive rods have been unsuccessful.
The Unit Supervisor has determined that SBLC injection is necessary.
The control switch was operated in the SYS 1&2 position.
The pump running lights on the 901-5 panel are lit.
The squib valve continuity lights are lit.
The flow indicating light on the 901-5 panel is NOT lit.
Pump discharge pressure is 1460 psig.
Reactor Pressure is currently 1025 psig.

Based on these indications what is the next action the operator should take?

A. Place the SBLC Initiation switch to the OFF position and back to the SYS 1&2 position

B. Take the SBLC Initiation switch to the OFF position and attempt to start the system locally.

C. Begin SBLC injection via the RWCU system.

D. Place the SBLC Initiation switch in the SYS 2&1 position.

Answer: D

**Question 1 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | ILT.11576 : NO TOPIC |
| System ID: | 6026 |
| User ID: | 81117 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 0.00 |
| Time to Complete: | 1 |
| Point Value: | 1.00 |
| Cross Reference: | |
| User Text: | 211000K4.03 |
| User Number 1: | 4.10 |
| User Number 2: | 4.10 |
| Comment: | By taking the control switch to the opposite direction the secondary or backup squib primers are fired. |

Is this supplied to Bus 28?

**80** **ID: SR-1100-K23** **Points: 1.00**

A fire has occurred on Bus 23. The NSO de-energized the bus and dispatched the fire brigade.

The 1/2 EDG did NOT autostart and has NOT been given a manual start signal. ← supply Bus

The NSO manually scrammed the reactor but no rod movement resulted.
No other operator action has been taken.
The US has ordered SBLC injection.

After the NSO positions the keylock switch A AND B SELECT to SYS 1 & 2, ~~the SQUIB A continuity light~~ SYS A Pump will be __________(1)______________ and the ~~SQUIB B continuity light~~ SYS B pump ~~will~~ be _(2)_ ~~due to the squib valve firing~~.

A. (1) ON ~~due to the loss of power from Bus 28;~~
(2) ON

B. (1) ON ~~due to the loss of power from Bus 28;~~
(2) OFF

C. (1) OFF ~~due to the loss of power from Bus 28;~~
(2) OFF

D. (1) OFF ~~due to the squib valve firing;~~
(2) ~~OFF~~ ON

potential double jeopardy with Q #79 (keep this change #79)

Answer: C

**Question 80 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #80 (RO) |
| System ID: | 6088 |
| User ID: | SR-1100-K23 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 3.50 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | LIC-1100, pg. 14 |
| User Text: | 211000K6.03 |
| User Number 1: | 3.20 |
| User Number 2: | 3.30 |
| Comment: | (81179) Bank. Higher. The A squib light will be off due to the loss of Bus 28 (squib relay is fed from 28-1A). The B squib light will be off due to system initiation. *Do not use with question 11637* QCAN 901(2)-5 H-6, R. 3; QCOP 1100-02, R. 8; FIG 6500-02, R. 2 |

#80

| Content/Skills | Activities/Notes |
|---|---|
| C. Power Supplies<br>1. SBLC Pump "A" is powered from MCC 18-1A(28-1A).<br>2. SBLC Pump "B" is powered from MCC 19-1(29-1).<br>3. Each injection valve and monitoring circuit is powered from its respective 480/120V AC breaker control power transformer. The 1(2)-1106A valve receives its power from the "A" pump breaker control power MCC 18-1A (28-1A) while the 1(2)-1106B valve receives its power from the "B" pump breaker control power MCC 19-1 (29-1). This means that if the "A" pump was Out-of-Service, the 1(2)-1106A would never fire and the continuity light should be extinguished. The same would hold true for the "B" pump and valve. | **SR-1100-K17**<br>****SR-1100-K23** |

#80



**DESCRIPTION** STANDBY LIQUID SQUIB VALVE CIRCUIT FAILURE

| STANDBY LIQ SQUIB VALVE CIRCUIT FAIL |
|---|

**SETPOINT**

**Actual:**

1. No current Squib Valve A Continuity Monitor.

OR

2. No current Squib Valve B Continuity Monitor.

**Tech Specs:** None.

**SENSOR**

1. 1(2)-1140-3A, Continuity Monitor Squib Valve A Panel 901(2)-5.
2. 1(2)-1140-3B, Continuity Monitor Squib Valve B Panel 901(2)-5.

## A. AUTOMATIC ACTIONS

None.

## B. OPERATOR ACTION

1. Verify SBLC pumps NOT injecting to Reactor if not needed:
   a. Verify SBLC pump indicating lights OFF.
   b. Verify FLOW indicating light OFF, indicating no flow to Reactor.
   c. IF SBLC pump is inadvertently running, THEN place 1A AND 1B PUMP SELECT switch to OFF.
   d. IF SBLC pump will NOT stop with switch in OFF, THEN open pump breaker at:
      (1) SBLC pump A: Rx Bldg. MCC 18-1A (28-1A), cubicle F-4(E-3).
         (a) For Unit 2 only: MCC 28-1A, cubicle E-1.
      (2) SBLC pump B: Rx Bldg. MCC 19-1 (29-1), cubicle A-4(A-4).
2. Verify alarm is valid AND determine which SBLC system is alarming by:



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## E. LIMITATION AND ACTIONS

E.1. The SLC storage tank is considered a low quality water injection source. **IF NOT** required by QGA procedures to inject low quality water, **THEN** prior to implementing this procedure as a means of RPV level control, **attempt** to inject with high quality water.

E.2. The neutron absorber solution (Boron) should be retained in the core until control rods have been repaired and inserted.

E.3. Avoid diluting the Reactor water containing the Boron as long as the control rods are withdrawn.

## F. PROCEDURE

F.1. **Inject** SLC by selecting either SYS 1 & 2 **OR** SYS 2 & 1 with keylock switch A AND B PUMP SELECT. ____

**CAUTION**

Do **NOT** allow Storage Tank level to go below 0% to prevent damage to SLC pumps due to loss of suction.

F.2. **Verify** the following for indication of SLC system injection:

**NOTE**

SQUIB valve continuity lights may still be lit following firing due to fragmenting of the firing mechanism internals and the possibility of a continuity path still existing. The combination of the following indications below will verify system injection into the Reactor vessel.

a. SQUIB A **AND** SQUIB B continuity lights are OFF. ____

b. FLOW light is ON indicating flow to Reactor. ____

c. Reactor Water Cleanup System isolates. ____

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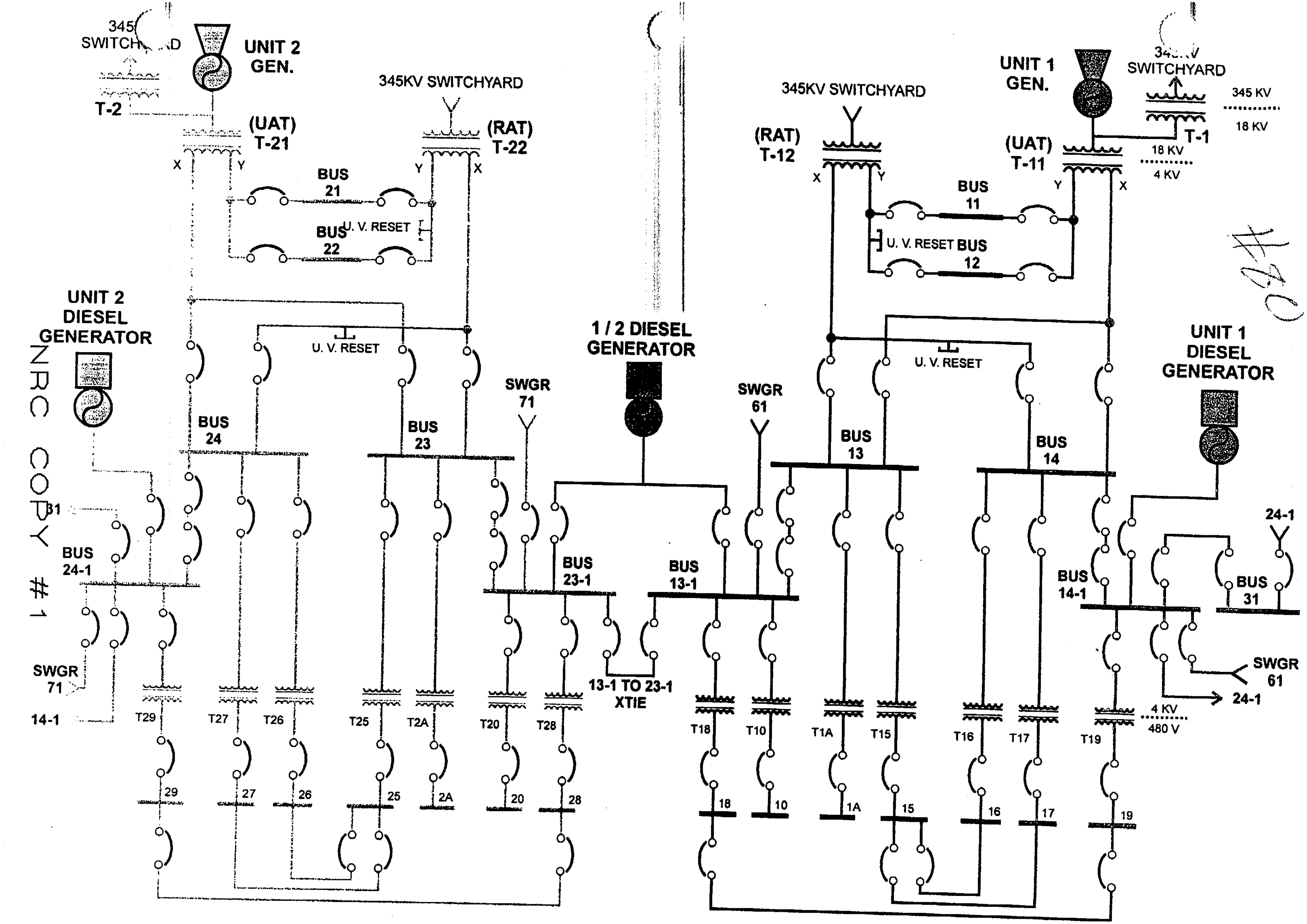


| Figure 6500-02 | Revision: 2 |
|---|---|
| 4 KV / 480 VAC DISTRIBUTION | |

81 ID: SR-0701-K28 Points: 1.00

Which statement below best describes the reason the drywell grating is removed and the carousel locked in place prior to withdrawing the SRM's and IRM's?

A. To keep the drive mechanisms from impinging on the grating/carousel.

B. An interlock prevents SRM/IRM withdrawal with the grating in place.

C. To allow access for maintenance to work on the drives if necessary.

D. To prevent access in case the detectors overtravel out.

Answer: A

**Question 81 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #81 (RO) |
| System ID: | 1780 |
| User ID: | SR-0701-K28 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 2.50 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | LIC-0701, R. 5 pg 26 |
| User Text: | 215003G2.1.32 |
| User Number 1: | 3.40 |
| User Number 2: | 3.80 |
| Comment: | ILT.04032 (76859) Bank. Lower. Grating must be removed to prevent damage to drive mechanisms. No interlocks. |

When testing the retract permit interlock/rod block during a refuel outage, the drywell grating is opened and the CRD overhaul platform (carousel) is locked in place. This is to prevent damage to the detector drive mechanisms:

1) When retracting either SRM or IRM detectors.

2) By the overhaul platform rotating with the detector drive mechanisms fully withdrawn.

B. Abnormal Operations

**SR-0701-K22
**SR-0701-K26

1. SRM Insert Or Withdraw Failure (QOA 0700-01)

   a. Malfunction of the SRM detector insert/retract mechanism can be caused by either a mechanical or electrical malfunction. A mechanical failure is defined as any failure of the drive motor, flexible drive shaft, drive mechanism, or the detector drive cable and detector. Electrical failures are defined as any failure of the control switches, control relays, or power buses. Power bus failure can be caused by fuse fatigue, fuse overload, or circuit breaker trip.

   Corrective actions for mechanical failures are confined to inside the drywell, while some electrical failures corrective actions can be performed outside the drywell. These type of failures can normally be corrected by the replacement of a fuse or resetting a breaker. Failure of and SRM drive could result in a rod block depending on where the detector is stuck and the reactor power level. The SRM channel can be bypassed at the 901(2)-5 panel by the Bypass "JOYSTICK".

2. Loss of SRM Flux Indication (count rate fails low/high)

   a. In the REFUEL mode, two SRM's shall be operable during core alterations; one in the quadrant where fuel and control rods are being moved and one in an adjacent quadrant. The SRM shall be inserted to the normal operating level and shall have the minimum count rate of 3 cps or .7 cps with a signal-to-noise ratio of at least 20:1 with all rods fully inserted except when:

**82** **ID: SR-0702-K26** **Points: 1.00**

Unit 2 is starting up with IRM's on range 4 and IRM 17 bypassed.
You receive a half scram on RPS A and the IRM High alarm (902-5 A5) comes in.
On the apron section for 902-5 the IRM 13 High and HIGH HIGH lights are lit.
The indication on the Recorder and on the drawer around back are pegged high for IRM 13.

Based on this information, you should: ~~with IRM 17 bypassed~~ but still can

leave IRM 13 as is, cannot bypass same time with IRM 17

A. reset the 1/2 scram and continue the startup.

B. bypass IRM 13, reset the 1/2 scram and continue the startup. and forced on 1 IRM 13

C. bypass IRM 13 but leave "A" RPS 1/2 scram inserted ~~because it does NOT have 3 operable IRM inputs~~.

D. discontinue the startup because there are NOT enough IRM inputs.

Answer: B

need to enhance other distractors to note how IRM 13 cannot be bypassed etc. and at least have one more dist with bypass IRM 13

**Question 82 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #82 (RO) |
| System ID: | 9789 |
| User ID: | SR-0702-K26 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 3.00 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | LIC-0702, PG.3,TS 3.3.1.1 |
| User Text: | 215003K6.05 |
| User Number 1: | 3.10 |
| User Number 2: | 3.20 |
| Comment: | New question. Higher. IRM 13 and 17 are on different channels, so there are still 3 operable IRMs on each channel, the minimum required for a startup. IRM 13 must be bypassed in order to reset the 1/2 scram. |

The 8 IRMs are located in the core as seen in Figure 0700-2-5. Four IRM channels are assigned to each RPS channel as follows:

| IRM Channel | RPS Channel |
|---|---|
| 11 | A |
| 12 | A |
| 13 | A |
| 14 | A |
| 15 | B |
| 16 | B |
| 17 | B |
| 18 | B |

Note: An "IRM HI-HI" trip from any IRM channel will cause a half scram in the assigned RPS channel; this is discussed further under the trip unit section below.

**Show Figure 0700-2-5**

**SR-0702-K14**
**SR-0702-K15**

B. DETECTOR DRIVE

1. The IRM drive unit (Figure 0700-2-6) is identical to the SRM drive unit. The axial arrangement and limits (Figure 0700-2-7) are identical to those for the SRM detector drives. (See SRM Presentation ILT-0700-1)

2. Drive Control

   a. As in the case of the SRM, the relays are located in the same chassis as the preamplifier.

   b. The IRM control switches on the 901(2)-5 panel (Figure 0700-2-8) have the same general location as the SRM switches.

   c. The IRM control switches comprise one pushbutton/indicator per IRM channel; the pushbutton operates identically to the SRM channel selection pushbuttons. (See SRM Presentation ILT-0700-1)

**Show Figure 0700-2-6 & 7**

**Q: What conditions result in automatic insertion of the IRM detectors?**
**A: SRM/IRM detectors position display switch is de-selected <u>and</u>**

**Individual IRM select switches are in the selected condition <u>and</u>**

**The reactor scrams**

**Q: How fast does the detector travel when being withdrawn or inserted?**
**A: 3 ft/min**

**Show Figure 0700-2-8**

**Q: How long does it take for the detector to travel from "full out" to "full in" on a full reactor scram signal?**
**A: Approximately 3 minutes. 9.5 ft/ 3ft/min = 3.167 minutes, or 3 minutes 10 seconds.**

**Q: What actually generates the "drive in" signal to the selected SRMs and IRMs on a scram?**
**A: When the Backup Scram Valves are energized, the "drive in" signal is generated (590-125A relay 4E-1467, Sh. 1 & 3, and 4E-1477).**



NRC COPY # 1

#82

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation

| FUNCTION | APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS | REQUIRED CHANNELS PER TRIP SYSTEM | CONDITIONS REFERENCED FROM REQUIRED ACTION D.1 | SURVEILLANCE REQUIREMENTS | ALLOWABLE VALUE |
|---|---|---|---|---|---|
| 1. Intermediate Range Monitors | | | | | |
| a. Neutron Flux-High | 2 | 3 | G | SR 3.3.1.1.1<br>SR 3.3.1.1.4<br>SR 3.3.1.1.5<br>SR 3.3.1.1.6<br>SR 3.3.1.1.7<br>SR 3.3.1.1.16<br>SR 3.3.1.1.17 | ≤ 121/125 divisions of full scale |
| | 5(a) | 3 | H | SR 3.3.1.1.1<br>SR 3.3.1.1.4<br>SR 3.3.1.1.5<br>SR 3.3.1.1.16<br>SR 3.3.1.1.17 | ≤ 121/125 divisions of full scale |
| b. Inop | 2 | 3 | G | SR 3.3.1.1.4<br>SR 3.3.1.1.5<br>SR 3.3.1.1.17 | NA |
| | 5(a) | 3 | H | SR 3.3.1.1.4<br>SR 3.3.1.1.5<br>SR 3.3.1.1.17 | NA |
| 2. Average Power Range Monitors | | | | | |
| a. Neutron Flux-High, Setdown | 2 | 2 | G | SR 3.3.1.1.1<br>SR 3.3.1.1.4<br>SR 3.3.1.1.5<br>SR 3.3.1.1.7<br>SR 3.3.1.1.9<br>SR 3.3.1.1.14<br>SR 3.3.1.1.17 | ≤ 17.1% RTP |
| b. Flow Biased Neutron Flux-High | 1 | 2 | F | SR 3.3.1.1.1<br>SR 3.3.1.1.2<br>SR 3.3.1.1.3<br>SR 3.3.1.1.5<br>SR 3.3.1.1.9<br>SR 3.3.1.1.10<br>SR 3.3.1.1.14<br>SR 3.3.1.1.16<br>SR 3.3.1.1.17<br>SR 3.3.1.1.18 | ≤ 0.56 W + 67.4% RTP and ≤ 122% RTP(b) |

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) 0.56 W + 63.2% and ≤ 118.4% RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

**83** **ID: SR-1300-K28** **Points: 1.00**

Which of the following is the reason to minimize the time the RCIC system is operating with pump flowrates of less than 400 gpm?

Flows less than 400 gpm may:

A. cause high turbine temperatures due to lack of flow for steam cooling of turbine components.

B. result in inadequate pump seal cooling water flow causing pump seal damage.

C. cause cycling of the minimum flow valve, routing water into the torus.

D. cause cycling of the turbine exhaust check valve, possibly causing damage to the exhaust piping.

Answer: D

**Question 83 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #83 (RO) |
| System ID: | 3994 |
| User ID: | SR-1300-K28 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 3.00 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | QCOP 1300-02, R. 22 |
| User Text: | 217000G2.1.32 |
| User Number 1: | 3.40 |
| User Number 2: | 3.80 |
| Comment: | LN.07309 (79082) Bank question. Memory. RCIC ops less than 400 gpm may cause exhaust valve cycling, damaging the exhaust piping. |

# RCIC SYSTEM MANUAL STARTUP (INJECTION/PRESSURE CONTROL)

## A. PURPOSE

The purpose of this procedure is to provide the steps necessary to manually start up the Reactor Core Isolation Cooling (RCIC) System for injection to the Reactor Vessel. Also, steps are provided for Reactor pressure control.

## B. DISCUSSION

B.1. The attachments to this procedure can be prepared and used as Hard Cards in accordance with OP-AA-101-403.

## C. PREREQUISITES

C.1. RCIC in standby lineup per QCOP 1300-01. ____

## D. PRECAUTIONS

D.1. Injection of RCIC into Reactor during power operation could cause a reactivity transient.

D.2. System operation below 400 gpm should be minimized to prevent the possible cycling of the Turbine Exhaust check valve.

D.3. RCIC operation below 2200 rpm should be minimized because it may result in unstable system operation and equipment damage. (H.8.c)

D.4. RCIC operation with torus pressure elevated above 25 psig may cause a RCIC trip due to a high exhaust pressure signal. (H.8.c)

D.5. RCIC operation with torus temperature above 140°F should be avoided because it may result in equipment damage due to inadequate lube oil cooling. (H.8.c)

D.6. Oxygen concentrations in the Torus may increase during extended RCIC operation. **IF** extended RCIC operation is anticipated, **THEN** **monitor** the oxygen concentration **AND** **operate** the nitrogen inerting system as necessary.



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**84** **ID: SR-0203-K26** **Points: 1.00**

The plant is operating at 100% power, steady state conditions, with all systems operable when the following alarms are received at the 901-3 panel:

E-14 ACOUSTIC MON SAFETY-RELIEF VALVES OPEN.
E-16 VALVE LEAK DET SYS TEMP.

Based on the information available, what should be the operators next response per QCOA 0203-01, FAILURE OF A RELIEF VALVE TO CLOSE OR RESEAT PROPERLY?

A. Cycle the affected valve key switch between MANUAL and AUTO.

B. Initiate suppression pool cooling.

C. Scram the reactor per QCGP 2-3.

D. Place the affected valve key switch to the OFF position.

Answer: D

**Question 84 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #84 (RO) |
| System ID: | 651 |
| User ID: | SR-0203-K26 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 2.50 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | QCOA 0203-01, R. 8 |
| User Text: | 218000A3.01 |
| User Number 1: | 4.20 |
| User Number 2: | 4.30 |
| Comment: | 75723 Bank question. Higher. These annunciators are an indication of a stuck open relief valve. The immediate operator action is to take the switch to OFF. If it does not close, then scram the reactor. |

#84



# FAILURE OF A RELIEF VALVE TO CLOSE OR RESEAT PROPERLY

## A. SYMPTOMS

A.1. Possible alarms:

a. Panel 901(2)-3

(1) D-13, ELECT RELIEF VALVE 3A/3B OPEN.

(2) E-13, ELECT RELIEF VALVES 3C/3D/3E OPEN.

(3) E-14, ACOUSTIC MON SAFETY RLF VALVES OPEN.

(4) E-16, VALVE LEAK DET SYS HIGH TEMP.

b. Panel 901(2)-4

(1) G-17, TORUS WATER HIGH TEMP.

A.2. Possible Relief Valve indications:

a. Relief Valve open indication on 901(2)-3.

b. Relief Valve Acoustic Monitor indicates valve open on Panel 901(2)-21, SAFETY/RELIEF VLV ACOUSTIC MONITORS.

c. Relief Valve discharge temperature does not return to value indicated prior to valve opening at recorder 1(2)-260-20, VLV LEAK AND CNMT AIR TEMP, on Panel 901(2)-21.

A.3. Possible affected system changes:

a. **IF** Relief Valve Surveillance QCOS 0203-03 in progress, **THEN** Bypass Valve position remains unchanged.

b. Turbine steam flow decrease.

c. Generator output decrease.

d. Torus water temperature increase.

## B. AUTOMATIC ACTIONS

None.

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**85** **ID: SR-0203-K08** **Points: 1.00**

Given the following plant conditions:

- Reactor vessel water level has just decreased to -59 inches.
- Reactor water level is continuing to decrease.
- Drywell pressure is 2.2 psig and steady.
- All systems are assumed to operate as expected.

Assuming no operator actions taken, how soon would the Automatic Depressurization System begin to depressurize the reactor?

A. Immediately

B. In 110 seconds

C. In 510 seconds

D. In 720 seconds

Answer: C

**Question 85 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #85 (RO) |
| System ID: | 732 |
| User ID: | SR-0203-K08 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 3.50 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | LIC-0203, pg. 4 |
| User Text: | 218000K5.01 |
| User Number 1: | 3.80 |
| User Number 2: | 3.80 |
| Comment: | ILT.01607 (75805) Bank question. Higher. Just hit -59 inches, but do not have 2.5 psig in the Drywell, so the 8.5 minute (510 second) timer will start. |

#85



3) 110 second timer timed out, and

4) CS or RHR pump running > 100 psig

- OR -

1) Low-Low RWL (-59"), and

2) CS or RHR pump running > 100 psig, and

3) 8.5 minutes timer timed out.

4. Electromatic Relief Valve Control (Figure 0203-1)

**Show Figure 0203-1**

a. Each relief valve has a control switch with 3 positions:

****S/R-0203-EK007a**
****S/R-0203-EK021**

Auto: An ADS signal or exceeding high pressure setpoint will actuate the valve.

Off: The valve will actuate on an ADS signal only.

Man: This position opens the relief by directly energizing the valve solenoid.

**Q: What signal will cause the ERV's to open with their conti switches in OFF?**
**A: ADS signal only.**

b. When the 203-3B and 203-3C open automatically due to an ADS signal or high pressure, a 14.5 seconds delay is activated which prevents the valves from automatically reopening for 14.5 seconds after closing.

**S/R-0203-EK013**
**S/R-0203-EK028**
****S/R-0203-EK020**

This time delay allows the tailpipe vacuum breakers to cycle, ensuring water hasn't been drawn up into the line as a vacuum is being formed when the steam in the line condenses. The subsequent reopening of a relief valve when the tailpipe is partially filled with water could over-pressurize the relief line and/or result in structural damage to the suppression pool when this slug of water is blown into the suppression pool.

**Q: The 3C ERV is cycled open then closed and the green and amber lights come on. What does the amber light indicate?**
**A: 10 second inhibit timer actuated.**

During these 14.5 seconds, manual actuation is physically possible; however, procedure cautions direct the operator not to actuate the valves for 14.5 seconds after closing. A light labeled INHIBIT is illuminated during these 14.5 seconds to warn the operator of the 14.5 second limitation.

**Q: What is the purpose of this interlock?**
**A: Prevent possible damage to the tail pipe/suppression pool b warning operator not to open RV for 14.5 seconds.**

The inhibit time delay was changed to 14.5 seconds, from 10 seconds, to account for valve stroke time from full open to full closed.

**Q: Can the relief valve be opened manually?**
**A: Yes.**

**REF. ISC 96-001E**

**86** **ID: SRN-1603-K12** **Points: 1.00**

Given:

- Rx Power: 100%
- Rx water level: +32" and rising slowly
- Rx Pressure: 815 psig and decreasing
- No operator actions have been taken.

MSIVs should indicate ________ and the Primary Containment O2 Analyzer valves should indicate _________.

A. closed; closed

B. open; open

C. closed; open

D. open; closed

Answer: C

**Question 86 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #86 (RO) |
| System ID: | 9790 |
| User ID: | SRN-1603-K12 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 3.00 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | LN-1603, pg. 16, 17 |
| User Text: | 223002A4.04 |
| User Number 1: | 3.50 |
| User Number 2: | 3.60 |
| Comment: | Modified from ILT.00970. Higher. This is a Group 1 signal. The O2 valves are part of Group 2. |

#/6

3. Automatic Functions

1. Initiation

SR-1603-K10(a)

| Name/Purpose | Device/Setpoint/ Logic | Bypass/Reset | Response |
|---|---|---|---|
| GROUP I CONTAINMENT ISOLATION | **Low main steam pressure:** PS-X-261-30 A D, 825 psig (in RUN Mode) after a 0.5 second delay, switch contact opens.<br>**High main steam tunnel temperature:** TS-X-261-15A D, -16A D, -17A D, -18A D, 200 F, switch contact opens.<br>**High main steam line flow:** DPS-X-261-2A H, -2J N, -2P, R, S, 140% rated main steam line flow, switch contact opens.<br>**Low-Low reactor water level:** LT-X-263-57A & B, LT-X-263-58A & B, -59" reactor water level, trip unit relay contact opens. | Low main steam pressure: Bypassed when mode switch not in run; High main steam tunnel temperature, High steam line flow, and low - low reactor water level trips are never bypassed and auto reset. The Group I isolation seals - in and is reset with the MAIN STM ISOL RESET switch on the 90X-5 panel. | The following valves close: MSIVs 203-1A, B, C, D, 203-2A, B, C, D. Main Steam Drain Valves 220-1, 2, Reactor Water Sample Valves 220-44, 45. |

**NOTE: For the Main Steam Line High Flow Group I Isolation, there are four D/P switches per steam line. For a single steam line two D/P switches input to Division I PCIS Logic and the other two D/P switches input to Division II PCI Logic.**

#96



| Name/Purpose | Device/Setpoint/ Logic | Bypass/Reset | Response |
|---|---|---|---|
| GROUP II CONTAINMENT ISOLATION | **Low reactor water level:** LT-X-263-57A & B, LT-X-263-58A & B, 0" reactor water level, trip unit relay contact opens. **High drywell radiation:** RM-X-2419A & B, 100 R/hr in the drywell, trip relay de-energizes opening relay contact. **High drywell pressure:** PS-001-8A D, 2.5 psig drywell pressure, switch contact opens. | The Group II isolation signals are never bypassed. The low reactor water level and high drywell pressure auto reset. The drywell high radiation monitor is reset by depressing the lighted red push button. This must be done before the Group II isolation can be reset. The push buttons are located on RM-X-2419A & B. The Group II isolation seals - in and is reset with the ISOL VALVE RESET switch on the 90X-5 panel. | The following valves close: RHR Valves 1001-20, 21, 47, 50, 29 (in SDC), Containment Purge and Exhaust Valves 1601-21 thru 24, 55 thru 63, Trip Drywell/ Torus Purge Fans, Drywell Floor and Equipment Drain Discharge Valves 2001-3, 4, 15, 16, Oxygen Analyzer Valves 8801-A, B, C, D; 8802-A, B, C, D; 8803, 4, Tip Purge Valve 700-743, Automatic Tip withdrawal is initiated followed by ball valve closure. The following pumps are tripped: RB floor drain sump and RB equipment drain tank. |
| GROUP III CONTAINMENT ISOLATION | **Low reactor water level:** LT-X-263-57A & B, LT-X-263-58A & B, 0" reactor water level, trip unit relay contact opens. | The Group III isolation signal is never bypassed. The Low reactor water level auto resets. The Group III isolation seals - in and is reset with the ISOL VALVE RESET switch on the 90X-5 panel. | The following valves close: 1201-2, 5, 80. NOTE: The RWCU pumps will trip from the 1201-2 or 5 valves not full open, or the 1201-80 closed. |

**SR-1603-K10(b)**

**NOTE: Tip withdrawal occurs regardless of TIP mode (auto or manual). The pump trips are not required by Tech Specs.**

**NOTE: Drywell/torus purge fans trip on signal from SBGTS start logic.**

**SR-1603-K10(c)**

**80 valve not required by Tech Specs.**



NRC COPY #1

Quest #86

# EXAMINATION ANSWER KEY

Modified ?

**1** **ID: 75444** **Points: 1.00**

Given:

- Rx Power: 40%
- Rx water level: +42" and stable
- Rx Pressure: 815 psig and decreasing
- No operator actions have been taken.

OLD Question

What automatic actions should have already occurred?

A. Main Turbine Trip only

B. Group 1 Isolation and Reactor Scram

C. Main Turbine Trip and Group 2 & 3 isolations

D. Group 1 Isolation only

Answer: B

**Question 1 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | ILT.00970 : Recognize failure of GR I on low pressure |
| System ID: | 372 |
| User ID: | 75444 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 0.00 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | |
| User Text: | 223002G12 |
| User Number 1: | 3.60 |
| User Number 2: | 3.40 |
| Comment: | |



NRC COPY #1

**87** **ID: SR-1000-K21** **Points: 1.00**

You have the following plant conditions:

| | |
|---|---|
| - Drywell pressure | ~~3.2~~ 2 psig |
| - Drywell temperature | 170 degrees F |
| - Torus pressure | 1.8 psig |
| - Torus temperature | 96 degrees F |
| - Reactor water level | +30 inches |

RPV press 300 psig

The plant has scrammed and QCGP 2-3 is being carried out.
The RHR system was in a normal lineup at the beginning of the transient a~~nd all automatic actions occurred as designed~~.

The Unit Supervisor orders Torus Cooling started on the "A" RHR Loop.

The RHR Loop "A" RHR SW START PERMISSIVE SWITCH 19 cannot physically be moved to the MANUAL OVERRIDE position.

~~Because of this~~, Containment temperatures will:

Too obvious that w/o RHR SW - temp will rise. which eliminates 2 dists A+B - due to temp decrease

A. decrease unless RPV Water Level reaches -59 inches.

B. decrease unless RPV Water Level reaches -191 inches.

C. increase unless the "B" RHR SW Pump is started.

D. increase unless the "B" loop of Torus Cooling is started.

Answer: ~~D~~ A

**Question 87 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #87 (RO) |
| System ID: | 9791 |
| User ID: | SR-1000-K21 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 3.00 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | LNF-1000, R. 7, pg. 27 |
| User Text: | 226001K3.02 |
| User Number 1: | 3.50 |
| User Number 2: | 3.50 |
| Comment: | New question. Higher. Drywell presssure > 2.5 psig trips the RHRSW pumps. RPV water level irrelevant. "B" RHRSW pump is on the "A" Loop also. |

| Device/Setpoint | Logic | Bypass/Reset | Response |
|---|---|---|---|
| LPCI INITIATION | | | |
| High drywell pressure (2.5 psig) (PS-1001-90A-D)<br>OR<br>Low-low RPV water level (-59") (LIS-263-72A-D)<br>AND<br>Low RPV pressure (325 psig) (PS-263-52A(B)<br>OR<br>Low-low RPV water level (-59") for 8.5 min (via ADS logic) | DW pressure: A or B AND C or D<br><br>RPV level and pressure: 72A or B AND C or D AND 52A or B<br><br>RPV level for 8.5 min: Relay 287-124A for "A" loop Relay 287-124B for "B" loop | Can be reset by depressing LPCI INITIATION RESET pushbutton when signals have cleared. | RHR pumps start, RHRSW pumps are interlocked off, MO-1001-16A/B receives an open signal for 60 sec., and the following valves are interlocked closed: MO 1(2)-1001-23A/B, 26A/B, 34A/B, 36A/B, and 37A/B. |

**N-1000-K08

2. Trips and Isolations

**SR-1000-K11
SR-1000-K12

| Purpose | Device Setpoint/Logic | Bypass/Reset | Response |
|---|---|---|---|
| Inhibit opening of MO-1001-47 & 50 valves if RPV pressure is above 100 psig to prevent overpressurizing low pressure piping. | PS-261-23A OR B sensing reactor pressure greater than 100 psig. | This signal is reset by using the ISOLATION VALVE RESET switch when Reactor pressure is less than 100 psig. | Valves cannot be opened if Rx pressure is > 100 psig and will auto close if Rx pressure rises above 100 psig. |

| Purpose/Response | Device Setpoint/Logic | Bypass/Reset |
|---|---|---|
| When Rx pressure is greater than 325 psig, then only one of the following valves may be opened at a time to prevent overpressurizing the low pressure piping: MO-1001-28A(B) or MO-1001-29A(B). If LPCI is injecting and Rx pressure rises above 325 psig, the 28A(B) & 29A(B) valves WILL NOT automatically close. | Limit switch on associated valves <u>AND</u> Rx pressure 325 psig (PS-1001-52A for 28A & 29A, PS-1001-52B for 28B & 29B). | Cannot be bypassed. |
| RHRSW pumps are interlocked off upon a LPCI Initiation signal | LPCI Initiation signal. (See above) | Can be bypassed by placing the RHRSW PUMP START PERMISSIVE keylock switch to "MANUAL OVERRIDE". |



NRC COPY # 1

**88** **ID: SR-0803-K21** **Points: 1.00**

The Unit One refueling platform is traveling in the reverse direction over the reactor core with the main hoist loaded.

What will happen if the REFUELING INTERLOCK CHECK pushbutton on the 901-28 panel fails in the depressed position?

A. Bridge trolley motion will be prohibited.

B. Bridge will NOT be able to move either forward or reverse.

C. Bridge will continue to travel towards the core.

D. Bridge reverse motion will stop.

Answer: D

**Question 88 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #88 (RO) |
| System ID: | 9794 |
| User ID: | SR-0803-K21 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 3.50 |
| Time to Complete: | 1 |
| Point Value: | 1.00 |
| Cross Reference: | QCFHP 0500-8, R. 10 |
| User Text: | 234000K3.03 |
| User Number 1: | 3.10 |
| User Number 2: | 3.80 |
| Comment: | Bank question. Higher. This pushbutton simulates a control rod withdrawn, which will cause bridge reverse motion to stop. |

Ques 85 (RO)



H.6. (cont'd)

---

## NOTE

**IF** a control rod is being withdrawn in MODE 5, **THEN** the control rod must be OPERABLE, including the associated accumulator, except those withdrawn per Technical Specification 3.10.5.

---

p. **Perform** one of the following:

(1) **IF** a Control Rod can be withdrawn, **THEN withdraw** one Control Rod one notch. ____

(a) **Verify** the withdrawn control rod has **NO** "full-in" indication. ____

(2) **IF** a Control Rod can **NOT** be withdrawn, **THEN depress** and **hold** the REFUELING INTERLOCK CHECK pushbutton in Panel 901(2)-28. ____

q. **Attempt** to move the Refueling platform over the Reactor. ____

(1) **Verify** bridge travel is interrupted. ____

(2) **IF** testing Unit 1(2) Refueling Platform, over Unit 1(2) Core, **THEN verify** BRIDGE REVERSE STOP #1 light comes ON. ____

(3) **IF** testing Unit 1(2) Refueling Platform, over Unit 2(1) Core, **THEN verify** BRIDGE FORWARD STOP #1 light comes ON. ____

(4) Verify ROD BLOCK INTERLOCK #1 light comes ON.

(5) Verify FUEL HOIST INTERLOCK light comes ON.

r. **Insert** the Control Rod withdrawn in step H.6.p. **OR release** the REFUELING INTERLOCK CHECK pushbutton. ____

F.2. Each of the required refueling equipment interlocks associated with the Reactor mode switch refuel position shall be demonstrated OPERABLE by performance of CHANNEL FUNCTIONAL TEST at least once per 7 days during in-vessel fuel movement. (J.1.a)

F.3. **IF** operating conditions do **NOT** allow the withdrawal of a Control Rod, (i.e., CRD system not available), **THEN** this condition can be simulated by depressing the REFUELING INTERLOCK CHECK button in Panel 901(2)-28 and holding it depressed.

F.4. **IF** the interlock checks for either the frame mounted **OR** monorail hoists are **NOT** performed, **THEN** an Equipment Status Tag must be placed on the control pendant stating that the interlock must be checked prior to using either hoist for core alterations.

F.5. The Zone Computer must be in the Bypass Mode when transferring items between pools **OR** lowering the Fuel Grapple past the top of the Fuel Storage Racks in the Fuel Storage Pools.

## G. PERFORMANCE ACCEPTANCE CRITERIA

G.1. Generally smooth and uninterrupted motion for the Refueling platform, trolley, and main hoist motion check is expected without Fault Lockouts. (H.3)

G.2. All lamps are expected to light up solid and generally bright. (H.3.d)

G.3. The Main Hoist Jam checks should produce Slack Cable Indication (H.4.c.), Grapple Engage Indication (H.4.e.), Hoist Jam Indication (H.4.f.) as appropriate.

G.4. During the Main Hoist unloaded checks, the Bridge Travel Stop #2 and Rod Block #2 is verified when the Reactor Mode Switch is in START-UP.

## H. PROCEDURE

H.1. **IF** the Control Rod accumulator low pressure interlock is giving a rod block, **THEN**:

a. **Verify** all rods in, except those withdrawn per Technical Specification 3.10.5.

b. **Administratively block** Control Rod withdrawal by taking Rod Motion Control Switch out of service.

H.6. (cont'd)

i. **Place** Fuel Grapple switch to ENGAGE position. ____

j. **De-energize** main hoist electrical power by pressing the System Stop Pushbutton. ____

(1) **Verify** mechanical brake integrity. ____

k. **Re-energize** the main hoist electrical power by pressing the System Start pushbutton. ____

l. **Raise** test weight to the Normal-Up position. ____

(1) **Verify** the Limit switch stops hoist raise motion. ____

(2) **Verify** the NORMAL-UP light comes ON. ____

m. **Raise** the test weight to the Back-Up Hoist Limit position by simultaneously operating the Hoist Override pushbutton and raise control. ____

(1) **Verify** the Limit switch stops hoist raise motion. ____

(2) **Verify** the BACK-UP HOIST LIMIT light comes ON. ____

n. **Lower** test weight to the NORMAL-UP position. ____

o. **Select** a Control Rod. ____

ref. At least two? ? 3 OK?

**89** and 1 **ID: SR-6100-K32** **Points: 1.00**

In Reactor Mode ~~1~~, 3 how many independent 345 KV lines must be available?

A. One

B. Two

C. Three

D. ~~Five~~ Four

Answer: B

OK Not Discriminating (Too simple!)

can elaborate more - diff allignments also modes RAT & Bus 13-1 & 14-1?

**Question 89 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #89 (RO) |
| System ID: | 3652 |
| User ID: | SR-6100-K32 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 3.00 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | QCOS 0005-08, R. 6 |
| User Text: | 262001K2.01 |
| User Number 1: | 3.30 |
| User Number 2: | 3.60 |
| Comment: | ILT.06799 : NO TOPIC 78738 Bank question. Memory. 2 345 lines are required to ensure 2 qualified circuits between offsite and onsite. |

#89



F.2. (cont'd)

d. **<u>IF</u>** <u>one</u> of the required AC **<u>OR</u>** DC electrical power distribution systems is inoperable, **<u>THEN</u> initiate** compensatory actions per Technical Specification 3.8.7, CONDITION A **<u>OR</u>** CONDITION B.

e. **<u>IF</u>** <u>one</u> or more required opposite unit AC **<u>OR</u>** DC electrical power distribution subsystems is inoperable, **<u>THEN</u> initiate** compensatory actions per Technical Specification 3.8.7, CONDITION C.

f. **<u>IF</u>** <u>two</u> or more electrical power distribution subsystems are inoperable that, in combination, result in a loss of function, **<u>THEN</u> enter** LCO 3.0.3 per Technical Specification 3.8.7, CONDITION E.

F.3. **<u>IF</u>** in MODE 4 or 5 **<u>OR</u>** when handling irradiated fuel assemblies in the Secondary Containment, **<u>THEN</u>**:

a. **<u>IF</u>** <u>one</u> required offsite circuit between the offsite transmission network and the onsite Class 1E distribution system is inoperable, **<u>THEN</u> initiate** compensatory actions per Technical Specification 3.8.2, CONDITION A **<u>AND</u>** suspend crane operations over the spent fuel storage pool.

b. **<u>IF</u>** <u>one</u> required Diesel Generator is inoperable, **<u>THEN</u> initiate** compensatory actions per Technical Specification 3.8.2, CONDITION B **<u>AND</u>** suspend crane operations over the spent fuel storage pool.

c. **<u>IF</u>** one or more required AC or DC electrical power distribution subsystems are inoperable, **<u>THEN</u> initiate** compensatory actions per Technical Specification 3.8.8, CONDITION A.

## G. PERFORMANCE ACCEPTANCE CRITERIA

G.1. **<u>IF</u>** Unit 1 is in MODE 1, 2, or 3, **<u>THEN</u>**:

a. Two qualified circuits between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System is determined operable by verifying correct breaker alignments and indicated power availability as follows:

(1) At least <u>two</u> 345 KV lines available.

(2) Unit 1 Reserve Auxiliary Transformer capable of energizing Bus 13-1 **<u>AND</u>** Bus 14-1.

**90** **ID: SRN-6900-K05** **Points: 1.00**

Unit One 125 VDC Battery Voltage is indicated ____________. It is measured ____________.

A. in the Battery Room;
at the charger output.

B. on the 901-8 panel;
directly from the battery

C. in the Battery Room;
directly from the battery.

D. on the 901-8 panel;
at the charger output.

Answer: B

**Question 90 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #90 (RO) |
| System ID: | 2627 |
| User ID: | SRN-6900-K05 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 3.50 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | LN-6900, pg 24 |
| User Text: | 263000A4.04 |
| User Number 1: | 3.20 |
| User Number 2: | 3.10 |
| Comment: | ILT.05526 (77709) Bank question. Lower. Measurement is taken directly off o fthe batteries. |

wrong K/A.
A402?
update.

# 90



C. Ground Detection Circuitry (continued)

| | Instrument/Control | Range/Function |
|---|---|---|
| 13. | 1(2) Non-ESS 250 VDC Ammeter | One provided for each unit. Range is –800A to +800A. |

SRN-6900-K05
**SRN-6900-K20b(2)

D. Control Room Instrumentation at 901(2)-8

**NOTE:** The following instrumentation is provided with a light bulb, setpoint pointer, and a voltage pointer. When battery voltage lowers to the setpoint pointer, the voltage pointer blocks the light and the alarm is received. If the light bulb burns out, the meter will not function and EM is required to change the bulbs due to the voltage (120VAC) involved.

SR-6900-K06

| | Instrument/Control | Range/Function |
|---|---|---|
| 1. | 1A(2A) 24/48 VDC Voltmeter | 0-100 VDC. Provided with 1A fuses. Monitors output directly from the 24/48 VDC battery. Provides annunciator input (A-6 BATTERIES UNDERVOLTAGE). |
| 2. | 1B(2B) 24/48 VDC Voltmeter | 0-100 VDC. Provided with 1A fuses. Monitors output directly from the 24/48 VDC battery. Provides annunciator input (A-6 BATTERIES UNDERVOLTAGE). |
| 3. | Unit 1(2) 125 VDC Voltmeter | 0-150 VDC. Provided with 1A fuses. Monitors output directly from the Unit 1(2) 125 VDC battery. Provides annunciator input (A-6 BATTERIES UNDER VOLTAGE). |
| 4. | Unit 1(2) 250 VDC Voltmeter | 0-300 VDC. Provided with 1A fuses. Monitors output directly from the Unit 1(2) 250 VDC battery. Provides annunciator input (A-6 BATTERIES UNDER VOLTAGE). |

**91** **ID: SR-1701-K24b** **Points: 1.00**

Unit One is operating at 100% power.
The "A" SJAE Radiation Monitor fails DOWNSCALE.

In addition to the alarm

What redundant protection is ~~provided~~ designed to auto close the offgas holdup valve?

A. ~~ONLY~~ An upscale signal from "B" SJAE Radiation Monitor.

B. None, the offgas holdup valve will close in 15 minutes.

C. None, the offgas holdup valve will close immediately.

D. A downscale OR upscale signal from "B" SJAE Radiation Monitor.

Either

Answer: A

**Question 91 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #91 (RO) |
| System ID: | 9799 |
| User ID: | SR-1701-K24b |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 3.00 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | LF-1701, pg. 12 |
| User Text: | 272000K4.01 |
| User Number 1: | 2.70 |
| User Number 2: | 2.80 |
| Comment: | New question. Lower. 2 upscale trips or an upscale and downscale trip are required to isolate offgas. |

c. Trip Auxiliaries Unit

The trip auxiliaries unit receives trip inputs from a downscale or gross upscale output from either monitor. Control logic in the auxiliaries unit is arranged so that two upscale trips or one upscale and one downscale trip are required to initiate a time delayed closure of the offgas holdup valve. The 15-minute time delay is half the time required for the gas to travel through the holdup volume without the recombiner in service. This allows the opportunity to evaluate the high radiation condition before valve closure without releasing radioactive gases to the atmosphere.

****SR-1701-K24b**

d. Interval Timer

The time delay is accomplished by use of an adjustable delay switch with knob on the interval timer. A manual switch (air ejector suction switch on 90X-7 panel) supplies the timer output to the offgas isolation. This switch is placed in "AUTO" during normal plant operation. Placing the switch to "CLOSED" will immediately isolate the offgas system. Two lights are available on the timer. Normally, one light is on. When the time starts, both lights go off. Upon failure of the normal light, the other light will come on.

**SR-1701-K05b(3)**
**SR-1701-K14b(4)**
**SR-1701-K15c(3)**
**SR-1701-K16c(2)**
**SR-1701-K05d**
**SR-1701-K20b(3)**

e. Offgas Vial Sampler System

The offgas vial sampler system is used in conjunction with the SJAE offgas radiation monitors. The system consists of two pumps, a sample vial with a hypodermic type connector, a vial positioner, four normally closed solenoid operated valves, manual valves and a control panel. The system allows offgas samples from A & B air ejectors and A & B recombiners to be removed for laboratory analysis during normal operation and shutdown conditions.

**SR-1701-K01b**
**SR-1701-K05b**
**SR-1701-K14b(5)**
**SR-1701-K15c(4)**
****SR-1701-K06**

#91



# ABNORMAL OFF GAS RADIATION

## A. SYMPTOMS

A.1. Alarms:

a. 901(2)-3 D-2, OFF GAS HI RADIATION.

b. 901(2)-3 C-2, OFF GAS HIGH HIGH RADIATION.

c. 901(2)-3 A-2, MAIN STM LINE HI RADIATION.

d. 901(2)-5 B-9, CHANNEL A MAIN STM LINE HI HI RADIATION.

e. 901(2)-5 B-16, CHANNEL B MAIN STM LINE HI HI RADIATION.

A.2. Significant increase in Off Gas or Main Chimney activity.

A.3. Significant increase in Reactor water activity or iodine content.

A.4. Significant increase in general plant background radiation.

## B. AUTOMATIC ACTIONS

B.1. **IF** two SJAE Rad Monitor high-high trips are received **OR** one high-high trip and one downscale trip is received, **THEN** the Off Gas Isolation 15 Minute Timer will start.

B.2. **IF** the Off Gas Isolation Timer is **NOT** reset within 15 minutes, **THEN** the Off Gas System will isolate:

a. AO 1(2)-5406, OG DISCH TO STACK OR VENT, closes.

b. **IF** Unit 1 **THEN**:

(1) AO 1-5408A, OFFGAS FILT DRN, closes.

(2) AO 1-5408B, HOLDUP PIPE DRN, closes.

c. **IF** Unit 2 **THEN** AO 2-5408, HOLDUP PIPE DRAIN AND FINAL FILT DRN, closes.

d. SO 1(2)-5437, U1(2) PRESS DRN TK OUTLET, closes.

**92** **ID: SR-5750-K26** **Points: 1.00**

One of the Unit 2 Refuel Floor Radiation Monitors indicates 150 mr/hr.

What is the expected plant response due to this and what action would be required? (Assume all automatic actions happen.)

A Reactor Building Vent isolation would occur on:

A. Both Units.
Manually start the 1/2A SBGT Train.

B. Unit 2 ONLY.
Verify Rx Bldg Vents isolated and investigate the cause of the High Radiation.

C. Unit 2 ONLY.
Manually start the 1/2A SBGT Train.

D. Both Units.
Verify Rx Bldg Vents isolated and investigate the cause of the High Radiation.

Answer: D

**Question 92 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #92 (RO) |
| System ID: | 9764 |
| User ID: | SR-5750-K26 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 3.00 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | LNF-5750 pg. 9 |
| User Text: | 288000A2.04 |
| User Number 1: | 3.70 |
| User Number 2: | 3.80 |
| Comment: | New question. Higher. An isolation signal from either unit will isolate reactor building ventilation on BOTH units. SBGT would auto start on the high rads. |

| 93 | ID: SRNLF-00-K09 | Points: 1.00 |
|---|---|---|

The ANSO takes 1B Core Spray to pull-to lock as directed by a surveillance procedure.

The 1B Core Spray Pump is Operable:

A. as soon as it is taken out of pull-to-lock.

B. as long as an operator is IMMEDIATELY available to return the switch to normal if needed.

C. since it was placed in pull-to-lock as directed by a procedure.

D. ONLY after a satisfactory operational test on 1B Core Spray.

Answer: A

**Question 93 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #93 (RO) |
| System ID: | 9765 |
| User ID: | SRNLF-00-K09 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 3.00 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | QAP 0300-02, R. 62 pg. 3 |
| User Text: | G.2.1.28 |
| User Number 1: | 3.20 |
| User Number 2: | 3.30 |
| Comment: | Bank question. Lower. Per QAP 0300-02, ECCS pumps are INOP when they are in PTL and are considered operable as soon as they are taken out of PTL. |

#93



(1) It may be necessary upon occasion to temporarily withdraw a system from operation by placing it in a manual or pull-to-lock mode. This should be done only when conditions are "stable and under control", or when it is apparent that continued operation would aggravate or worsen the plant condition. In all instances such action should be taken only after careful consideration, and it must be reviewed and approved by the licensed Senior Reactor Operator with the Unit Supervisor duties for that unit. It is not expected that such operations will be conducted for prolonged periods.

Whenever a system is withdrawn from operation as outlined, continuing surveillance of the relevant parameters must be maintained by a licensed Reactor Operator to assure the safe operation of the plant until the system can be restored to its normal operating mode or until it is no longer needed, as prescribed by the Technical Specifications.

(2) Placement of an ECCS component, with auto-start capability, in PULL-TO-LOCK, renders that component inoperable. Assignment of an Operator to the switch does not compensate for loss of auto-start capability. The decision must be reviewed and approved by the Senior Reactor Operator with the Unit Supervisor duties for that unit who determines that the placement of the switch in PULL-TO-LOCK will prevent other unwanted safety system challenges. <u>IF</u> all control rods are inserted to or beyond 04, <u>THEN</u> ECCS components may <u>NOT</u> be placed in PULL-TO-LOCK, as a means of limiting the rate of level increase, until the reactor level has increased to above the top of the active fuel. <u>IF</u> any control rod is <u>NOT</u> inserted to or beyond 04, <u>THEN</u> placement of ECCS components in PULL-TO-LOCK may be done in accordance with the QGA procedures regardless of vessel level.

#92



5. Power supply to the supply fans are:

   a. 1A (2A) - Bus 19 (29)

   b. 1B (2B) - Bus 18 (28)

   c. 1C (2C) - Bus 18 (28)

D. Emergency Dampers

1. There are four emergency air operated dampers that shut during emergency conditions, to prevent the release of contaminants to the environment. There are two dampers on the supply fan discharge duct and two on the exhaust fan inlet duct. The dampers auto close on any one of the following conditions:

   a. High drywell pressure (+2.5 psig).

   b. Low reactor water level (0 inches).

   c. High drywell radiation (100 R/hr).

   d. High refuel floor radiation (100 mR/hr).

   e. High radiation level in the Reactor Building vent exhaust duct (10 mR/hr).

   f. Rx Bldg vent exhaust or Refuel floor radiation detectors downscale.

   g. Low instrument air pressure at the damper (65 psig).

2. These four (4) emergency air operated dampers may also be secured in the closed position by use of a manual handwheel operator. The Hand wheels are located below the individual dampers on the Turbine Building 658'10" level on each side of the supply duct.

3. The isolation dampers are energized to open, and require air to open and air to close.

   a. A 4-way solenoid valve will port air to the top of the air operator to open the damper, when the solenoid is energized, and allow the underside of the air operator to vent to atmosphere.

**Refer to M-4-1(2)-85-47 (OTR 89-58). This modification installed inlet & outlet damper control switches and reset buttons on the 912-1 panel. Prior to this mod, these controls were only available on the 2251(2)-24X panel.**

**N-5750-K14a.(5)**
**SR-5750-K14a.(5)**

**N-5750-K15f.(2)**
**SR-5750-K15d.(2)**

**Refer to DVR 4-2-88-061, "Rx Bldg Vents Started Without Starting a Sample Pump" (OTR 89-112). This DVR is the result of an NSO starting a Reactor Building exhaust fan without having the particulate sampler turned on. The Unit NSO did not recognize the significance of having the "Rx Bldg Stack Monitor Low Flow" alarm up. This alarm is annunciated on the 90X-3 panel.**

**Prior to barrier fuel (when there were leakers). A typical release rate was 600 micro ci/sec, now a typical release rate for Unit One is 1 micro ci/sec and U-2 is not detecting any release.**

**Effluent air is also sampled to check for tritium. The air sample must be frozen and the frozen condensation is sampled for tritium. This must be done (frozen) because tritium emits a very low energy beta which would otherwise be undetectable.**

**Halogens are sampled weekly.**



NRC COPY #1

#92



# REACTOR BUILDING VENTILATION ISOLATION

A. SYMPTOMS

1. Alarms.

   a. RX BLDG 1 SPLY/EXH FAN TRIP, panel 912-5 A-1.

   b. RX BLDG 2 SPLY/EXH FAN TRIP, panel 912-5 A-4.

   c. RX BLDG 1 LOW DP, panel 912-5 C-1.

   d. RX BLDG 2 LOW DP, panel 912-5 C-4.

2. RX BLDG ISOL DAMPERS indicate closed on panel 912-1.

3. Local indicating lights for the RX BLDG ISOL DAMPERS indicate closed on local panels 2251-24X and 2252-24X.

B. AUTOMATIC ACTIONS

1. All eight Reactor Building isolation dampers close.

2. All Reactor Building supply and exhaust fans trip.

3. The Standby Gas Treatment System auto-starts.

C. IMMEDIATE OPERATOR ACTIONS

1. None.

D. SUBSEQUENT OPERATOR ACTIONS

1. Verify all AUTOMATIC ACTIONS have occurred.

2. Trip the fan control switches, for the fans that tripped, to prevent an auto start when power is restored or ventilation is reset.

3. Notify Radiation Chemistry that the Reactor Building ventilation has isolated.

4. Return the Reactor Building ventilation system to normal as soon as the cause is corrected. For resetting the isolation signal and to restart the fans, refer to QOP 5750-02, Reactor Building Ventilation System.

E. DISCUSSION

1. It is imperative that Reactor Building Ventilation be restored as soon as possible to ensure temperatures in the MSIV Room remain below the MAX NORMAL OPERATING TEMPERATURE limits.

(final)

NRC COPY #1

**94** **ID: SRLF-805-K10** **Points: 1.00**

Who has the specific responsibilities for approving the performance of each step during core alterations in accordance with the Nuclear Component Transfer List?

A. Shift Manager

B. Unit Supervisor

C. Nuclear Station Operator

D. Nuclear Engineer

Answer: C

**Question 94 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #94 (RO) |
| System ID: | 9817 |
| User ID: | SRLF-805-K10 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 2.50 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | QCFHP 0100-01, R. 18 |
| User Text: | G.2.2.34 |
| User Number 1: | 2.80 |
| User Number 2: | 3.20 |
| Comment: | Bank ILT.08228 : NO TOPIC user defined ID 79317 Bank question. Fundamental. The NSO is responsible for approving each move on the NCTL. |

#94



E.17. To prevent contamination of the Move Sheet, the original shall be kept in a designated, clean area. Copies will be made of the original as needed for use on the refuel floor and control room. All initials and signatures are to be transferred to the original by the end of the shift. The other copies may be discarded when no longer needed.

E.18. The SRO(L)/SRO License Holder will directly supervise the execution of steps in the Move Sheet.

E.19. **NO** more than <u>one</u> fuel assembly should be suspended above the fuel storage array. This assembly shall **NOT** exceed 24 inches height above the storage array to limit potential fuel damage if the assembly is dropped.

E.20. Steps of this procedure may be performed in any logical order and repeated as necessary per the discretion of the SRO(L)/SRO License Holder.

E.21. Movement of <u>any</u> object over irradiated fuel shall **NOT** take place unless Secondary Containment as defined by Technical Specifications is in effect.

E.22. The following individual responsibilities for refueling reactivity management have been established to ensure adequate supervision and control of core alterations. (Confirmatory Order 254/90-00105)

a. Unit Supervisor

(1) Maintains direct supervisory contact with reactor operations and evolutions controlled from the Control Room.

(2) Communicates to the Nuclear Station Operator (NSO) the requirements for procedural adherence, conservative response to abnormal reactivity events, and proper attitude toward reactivity controls.

(3) Has the authority to halt refueling operations as deemed necessary.

b. Nuclear Station Operator (NSO)

(1) Controls refueling activities which have the potential for affecting core reactivity by maintaining continuous communication with the refueling bridge.

(2) Verifies adequate count rate ($\geq$ 3 cps) on required SRM prior to each core alteration step.



NRC COPY #1

**95** **ID: NGET** **Points: 1.00**

The plant has experienced a transient.
Emergency exposure limits have been authorized.
Estimated dose will be 7 Rem TEDE.

Can an individual perform work to protect the main turbine from damage? If the individual can perform this work, what will be their emergency exposure TEDE limit?

A. No, limit is 5 Rem for repair work.

B. Yes, 25 Rem

C. Yes, 10 Rem

D. Yes, 50 Rem

place in ascending or descending order - Also, it depends on what the individual has as his present TEDE dose - i.e. if he already has 4 Rem

Answer: C

**Question 95 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #95 (RO) |
| System ID: | 9813 |
| User ID: | NGET |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 3.50 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | RP-AA-203 R. 2 |
| User Text: | G.2.3.4 |
| User Number 1: | 2.50 |
| User Number 2: | 3.10 |
| Comment: | New question. Higher. Limit is 10 REM TEDE for protecting valuable property. |

#95



4.4.11. **SUBMIT** a written report of the PSE assigned dose to the individuals involved within 30 days of the PSE.

4.4.12. The dose equivalent received from a PSE is always tracked separately from routine occupational exposure.

4.4.13. Once an exposure is authorized as a PSE, it **cannot** later be treated as a routine occupational exposure. It must be recorded as a PSE, and all the unique limitations, reporting, and record keeping requirements for PSEs shall apply.

4.5. Emergency Exposure Limits **(CM-1)**

4.5.1. Emergency exposure in excess of 25 rem TEDE is to be limited to once in a lifetime.

4.5.2. Emergency personnel are to be informed "before the fact" of possible health effects at the anticipated exposure levels.

4.5.3. For the control of personnel exposures under emergency conditions, **LIMIT** an individual's dose equivalent per activity as follows:

TABLE 2 – EMERGENCY EXPOSURE LIMITS (REM)

| TEDE | LDE | SDE | TODE | ACTIVITY |
|---|---|---|---|---|
| 10 | 30 | 100 | 100 | Protecting Valuable Property |
| 25 | 75 | 250 | 250 | Lifesaving or Protection of Large Populations |
| > 25 | > 75 | >250 | > 250 | Lifesaving or Protection of Large Populations to Workers Fully Aware of the Risks Involved |

4.5.4. Emergency exposures shall be voluntary on the part of the involved individual.

4.5.5. **CONSULT** the Emergency Plan Implementing Procedures regarding approval to exceed NRC exposure limits.

5. **DOCUMENTATION**

5.1. **RETAIN** completed exposure authorizations, including Attachments 1, 2, and 3, in accordance with the station records management program. This records program will include appropriate controls for storage and preservation.

**96** **ID: SR-6900-K22** **Points: 1.00**

Unit 2 was operating at 100% core thermal power when the NSO reports a loss of annunciator power.
Reactor power is lowering.
Further observation reveals that the indicating lights have been lost for the 1C & 1D and 2A & 2B RHR pumps as well as Buses 21, 23, 23-1, 25, and 28.

Why is Unit 2 Reactor Power lowering?

A. The 2B recirc pump is coasting to a stop due to loss of MG Set oil pumps.

B. The 2A recirc pump breaker is tripped due to loss of control power.

C. The 2A Recirc Pump is coasting to a stop due to loss of MG Set oil pumps.

D. The 2B recirc pump breaker is tripped due to loss of control power.

Answer: C

**Question 96 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #96 (RO) |
| System ID: | 6222 |
| User ID: | SR-6900-K22 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 3.50 |
| Time to Complete: | 1 |
| Point Value: | 1.00 |
| Cross Reference: | QOA 6900-04, R. 21, pg. 1 |
| User Text: | 295004AA2.02 |
| User Number 1: | 3.50 |
| User Number 2: | 3.90 |
| Comment: | LN.11863 (81316) Bank question. Higher. Symptom 1,4, and 6 of QOA 6900-04 are given.QOA 6900-04 states that 2A Recirc Pump will coast to a stop due to loss of MG set oil pumps. D1 is wrong - 2B pump is unaffected D2 and D3 are wrong loss of control power does not cause pumps to trip. / ILT.11863 replaced redundant NLO.02974 |

#96

QOA 6900-04
Revision 21
**Continuous Use**

# TOTAL LOSS OF UNIT 2 125 VDC SUPPLY

A. SYMPTOMS

1. All annunciator windows on Control Room Panels 902-3,4,5,6,7 and 8 extinguish except "ANN DC POWER FAILURE."
2. 1B and 2A Recirculation pumps coast to a stop, but the MG sets do NOT trip because the oil pumps trip.
3. Unit 2 MVAR (reactive power) increases due to voltage regulator failure.
4. Loss of indication and control of breakers at Control Room Panels for Buses 12, 14, 16, 17, 19, 21, 23, 25 and 28.
5. Reactor Building isolation dampers 1-5741A,B and 1-5742A,B close.
6. Loss of indication and control of pumps in 1B and 2A RHR loops.
7. Inability to remotely start, trip, or control the Unit 1 Diesel Generator.
8. DC pilot valves of Unit 1 outboard and Unit 2 inboard MSIVs fail open as indicated by lack of illuminated lights on Panel 901(2)-61 in the Cable Spreading Room.
9. Possible loss of Secondary Containment due to failure of Rx/Turb Interlock Doors.
10. Panel 901(2)-74 alarms 109 and 110.

B. AUTOMATIC ACTIONS

1. **Unit 2 only**, half scram A channel and 1/2 PCIS Groups 1, 2, and 3 isolation.
2. Unit 2 Main Generator voltage regulator transfers to manual control (but can NOT be adjusted).
3. The 250V DC auxiliary oil pumps start for the 1B and 2A Recirculation pump MG sets.
4. Unit 1 HPCI logic transfers to Unit 1 125V DC.
5. Unit 1 Relief Valve controllers transfer to Unit 1 125V DC.
6. Portions of Unit 1B ADS logic transfer to Unit 1 125V DC.

**97** **ID: SR-1300-K20** **Points: 1.00**

Unit 1 has experienced a small line break LOCA.
The HPCI system is OUT-OF-SERVICE.
A LOSS of normal feedwater occurs.

The RCIC system auto initiates and INJECTS into the vessel.
RCIC operates for several minutes and then TRIPS.
Several minutes later RCIC restarts and injects into the vessel.

ASSUMING no operator action, what was the cause of the RCIC turbine trip?

A. Turbine overspeed.

B. Low pump suction pressure.

C. High turbine exhaust pressure.

D. High Reactor Water level.

Answer: D

**Question 97 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #97 (RO) |
| System ID: | 7877 |
| User ID: | SR-1300-K20 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 3.50 |
| Time to Complete: | 2 |
| Point Value: | 1.00 |
| Cross Reference: | QCOA 1300-02, R. 10 |
| User Text: | 295008AA1.05 |
| User Number 1: | 3.30 |
| User Number 2: | 3.30 |
| Comment: | 124490 Bank question. Higher. On a high reactor water level trip, RCIC will auto restart on low low reactor water level. All other RCIC trips require the operator to reset. |

#97



## E. DISCUSSION

E.1. RCIC auto-starts on Reactor low low level for Unit 1 ≤ -56.78" or Unit 2 ≤ -55.2".

E.2. RCIC isolation signals are:

a. RCIC Turbine area high temperature; ≤ 169°F.

b. RCIC steam line high flow; ≤ 175% of rated steam flow with a time delay of 3.2 ≤ t ≤ 8.8 seconds.

c. RCIC steam line low pressure; ≥ 54 psig.

E.3. RCIC Turbine trips are:

**NOTE**

Item E.3.a will close the TRIP THROTTLE VLV.

Item E.3.b through E.3.e will close MO 1(2)-1301-60, MIN FLOW VLV and MO 1(2)-1301-61, STM TO TURB VLV.

**IF** RCIC trip was caused by Reactor high level, item E.3.b., **THEN** RCIC will auto restart on a Reactor Vessel Low-Low Level Signal. All other trips require operator actions for resetting the trips.

a. Turbine overspeed; 5600 rpm. (Mechanical)

b. High Reactor Vessel water level; for Unit 1 ≤ 54.23 inches or Unit 2 ≤ 50.34 inches.

c. Low pump suction pressure; 15 inches Hg vacuum.

d. High RCIC Turbine exhaust pressure; 25 psig.

e. Automatic RCIC Isolation.

## F. ATTACHMENTS

None.

**98** **ID: SRN-EVAC-K09** **Points: 1.00**

The Control Room has been evacuated due to a fire per QCOA 0010-05, CONTROL ROOM EVACUATION.

Operators will be dispatched to monitor Reactor Water level from the:

A. 2201(2) - 5 and 2201(2) - 6 Instrument Racks AND the ATWS level indicators in the Aux Electric Room.

B. 2201(2) - 5 and 2201(2) - 6 Instrument Racks.

C. ATWS level indicators in the Aux Electric Room.

D. ~~ATWS~~ level indicators in the Cable Spreading Room.

are there 2 ATWS lvl indicators?

Analog trip

~~not addable~~ available

Is there [illegible] ? - If no such [illegible] - [illegible] in on [illegible] on station Blk out?, procedure

Answer: B

**Question 98 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #98 (RO) |
| System ID: | 9763 |
| User ID: | SRN-EVAC-K09 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 3.50 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | QCOA-0010-05, R. 21, p. 3 |
| User Text: | 295016AA2.02 |
| User Number 1: | 4.20 |
| User Number 2: | 4.30 |
| Comment: | New question. Memory. On control room evac, operators are directed to monitor level from the 5 and 6 racks. The aux electric room is directed for level monitoring on a station blackout. |

#98



NOTE

Unit Assist NSOs will dispatch NLOs to appropriate areas of the plant as needed.

e. NLO to Feedwater regulating valves.

f. NLO to Turbine Building main floor.

g. Any discharge from Radwaste should be terminated. This will free one Operator.

7. Establish communications between the stations listed above. The emergency telephones may be used for this purpose (reference procedure QOP 9000-05).

8. IF unable to scram the Reactor prior to Control Room evacuation, THEN proceed as follows:

a. As directed by the Shift Manager, the Reactor will be scrammed from the Auxiliary Electrical Room by manually tripping the circuit breakers in the RPS distribution panels which feed 901(2)-15 AND 901(2)-17 Panels.

(1) IF the Auxiliary Electrical Room is NOT habitable, THEN the Reactor may be scrammed by tripping the RPS MG sets at MCCs 18(28)-2 AND 19(29)-2 OR if RPS is on alternate power, 15-2(25-2).

9. The appropriate Unit Assist NSO is standing by listed panels to initiate an MSIV isolation from the Auxiliary Electrical Room by pulling the following fuses:

| **901(2)-40** | Panel | **901(2)-41** | Panel |
|---|---|---|---|
| Fuses: | 595-709A | Fuses: | 595-709B |
| | 595-710A | | 595-710B |
| | 595-711A | | 595-711B |

10. Monitor Reactor water level and pressure using the local indicators on 2201(2)-5 AND 2201(2)-6 Instrument Racks.

a. IF water level is high, THEN a Reactor Feed Pump can be tripped locally from Bus 11(21) OR 12(22). Control FWRVs locally using QCOP 0600-18.

99 ID: SR-7500-K20 Points: 1.00

Reactor Building Differential Pressure is 0.25" $H_2O$.
1/2 A SBGTS is operating at 4000 scfm for a monthly surveillance.
Reactor Building Ventilation failed such that all supply fans trip causing Reactor Building Differential Pressure to increase to 0.75" $H_2O$.

Predict the change in flow through the SBGTS.

A. Flow would decrease and remain at 3600 scfm due to increased Reactor Building Differential Pressure.

B. Flow would increase and remain at 4400 scfm due to the flow restricting orifice.

C. Flow would decrease initially then return to 4000 scfm due to action of the Flow Control Valve.

D. Flow would increase intially then return to 4000 scfm due to action of the Flow Control Valve.

Answer: C

**Question 99 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #99 (RO) |
| System ID: | 9785 |
| User ID: | SR-7500-K20 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 4.50 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | LF-7500, pg. 6 & 20 |
| User Text: | 295035EA1.02 |
| User Number 1: | 3.80 |
| User Number 2: | 3.80 |
| Comment: | New. Higher. SBGT would initially be taking a suction on an area at a lower pressure, resulting in lower flow. The flow contol valve will open to restore flow to 4000 scfm. |

2. The adsorber is made up of removable drawer assemblies which contain the charcoal filters. Each two-inch thick filter is composed of granular activated charcoal that is impregnated with potassium iodide. The adsorber is rated for a 99.9% efficiency for iodine removal.

J. Mixing Section

The mixing section is installed to allow a representative sample, of process flow, to be drawn. This allows a method of evaluating performance of train components. A flow control orifice installed in the mixing section will limit flow on a loss of instrument air to no more than 4400 scfm.

K. High Efficiency After Filter

1. The high efficiency after filter is provided to remove any activated carbon particles or dust that may be released from the carbon adsorbers.

2. It is identical in design to the high efficiency prefilter.

L. AO-1/2-7510 A(B), SBGTS Train Flow Control Valve

1. The flow control valve regulates SBGTS train process flow at a value that is preset on FIC 7541-7A on local panel 2212-29A(B). The FCV is normally full open, when the train is not operating, and then closes down to regulate flow when the train is started.

2. System flow is sensed by a flow element on the suction piping, upstream of the Train Inlet Damper.

3. AO-1/2-7510 fails open upon loss of its instrument air supply.

M. Crosstie Line

1. An eight inch crosstie line provides an interconnection between the SBGTS trains so that the operating trains fan can provide cooling air flow through the idle train for decay heat removal.

**Treating the charcoal adsorbers with potassium iodide increases their affinity for methyl iodides. Therefore, during SBGTS operation, the methyl iodides (radioactive) will remain in the adsorber and the potassium iodides (non-radioactive) will be given off and released out the chimney. In doing this, the methyl iodides will be delayed for a longer period of time ensuring more decay prior to release, thus less activity released to the environment.**

****SR-7500-K02**
****SR-7500-K03**
**SR-7500-K14**
**SR-7500-K15**
****SR-7500-K20**
****SR-7500-K21**
****SR-7500-K23**

**Flow setpoint adjustment is made locally.**

**Q: Is cooling flow through the idle train par of the 4000 scfm flow that is measured by the FE?**
**A: No.**

**Refer to SOER 88-1 which discusses various industry loss of air casualties.**

c. High Reactor Building Ventilation Exhaust Radiation (3 mr/hr).

d. High Refuel Floor Radiation (100 mr/hr).

e. High Drywell Radiation (100 r/hr).

f. Reactor Building Ventilation Radiation Monitor Downscale (Both Channel A and B).

g. Refuel Floor Radiation Monitor Downscale (Both Channels A and B).

h. Failure of the Primary Train to Start (25-Second Time Delay), or Subsequent Loss of the Primary Train.

2. Upon receipt of an automatic start signal, the following actions will occur:

a. Primary train fan will start.

b. MO-7503, Reactor Bldg. Suction Damper, opens on the unit that generated the initiation signal.

c. MO-7503, Reactor Bldg. Suction Damper, not associated with the unit that generated the initiation signal, closes.

d. MO-7505, Primary Train Inlet Damper, opens.

e. MO-7504, Primary Train Turb Bldg Clg Air Damper, closes.

f. MO-7507, Primary Train Fan Discharge Damper, opens.

g. Primary train electric heater starts when flow exceeds 2600 scfm.

h. AO-7510, Primary Train Flow Control Damper, throttles closed to maintain the preset flow setpoint of 4000 scfm.

****SR-7500-K08**
****L-7500-K08**

**DVR 04-01-91-025**

**While swapping the Unit 1 RPS B power supply from dirty to clean power, the 1/2 B SBGTS train auto started even though it was selected to standby and RPS was de-energized for approximately 1 second. The auto start was not expected to occur since RPS had been de-energized for less than 25 seconds.**

**On 1-31-91, Unit 1 was in the Shutdown mode per scheduled refueling outage Q1R11. Operating personnel performed procedure QOP 7000-1. An EO switched the power supply for RPS Channel B from reserve to normal. This power supply switching caused an expected momentary loss of power to RPS B which de-energized the PCI Group II B relay logic. This logic de-energization, in turn, started the 1/2B SBGTS 25 sec. auto-start timer. At the same time, the U-1 NSO received a 1/2 scram and PCI Group II and III isolation signal as expected. The NSO restored normal power to RPS B and reset the 1/2 scram.**

100 ID: N-4600-K34 Points: 1.00

Unit 1 has just achieved criticality.
Unit 2 is at full power.

A small fire in the carpeting has prompted the Shift Manager to direct that the following control room personnel utilize air supply masks supplied from the control room breathing air headers. All other personnel are directed to leave the control room.

| Unit 1 | Unit 2 |
|---|---|
| Unit Supervisor | Unit Supervisor |
| UNSO | UNSO |
| ANSO | ~~ANSO~~ |
| Reactivity SRO | |

With this control room manning, the control room breathing air will deplete ____(1)____ designed ~~but will~~ and can be extended ____(2)____

A. (1) at the same rate as
(2) by MANUALLY replacing the bottles. location of bottles?

B. (1) faster than
(2) by MANUALLY replacing the bottles.

C. (1) at the same rate as
(2) AUTOMATICALLY by Service Air when header pressure decreases by 500 psig.

D. (1) faster than
(2) AUTOMATICALLY by Service Air when header pressure decreases by 500 psig.

Answer: B

need additional ref?

new

# EXAMINATION ANSWER KEY

2002 Quad Cities NRC Exam

## Question 100 Details

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #100 (RO) |
| System ID: | 9800 |
| User ID: | N-4600-K34 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 2.50 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | UFSAR pg. 6.4-9 |
| User Text: | 600000AA2 ~~10~~ 2.17 |
| User Number 1: | 2.90 |
| User Number 2: | 3.10 |
| Comment: | New question. Higher Answer is correct due to UFSAR design basis is to supply five people for six hours. With seven people it would need to be changed out sooner. There is no automatic or manual interconnection with service air system. Bottles do not automatically align, they must be manually changed out. |

Ref for new question

F.2. (cont'd)

## NOTE

**IF** the 1/2 Service Water Pump is energized from Bus 14(24), **THEN** a fire in the Unit 2(1) Turbine Building could result in a station black-out on Unit 1(2). The 1/2 Service Water Pump should be the last available Service Water Pump placed into operation.

d. **Prepare** Service Water Pump for startup:

(1) **Verify** proper upper and lower motor bearing oil levels. ____

(2) **IF** starting Service Water Pump 1(2)A, **THEN throttle** 1(2)-3999-3, SERV WTR PMP 1(2)A DISCH VLV approximately 90% closed. ____

(3) **IF** starting Service Water Pump 1(2)B, **THEN throttle** 1(2)-3999-1, SERV WTR PMP 1(2)B DISCH VLV approximately 90% closed. ____

(4) **IF** starting Service Water Pump 1/2, **THEN throttle** 1/2-3999-5, SERV WTR PMP 1/2 DISCH VLV approximately 90% closed. ____

e. At 912-1 panel **start** the Service Water Pump and **monitor** for proper starting current. ____

f. At the Service Water Pump:

(1) **Slowly open** the SERV WTR PMP DISCH VLV. ____

(2) **Check** pump for any indications of abnormal operation. ____

(3) **Monitor** local Pump Discharge Pressure indicator 1(2)-3941-8A/8B or 1/2-3941-8C for normal discharge pressure of 90 to 110 psig. ____

g. **WHEN** the Service Water Strainer venting is complete, **THEN close** the Strainer Vent. ____

#100



6.4-24 To comply with Regulatory Guide 1.120, "Fire Protection Guidelines for Nuclear Power Plants", which covers control room breathing air capabilities, the station established an emergency breathing apparatus system, utilizing a bottle reservoir located outside the control room. The system is designed to provide a crew of five men with six hours of air apiece.

6.4-25 This equipment consists of self-contained breathing apparatus which has an independent supply of fresh air, and allows operators to remain at their positions until the fumes are evacuated. The system also has twelve 300-ft$^3$ bottles located outside the control room and distributed through three manifolds to pressure-demand full face masks.

#### 6.4.4.4 Hydrogen Storage Facility

75 ft3 in addition to the bottles already aligned to the system - to meet the six hour estimate? OR is this in addition to...

6.4-26 As part of the HWC system, liquid hydrogen and liquid oxygen storage facilities are installed at the site. These facilities are described in Section 2.2.3.2 and are located 1500 feet south of the control room. The postulated hazards are failure at the gaseous or liquid storage vessels, which could result in an explosion and/or fireball, and a break in the gaseous or liquid pipeline, which could result in an atmospheric hydrogen concentration which exceeds the lower flammability limit of 4%. The location of these facilities is sufficiently far away from the control room so that these accidents will not affect habitability.

### 6.4.5 Testing and Inspection

Requirements for testing of instrumentation which isolates the control room HVAC system are given in Technical Specifications. Periodic inspection and testing of the AFU is performed as explained in Section 6.5.1. The balance of the system is used continuously during normal plant operations, therefore no additional testing is required.

### 6.4.6 Instrumentation Requirement

6.4-27 The isolation mode of the control room HVAC system is initiated automatically by signals received from the reactor pressure vessel (RPV) water level sensors, main steam line flow sensors, drywell pressure sensors, reactor building (including drywell and fuel pool) HVAC system radiation monitors, toxic gas analyzer, and smoke detectors. Reactor building HVAC system instrumentation is addressed in Section 9.4. Toxic gas monitoring instrumentation and smoke detectors were previously addressed in Section 6.4.4. Information about the RPV level sensors, main steam line flow sensors, and drywell pressure sensors is contained in Section 7.3.



NRC COPY #1

#100



6. IF deemed necessary, THEN initiate an orderly shutdown on both units. This decision should be based on the probability of continued occupation of the control room and intensity of the toxic fumes or smoke.

7. Notify Shift Manager to classify the event as a possible E-Plan condition and initiate E-Plan as necessary.

8. IF the TOXIC GAS CONCENTRATION HIGH alarm annunciates after the TOXIC GAS SAMPLE POINT SELECTOR SWITCH has been placed in the OPEN C position (control room return air duct), AND occupation of the control room is NO longer possible, THEN:

   a. Remove breathing air mask.

   b. Isolate the air line manifold supply valve.

   c. Don an air pack.

   d. Follow QOA 0010-05, Plant Operation With the Control Room Inaccessible.

CAUTION

Before removing air masks or packs, have Radiation Protection survey Control Room atmosphere to verify that cause is corrected and air is breathable.

9. Return the Control Room ventilation system to normal as soon as the cause is corrected by resetting the system isolation at Panel 901(2)-5 and local control panel 1/2-9400-105 in the "B" HVAC equipment room.

10. After the HVAC system is reset, place the TOXIC GAS SAMPLE POINT SELECTOR switch on local control panel 1/2-9400-105 to the appropriate position per QCOP 5750-09.

E. DISCUSSION

1. The breathing air bottles which supply the control room air manifolds are located on the East wall of the Unit 1 Turbine Building trackway. Bottle pressure indication is available in the control room.

2. The toxic gas analyzer system is equipped to detect Ammonia ($NH_3$); alarms at 20 ppm concentration.

# EXAMINATION ANSWER KEY

2002 Quad Cities NRC Exam

**101** **ID: SR-0704-K26** **Points: 1.00**

While performing a manual TIP trace on Unit 2, an area radiation alarm of 500 mr/hr on ARM # 8 first floor reactor building and a report of steam near the TIP room are received.

The operator performing the TIP trace states that the detector was placed into REVERSE. After some time, the operator reports the IN-SHIELD light EXTINGUISHED, the VALVE OPEN light LIT, and the detector position as shown on the display window NOT changing.

The Unit Supervisor shall: ~~Operator to take the~~

add specific procedure the SRO would use!

A. direct the TIP console Ball Valve Control Switch ~~taken~~ the to the CLOSED position.

B. directs the TIP console Speed Control Switch taken to the FAST position.

C. issue the Shear Valve key and direct the ~~Operator to fire the~~ shear valve fired.

D. ~~direct the TIP console Mode Switch taken to the AUTO position.~~ Issue the Ball Valve Override Key and direct ~~the Operator to close~~ the ~~one~~ Ball valve closed

Answer: C

**Question 101 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #76 (SRO) |
| System ID: | 9721 |
| User ID: | SR-0704-K26 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 3.00 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | QGA 300 step 27 |
| User Text: | 215001 2.4.6 |
| User Number 1: | 3.10 |
| User Number 2: | 4.00 |
| Comment: | New question. Higher. TIP is not fully retracted, so closing Ball valve is not possible. The TIP is already in reverse and no motion detected. QGA 300 step 27 directs isolation. |

This condition should be recognized by both RO & SROs & what action to take. This is more a abnormal mitigating strategy. But, why SRO only?

per 10CFR 55.43 (b)(5) assessment & selection of appropriate procedure

What does the ref have to do with quest.?

#101



D.3. **WHEN** running a TIP detector, the ball valve OPEN light **must** be indicating OPEN on the drive control unit ~~**AND** the valve control monitor.~~

D.4. **IF** the TIP ball valve fails to close automatically, **THEN**:

SRO ACTION

a. The Shift Manager shall **immediately issue** the shear valve control key to the Operator.

b. The problem shall then be investigated and repairs initiated.

c. **Refer** to QCOS 1600-08.

## E. LIMITATIONS AND ACTIONS

E.1. Access to the TIP room is permitted under Radiation Protection supervision provided dose rates at the TIPs have been obtained and all applicable exposure limits are adhered to.

E.2. Access to the Drywell is permitted under Radiation Protection supervision provided a survey of the TIP tubing for radiation exposure rates has been performed and all applicable exposure controls are adhered to.

E.3. **IF** a TIP is **NOT** in motion, **THEN** the Manual Direction Control switch should be left in the OFF position. This avoids reliance on proximity switches to prevent uncontrolled TIP movement.

E.4. **WHEN** the TIP is withdrawn to its shield, **THEN** the Manual Direction Control switch must **NOT** be left in the "REV" (reverse) position, or failure of the in-shield proximity switch can cause uncontrolled withdrawal to an unshielded position.

E.5. The computer and X-Y recorder can process only one TIP trace at a time. Although TIP machines may be operated simultaneously, ensure that only one TIP detector at a time is travelling within the core top and bottom limits.

E.6. **Before** a TIP is hand-cranked in past the ball valve with the unit in Mode 1, 2, or 3, **THEN** reference Technical Specification 3.6.1.3 for an inoperable primary containment isolation path.

#101

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Isolate all discharges into affected areas <u>except</u> systems needed for:

- Fire fighting
- Other QGA actions

?



NRC COPY #1

**102** **ID: SR-1601-K29** **Points: 1.00**

Which of the following conditions MEETS the requirement for primary containment integrity IAW Tech Specs and what is the bases?

(Assume the plant is operating at 50% power)

A. Drywell average air temperature is 148 degrees F.
This prevents exceeding heat capacity limit during an accident.

B. Drywell pressure is 1.53 psig.
This prevents exceeding Drywell design pressure during an accident.

C. Drywell pressure is 1.53 psig.
This prevents exceeding pressure suppression pressure during an accident.

D. Drywell average air temperature is 148 degrees F.
This prevents exceeding Drywell design temperature during an accident.

Answer: D

**Question 102 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #77 (SRO) |
| System ID: | 9722 |
| User ID: | SR-1601-K29 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 4.00 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | Tech Spec 3.6.1.5 |
| User Text: | 223001 2.1.33 |
| User Number 1: | 3.40 |
| User Number 2: | 4.00 |
| Comment: | New question. Memory. Entry condition for drywell temp is 150 degrees F, for drywell pressure is 1.5 psig and containment must be intact. |

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.5 Drywell Air Temperature

BASES

| | |
|---|---|
| BACKGROUND | The drywell contains the reactor vessel and piping, which add heat to the airspace. Drywell coolers remove heat and maintain a suitable environment. The average airspace temperature affects the calculated response to postulated Design Basis Accidents (DBAs). The limitation on the drywell average air temperature was developed as reasonable, based on operating experience. The limitation on drywell air temperature is used in the Reference 1 safety analyses. |
| APPLICABLE SAFETY ANALYSES | Primary containment performance is evaluated for a spectrum of break sizes for postulated loss of coolant accidents (LOCAs) (Ref. 1). Among the inputs to the design basis analysis is the initial drywell average air temperature (Ref. 1). Analyses assume an initial average drywell air temperature of 150°F. This limitation ensures that the safety analysis remains valid by maintaining the expected initial conditions and ensures that the peak LOCA drywell temperature does not exceed the maximum allowable temperature of 281°F (Ref. 2). Exceeding this design temperature may result in the degradation of the primary containment structure under accident loads. Equipment inside primary containment required to mitigate the effects of a DBA is designed to operate and be capable of operating under environmental conditions expected for the accident.<br><br>Drywell air temperature satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). |
| LCO | In the event of a DBA, with an initial drywell average air temperature less than or equal to the LCO temperature limit, the resultant peak accident temperature is maintained below the drywell design temperature. As a result, the ability of primary containment to perform its design function is ensured. |

(continued)

**103** **ID: SR-1603-K32** **Points: 1.00**

Unit 2 is operating at 100% power.
Instrument Maintenance reports one of the pressure switches for the MSIV low presssure isolation setpoint has drifted to 828 psig.

What is the required action and why?

to prevent offsite doses from exceeding 10 CFR 100 limits.

A. Restore to within tolerance within 24 hours or place the affected channel in trip to prevent ~~inadvertent injection with low pressure ECCS systems~~.

availability?

B. Restore to within tolerance within 12 hours or place the affected channel in trip to prevent exceeding the fuel cladding integrity safety limit.

C. Restore to within tolerance within 12 hours or place the affected channel in trip to prevent ~~inadvertent injection with low pressure ECCS systems~~.

D. Restore to within tolerance within 24 hours or place the affected channel in trip to prevent exceeding the fuel cladding integrity safety limit.

Answer: D

**Question 103 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #78 (SRO) |
| System ID: | 9723 |
| User ID: | SR-1603-K32 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 3.00 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | T.S 3.3.6.1 & bases |
| User Text: | 223002G.2.2.22 |
| User Number 1: | 3.40 |
| User Number 2: | 4.10 |
| Comment: | New question. Application. Function 1b is a 24 hr completion time as opposed to 12. The concern is exceeding safety limit for fuel cladding integrity. |

need to note in the write exam question!

Note: That Ref. T/S is provided - But not the Bases ?!

NO bases

#103

## 3.3 INSTRUMENTATION

### 3.3.6.1 Primary Containment Isolation Instrumentation

LCO 3.3.6.1 The primary containment isolation instrumentation for each Function in Table 3.3.6.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.6.1-1.

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each channel.
---------------------------------------------------------------------------

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| A. One or more required channels inoperable. | A.1 | Place channel in trip. | 12 hours for Functions 1.a, 2.a, 2.b, 3.d, 5.b, and 6.b<br><br>AND<br><br>24 hours for Functions other than Functions 1.a, 2.a, 2.b, 3.d, 5.b, and 6.b |
| B. One or more automatic Functions with isolation capability not maintained. | B.1 | Restore isolation capability. | 1 hour |

(continued)



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ACTIONS

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| C. Required Action and associated Completion Time of Condition A or B not met. | C.1 | Enter the Condition referenced in Table 3.3.6.1-1 for the channel. | Immediately |
| D. As required by Required Action C.1 and referenced in Table 3.3.6.1-1. | D.1 | Isolate associated main steam line (MSL). | 12 hours |
| | <u>OR</u> | | |
| | D.2.1 | Be in MODE 3. | 12 hours |
| | <u>AND</u> | | |
| | D.2.2 | Be in MODE 4. | 36 hours |
| E. As required by Required Action C.1 and referenced in Table 3.3.6.1-1. | E.1 | Be in MODE 2. | 8 hours |
| F. As required by Required Action C.1 and referenced in Table 3.3.6.1-1. | F.1 | Isolate the affected penetration flow path(s). | 1 hour |

(continued)



NRC COPY #1

#123

ACTIONS

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| G. Required Action and associated Completion Time for Condition F not met.<br><br>OR<br><br>As required by Required Action C.1 and referenced in Table 3.3.6.1-1. | G.1 | Be in MODE 3. | 12 hours |
| | AND | | |
| | G.2 | Be in MODE 4. | 36 hours |
| H. As required by Required Action C.1 and referenced in Table 3.3.6.1-1. | H.1 | Declare associated standby liquid control subsystem (SLC) inoperable. | 1 hour |
| | OR | | |
| | H.2 | Isolate the Reactor Water Cleanup System. | 1 hour |
| I. As required by Required Action C.1 and referenced in Table 3.3.6.1-1. | I.1 | Initiate action to restore channel to OPERABLE status. | Immediately |
| | OR | | |
| | I.2 | Initiate action to isolate the Residual Heat Removal (RHR) Shutdown Cooling System. | Immediately |

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SURVEILLANCE REQUIREMENTS

-----NOTES-----

1. Refer to Table 3.3.6.1-1 to determine which SRs apply for each Primary Containment Isolation Function.

2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains isolation capability.

---

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.3.6.1.1 | Perform CHANNEL CHECK. | 12 hours |
| SR 3.3.6.1.2 | Perform CHANNEL FUNCTIONAL TEST. | 92 days |
| SR 3.3.6.1.3 | Calibrate the trip unit. | 92 days |
| SR 3.3.6.1.4 | Perform CHANNEL CALIBRATION. | 92 days |
| SR 3.3.6.1.5 | Perform CHANNEL FUNCTIONAL TEST. | 24 months |
| SR 3.3.6.1.6 | Perform CHANNEL CALIBRATION. | 24 months |
| SR 3.3.6.1.7 | Perform LOGIC SYSTEM FUNCTIONAL TEST. | 24 months |

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Table 3.3.6.1-1 (page 1 of 3)
Primary Containment Isolation Instrumentation

| FUNCTION | APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS | REQUIRED CHANNELS PER TRIP SYSTEM | CONDITIONS REFERENCED FROM REQUIRED ACTION C.1 | SURVEILLANCE REQUIREMENTS | ALLOWABLE VALUE |
|---|---|---|---|---|---|
| 1. Main Steam Line Isolation | | | | | |
| a. Reactor Vessel Water Level—Low Low | 1,2,3 | 2 | D | SR 3.3.6.1.1<br>SR 3.3.6.1.2<br>SR 3.3.6.1.3<br>SR 3.3.6.1.6<br>SR 3.3.6.1.7 | ≥ -55.2 inches |
| b. Main Steam Line Pressure—Low | 1 | 2 | E | SR 3.3.6.1.2<br>SR 3.3.6.1.4<br>SR 3.3.6.1.7 | ≥ 831 psig |
| c. Main Steam Line Pressure—Timer | 1 | 2 | E | SR 3.3.6.1.2<br>SR 3.3.6.1.6<br>SR 3.3.6.1.7 | ≤ 0.331 seconds |
| d. Main Steam Line Flow-High | 1,2,3 | 2 per MSL | D | SR 3.3.6.1.1<br>SR 3.3.6.1.2<br>SR 3.3.6.1.6<br>SR 3.3.6.1.7 | ≤ 138% rated steam flow (Unit 1)<br>≤ 254.3 psid (Unit 2) |
| e. Main Steam Line Tunnel Temperature—High | 1,2,3 | 2 per trip string | D | SR 3.3.6.1.5<br>SR 3.3.6.1.6<br>SR 3.3.6.1.7 | ≤ 198°F |
| 2. Primary Containment Isolation | | | | | |
| a. Reactor Vessel Water Level—Low | 1,2,3 | 2 | G | SR 3.3.6.1.1<br>SR 3.3.6.1.2<br>SR 3.3.6.1.3<br>SR 3.3.6.1.6<br>SR 3.3.6.1.7 | ≥ 3.8 inches |
| b. Drywell Pressure—High | 1,2,3 | 2 | G | SR 3.3.6.1.2<br>SR 3.3.6.1.4<br>SR 3.3.6.1.7 | ≤ 2.43 psig |
| c. Drywell Radiation—High | 1,2,3 | 1 | F | SR 3.3.6.1.1<br>SR 3.3.6.1.2<br>SR 3.3.6.1.6<br>SR 3.3.6.1.7 | ≤ 70 R/hr |

(continued)



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Table 3.3.6.1-1 (page 2 of 3)
Primary Containment Isolation Instrumentation

| FUNCTION | APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS | REQUIRED CHANNELS PER TRIP SYSTEM | CONDITIONS REFERENCED FROM REQUIRED ACTION C.1 | SURVEILLANCE REQUIREMENTS | ALLOWABLE VALUE |
|---|---|---|---|---|---|
| 3. High Pressure Coolant Injection (HPCI) System Isolation | | | | | |
| a. HPCI Steam Line Flow – High | 1,2,3 | 1 | F | SR 3.3.6.1.2<br>SR 3.3.6.1.3<br>SR 3.3.6.1.6<br>SR 3.3.6.1.7 | ≤ 286% rated steam flow (Unit 1)<br>≤ 151% rated steam flow (Unit 2) |
| b. HPCI Steam Line Flow – Timer | 1,2,3 | 1 | F | SR 3.3.6.1.2<br>SR 3.3.6.1.6<br>SR 3.3.6.1.7 | ≥ 3.2 seconds and ≤ 8.8 seconds |
| c. HPCI Steam Supply Line Pressure – Low | 1,2,3 | 2 | F | SR 3.3.6.1.2<br>SR 3.3.6.1.3<br>SR 3.3.6.1.6<br>SR 3.3.6.1.7 | ≥ 113.0 psig |
| d. Drywell Pressure – High | 1,2,3 | 2 | F | SR 3.3.6.1.2<br>SR 3.3.6.1.4<br>SR 3.3.6.1.7 | ≤ 2.43 psig |
| e. HPCI Turbine Area Temperature – High | 1,2,3 | 2 | F | SR 3.3.6.1.5<br>SR 3.3.6.1.6<br>SR 3.3.6.1.7 | ≤ 169°F |
| 4. Reactor Core Isolation Cooling (RCIC) System Isolation | | | | | |
| a. RCIC Steam Line Flow – High | 1,2,3 | 1 | F | SR 3.3.6.1.2<br>SR 3.3.6.1.4<br>SR 3.3.6.1.7 | ≤ 175% rated steam flow |
| b. RCIC Steam Line Flow – Timer | 1,2,3 | 1 | F | SR 3.3.6.1.2<br>SR 3.3.6.1.6<br>SR 3.3.6.1.7 | ≥ 3.2 seconds and ≤ 8.8 seconds |
| c. RCIC Steam Supply Line Pressure – Low | 1,2,3 | 4[a] | F | SR 3.3.6.1.2<br>SR 3.3.6.1.4<br>SR 3.3.6.1.7 | ≥ 54 psig |
| d. RCIC Turbine Area Temperature – High | 1,2,3 | 2 | F | SR 3.3.6.1.5<br>SR 3.3.6.1.6<br>SR 3.3.6.1.7 | ≤ 169°F |

(continued)

(a) Only inputs into one trip system.



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Table 3.3.6.1-1 (page 3 of 3)
Primary Containment Isolation Instrumentation

| FUNCTION | APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS | REQUIRED CHANNELS PER TRIP SYSTEM | CONDITIONS REFERENCED FROM REQUIRED ACTION C.1 | SURVEILLANCE REQUIREMENTS | ALLOWABLE VALUE |
|---|---|---|---|---|---|
| 5. Reactor Water Cleanup System Isolation | | | | | |
| a. SLC System Initiation | 1,2 | 1 | H | SR 3.3.6.1.7 | NA |
| b. Reactor Vessel Water Level - Low | 1,2,3 | 2 | F | SR 3.3.6.1.1<br>SR 3.3.6.1.2<br>SR 3.3.6.1.3<br>SR 3.3.6.1.6<br>SR 3.3.6.1.7 | ≥ 3.8 inches |
| 6. RHR Shutdown Cooling System Isolation | | | | | |
| a. Reactor Vessel Pressure - High | 1,2,3 | 2 | F | SR 3.3.6.1.2<br>SR 3.3.6.1.4<br>SR 3.3.6.1.7 | ≤ 130 psig |
| b. Reactor Vessel Water Level - Low | 3,4,5 | 2(b) | I | SR 3.3.6.1.1<br>SR 3.3.6.1.2<br>SR 3.3.6.1.3<br>SR 3.3.6.1.6<br>SR 3.3.6.1.7 | ≥ 3.8 inches |

(b) In MODES 4 and 5, provided RHR Shutdown Cooling System integrity is maintained, only one channel per trip system with an isolation signal available to one shutdown cooling pump suction isolation valve is required.



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#103



BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Main Steam Line Isolation

1.a. Reactor Vessel Water Level — Low Low

Low reactor pressure vessel (RPV) water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, isolation of the MSIVs and other interfaces with the reactor vessel occurs to prevent offsite dose limits from being exceeded. The Reactor Vessel Water Level — Low Low Function is one of the many Functions assumed to be OPERABLE and capable of providing isolation signals. The Reactor Vessel Water Level — Low Low Function associated with isolation is assumed in the analysis of the recirculation line break (Ref. 5). The isolation of the MSLs supports actions to ensure that offsite dose limits are not exceeded for a DBA.

Reactor vessel water level signals are initiated from four differential pressure transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level — Low Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level — Low Low Allowable Value is chosen to be the same as the ECCS Reactor Vessel Water Level — Low Low Allowable Value (LCO 3.3.5.1) to ensure that the MSLs isolate on a potential loss of coolant accident (LOCA) to prevent offsite doses from exceeding 10 CFR 100 limits.

This Function isolates the Group 1 valves.

1.b. Main Steam Line Pressure — Low

Low MSL pressure indicates that there may be a problem with the turbine pressure regulation, which could result in a low reactor vessel water level condition and the RPV cooling down more than 100°F/hr if the pressure loss is allowed to continue. The Main Steam Line Pressure — Low Function is directly assumed in the analysis of the pressure regulator

(continued)

#103



BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.b. Main Steam Line Pressure—Low (continued)

failure (Ref. 6). For this event, the closure of the MSIVs ensures that the RPV temperature change limit (100°F/hr) is not reached. In addition, this Function supports actions to ensure that Safety Limit 2.1.1.1 is not exceeded. (This Function closes the MSIVs prior to pressure decreasing below 785 psig, which results in a scram due to MSIV closure, thus reducing reactor power to < 25% RTP.)

The MSL low pressure signals are initiated from four pressure switches that are connected to the MSL header close to the turbine stop valves. The switches are arranged such that, even though physically separated from each other, each switch is able to detect low MSL pressure. Four channels of Main Steam Line Pressure—Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be high enough to prevent excessive RPV depressurization.

The Main Steam Line Pressure—Low Function is only required to be OPERABLE in MODE 1 since this is when the assumed transient can occur (Ref. 6).

This Function isolates the Group 1 valves.

1.c. Main Steam Line Pressure—Timer

The Main Steam Line Pressure—Timer is provided to prevent false isolations on low MSL pressure as a result of pressure transients, however, the timer must function in a limited time period to support the OPERABILITY of the Main Steam Line Pressure—Low Function by enabling the associated channels after a certain time delay. The Main Steam Line Pressure—Timer is directly assumed in the analysis of the pressure regulator failure (Ref. 6). For this event, the closure of the MSIVs ensures that the RPV temperature change limit (100°F/hr) is not reached. In addition, this Function supports actions to ensure that Safety Limit 2.1.1.1 is not exceeded.

The MSL low pressure timer signals are initiated when the associated MSL low pressure switch actuates. Four channels

(continued)

#103

BASES

APPLICABLE SAFETY ANALYSES (continued)

2.1.1.1 Fuel Cladding Integrity

The use of the Siemens Power Corporation correlation (ANFB) is valid for critical power calculations at pressures > 600 psia and bundle mass fluxes > $0.1 \times 10^6$ lb/hr-ft$^2$ (Refs. 2 and 3). The use of the General Electric (GE) Critical Power correlation (GEXL) is valid for critical power calculations at pressures > 785 psig and core flows > 10% (Ref. 4). For operation at low pressures or low flows, the fuel cladding integrity SL is established by a limiting condition on core THERMAL POWER, with the following basis:

> Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be > 4.5 psi. Analyses show that with a bundle flow of $28 \times 10^3$ lb/hr (approximately a mass velocity of $0.25 \times 10^6$ lb/hr-ft$^2$), bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be > $28 \times 10^3$ lb/hr. Full scale critical power test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER > 50 % RTP. Thus, a THERMAL POWER limit of 25% RTP for reactor pressure < 785 psig is conservative. Although the ANFB correlation is valid at reactor steam dome pressures > 600 psia, application of the fuel cladding integrity SL at reactor steam dome pressure < 785 psig is conservative.

2.1.1.2 MCPR

The MCPR SL ensures sufficient conservatism in the operating MCPR limit that, in the event of an AOO from the limiting condition of operation, at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition. The margin between calculated boiling transition (i.e., MCPR = 1.00) and the MCPR SL is based on a detailed statistical procedure that considers the uncertainties in monitoring the core operating state. One specific uncertainty included in the SL is the uncertainty inherent

(continued)

**104** **ID: SR-5651-K29** **Points: 1.00**

Unit 2 is operating at 30% power.

Which of the following conditions on Unit 2 require entry into the Tech Spec for RFP/Main Turbine high level trip and why is this required?

RFP/Main Turbine High level trip setpoint is:

A. 52 inches to prevent exceeding peak cladding temperature on a LOCA.

B. 47 inches to prevent exceeding 1% plastic strain on the cladding.

C. 52 inches to prevent exceeding 1% plastic strain on the cladding.

D. 47 inches to prevent exceeding peak cladding temperature on a LOCA.

Answer: C

**Question 104 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #79 (SRO) |
| System ID: | 9724 |
| User ID: | SR-5651-K29 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 4.00 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | T.S. 3.3.2.2 |
| User Text: | 245000 2.1.33 |
| User Number 1: | 3.40 |
| User Number 2: | 4.00 |
| Comment: | New question. Memory. TS is not applicable below 25% power. Setpoint value is < 50.34 inches. |

BASES

LCO (continued)

calibration tolerances, instrument drift, and severe environment errors (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for and appropriately applied for the instrumentation.

APPLICABILITY

The Feedwater System and Main Turbine High Water Level Trip Instrumentation is required to be OPERABLE at ≥ 25% RTP to ensure that the fuel cladding integrity Safety Limit and the cladding 1% plastic strain limit are not violated during the feedwater controller failure, maximum demand event. As discussed in the Bases for LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," and LCO 3.2.3, "LINEAR HEAT GENERATION RATE," sufficient margin to these limits exists below 25% RTP; therefore, these requirements are only necessary when operating at or above this power level.

ACTIONS

A.1

With one or more channels inoperable, the Feedwater System and Main Turbine High Water Level Trip Instrumentation cannot perform its design function (Feedwater System and main turbine high water level trip capability is not maintained). Therefore, continued operation is only permitted for a 2 hour period, during which Feedwater System and main turbine high water level trip capability must be restored. The trip capability is considered maintained when sufficient channels are OPERABLE or in trip such that the Feedwater System and main turbine high water level trip logic will generate a trip signal on a valid signal. This requires two channels to each be OPERABLE or in trip. If the required channels cannot be restored to OPERABLE status or placed in trip, Condition B must be entered and its Required Action taken.

The 2 hour Completion Time is sufficient for the operator to take corrective action, and takes into account the likelihood of an event requiring actuation of Feedwater System and Main Turbine High Water Level Trip Instrumentation occurring during this period. It is also consistent with the 2 hour Completion Time provided in LCO 3.2.2 for Required Action A.1, since this instrumentation's purpose is to preclude a MCPR violation.

(continued)

Given Conditions?

**105** **ID: SR-6500-K29** **Points: 1.00**

With why is

For Unit 1, ~~which of the following plant conditions~~ MEETS Tech Specs requirements and why?

A. The Bus 13-1 to 23-1 cross-tie is OOS, T-22 can ~~still~~ supply Bus 14-1.

B. ~~The Bus 13-1 to 23-1 cross-tie is OOS~~, T-~~12~~ 22 can ~~still~~ supply Bus 13-1.

C. ~~Bus 14 trips~~, T-~~12~~ can ~~still~~ supply Bus ~~14~~ 13-1.

D. ~~Bus 14 trips~~, T-~~22~~ 21 12 can ~~still~~ supply Bus 14-1.

Answer: A

need more in stem - to focus on the aspect of T/S bases of condition -

## Question 105 Details

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #80 (SRO) |
| System ID: | 9725 |
| User ID: | SR-6500-K29 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 4.00 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | TS 3.8.1 |
| User Text: | 262001 2.1.33 |
| User Number 1: | 3.40 |
| User Number 2: | 4.00 |
| Comment: | New question. Higher. Bus 13-1/23-1 xtie inop is okay because TS requires only one cross tie operable. TS requires that T-12 be able to supply BOTH 13-1 AND 14-1. |

This is a T/F question - Each dist is a T/F statement that has nothing to do with any condition in the stem. In fact, there are no conditions in the stem!

BASES

LCO (continued)

Each offsite circuit from the 345 kV switchyard must be capable of maintaining rated frequency and voltage, and accepting required loads during an accident, while connected to the 4160 V ESS buses. An offsite circuit to each unit consists of the incoming breaker and disconnect to the respective 12 and 22 RATs, RATs 12 and 22, and the respective circuit path including feeder breakers to 4160 V ESS buses. A qualified circuit does not have to be connected to the ESS bus (i.e., the main generator can be connected to the ESS bus) as long as the capability to fast transfer to the qualified circuit exists. The other qualified offsite circuit for each unit is provided by a bus tie between the corresponding ESS buses of the two units. The breakers ~~connecting the buses must be capable of closure.~~ For Unit 1, LCO 3.8.1.a is met if RAT 12 is capable of supplying ESS buses 13-1 and 14-1 and if RAT 22 ~~(or UAT 21 on backfeed) can supply ESS bus 13~~-1 via ESS bus 23 and 23-1 and the associated bus tie or ESS bus 14-1 via ESS bus 24 and 24-1 and the associated bus tie. For Unit 2, LCO 3.8.1.a is met if RAT 22 can supply ESS buses 23-1 and 24-1 and if RAT 12 (or UAT 11 on backfeed) can supply ESS bus 23-1 via ESS bus 13 and 13-1 and the associated bus tie or ESS bus 24-1 via ESS bus 14 and 14-1 and the associated bus tie. For Unit 1, LCO 3.8.1.c is met if RAT 22 (or UAT 21 on backfeed) is capable of supplying ESS bus 29 to support equipment required by LCO 3.6.4.3. For Unit 2, LCO 3.8.1.c is met if RAT 12 (or UAT 11 on backfeed) is capable of supplying ESS bus 19, to support equipment required by LCO 3.6.4.3, and supplying ESS bus 18, to support equipment required by LCO 3.7.4 and LCO 3.7.5.

The respective unit DG and common DG must be capable of starting, accelerating to rated speed and voltage, and connecting to its respective 4160 V ESS bus on detection of bus undervoltage. This sequence must be accomplished within 10 seconds. Each respective unit DG and common DG must also be capable of accepting required loads within the assumed loading sequence intervals, and must continue to operate until offsite power can be restored to the 4160 V ESS buses. These capabilities are required to be met from a variety of initial conditions, such as DG in standby with the engine hot and DG in standby with the engine at ambient condition. Additional DG capabilities must be demonstrated to meet required Surveillances. Proper sequencing of loads, including tripping of nonessential loads, is a required function for DG OPERABILITY.

(continued)

**106** **ID: SL-REPT-K03** Also provide 1.8 **Points: 1.00**

1.6 & 1.7 provided

Both units are refueling with No testing in progress.
The Rx Building Vents and Control Room HVAC systems isolate due to an actual signal.

This event:

[illegible] the computer readings

A. must be reported immediately ~~is NOT reportable because all isolations are completed.~~

B. MUST be reported within 4 hours.

C. MUST be reported within 8 hours.

D. is NOT reportable because isolations ~~are NOT required to be operable~~. were due to an actual signal

Answer: C

**Question 106 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #81 (SRO) |
| System ID: | 9726 |
| User ID: | SL-REPT-K03 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 4.00 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | Rep. Man. Page 27 |
| User Text: | 290003 2.4.30 |
| User Number 1: | 2.20 |
| User Number 2: | 3.60 |
| Comment: | New question. Application. Criteria 2 of 10 CFR 50.73(a)(2)(iv)(A) states containment isolation signals affecting containment isolation valves in more than one system. 4 hours is for RPS. |

Ref req'd - need to note on master copy

#106

# REPORTABLE EVENT SAF 1.6:

## RPS Actuation

***Requirement:*** 10 CFR 50.72(b)(2)(iv)(B)
10 CFR 50.72(b)(3)(iv)(A)
10 CFR 50.72(b)(3)(iv)(B)
10 CFR 50.73(a)(2)(iv)(A)
10 CFR 50.73(a)(2)(iv)(B)

**§ 50.72(b)(2)(iv)(B):** The licensee shall notify the NRC as soon as practical and in all cases, within four hours of the occurrence of ... any event or condition that results in actuation of the reactor protection system (RPS) when the reactor is critical except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation.

**§ 50.72(b)(3)(iv)(A):** The licensee shall notify the NRC as soon as practical and in all cases, within eight hours of the occurrence of ... any event or condition that results in valid actuation of any of the systems listed in ... [§ 50.72(b)(3)(iv)(B)] ... except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation.

**§ 50.72(b)(3)(iv)(B):** The systems to which the requirements of ... [§ 50.72(b)(3)(iv)(A)] ... apply are: ... RPS ... including: reactor scram and reactor trip......

**§ 50.73(a)(2)(iv)(A):** The licensee shall report ... any event or condition that resulted in manual or automatic actuation of any of the systems listed in ... [§ 50.73(a)(2)(iv)(B)] ... except when:

(1) The actuation resulted from and was part of a pre-planned sequence during testing or operation; or

(2) The actuation was invalid and; (i) Occurred while the system was properly removed from service; or (ii) Occurred after the safety function had already been completed.

**§ 50.73(a)(2)(iv)(B):** The systems to which the requirements of ... [§ 50.73(a)(2)(iv)(A)] ... apply are: ...

(1) ... RPS ... including: reactor scram or reactor trip......

#106

**_REPORTABLE EVENT SAF 1.6 (Cont'd)_**

## *Discussion:*

- NRC guidance on this Reportable Event is provided in NUREG 1022, Revision 2, Section 3.2.6. Reporting under § 50.72 and § 50.73 is only required if this Reportable Event occurred within three years of the date of discovery.

- **RPS Actuation Summary Table**

| | Valid RPS Actuation | Valid RPS Actuation | Invalid RPS Actuation | Invalid RPS Actuation |
|---|---|---|---|---|
| Critical | 4 hr. telephone report | 60 day LER report | 4 hr. telephone report | 60 day LER report |
| Critical (preplanned) | No report | No report | No report | No report |
| Not Critical | 8 hr. telephone report | 60 day LER report | No report | 60 day LER or telephone report |
| Not Critical (preplanned) | No report | No report | No report | No report |

## *Related Reportable Events:*

- SAF 1.1, Declaration of Emergency Class
- SAF 1.2, Plant Shutdown Required by Technical Specifications

## *References:*

- NUREG 1022, Revision 2
- 10 CFR 50.72
- 10 CFR 50.73

TR106

# *REPORTABLE EVENT SAF 1.7:*

## System Actuation Not Including RPS

***Requirement:*** 10 CFR 50.72(b)(3)(iv)(A)
10 CFR 50.72(b)(3)(iv)(B)
10 CFR 50.73(a)(2)(iv)(A)
10 CFR 50.73(a)(2)(iv)(B)

**§ 50.72(b)(3)(iv)(A):** The licensee shall notify the NRC as soon as practical and in all cases, within eight hours of the occurrence of ... any event or condition that results in valid actuation of any of the systems listed in ... [§ 50.72(b)(3)(iv)(B)] ... except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation.

**§ 50.72(b)(3)(iv)(B):** The systems to which the requirements of ... [§ 50.72(b)(3)(iv)(A)] ... apply are:

(1) ...

(2) General containment isolation signals affecting containment isolation valves in more than one system or multiple main steam isolation valves ...

(3) ... ECCS for ... PWRs ... including: high-head, intermediate-head, and low-head injection systems and the low pressure injection function of residual (decay) heat removal systems.

(4) ECCS for ... BWRs ... including: high-pressure and low-pressure core spray systems; high-pressure coolant injection system; low pressure injection function of the residual heat removal system.

(5) BWR reactor core isolation cooling system; isolation condenser system; and feedwater coolant injection system.

(6) PWR auxiliary or emergency feedwater system.

(7) Containment heat removal and depressurization systems, including containment spray and fan cooler systems.

(8) Emergency ac electrical power systems, including: EDGs; ... and BWR dedicated Division 3 EDGs.

**§ 50.73(a)(2)(iv)(A):** The licensee shall report ... any event or condition that resulted in manual or automatic actuation of any of the systems listed in ... [§ 50.73(a)(2)(iv)(B)] ... except when:

(1) The actuation resulted from and was part of a pre-planned sequence during testing or operation; or

(2) The actuation was invalid and; (i) Occurred while the system was properly removed from service; or (ii) Occurred after the safety function had already been completed.



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*Reportable Event: SAF 1.7*

of response or protective measures taken, and (iii) information related to plant behavior that is not understood. [10 CFR 50.72(c)(2)] [I-29]

| *Time Limit* | *Required Written Report(s):* |
|---|---|
| 60 DAYS | Submit a Licensee Event Report to the NRC within 60 days of discovery of the occurrence of any event or condition that resulted in manual or automatic actuation of any of the systems listed in ... [10 CFR 50.73(a)(2)(iv)(B)] ... subject to exceptions allowed. Alternatively, pursuant to § 50.73(a)(1), in the case of an invalid actuation, telephone notification to the NRC Operations Center may be provided within 60 days after discovery of the event instead of submitting a written Licensee Event Report. [10 CFR 50.73(a)(2)(iv)(A)] [T-07] |

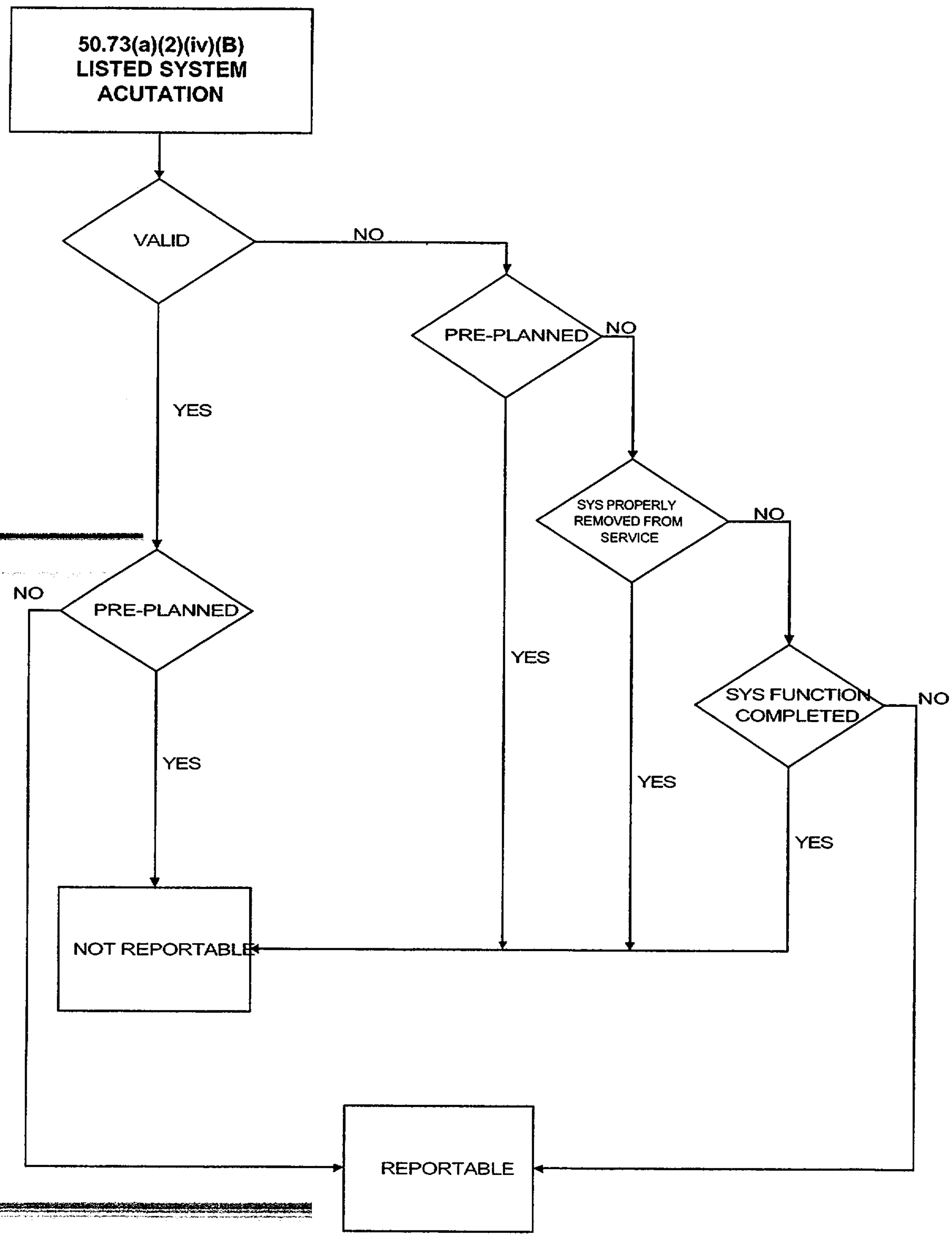
### *Discussion:*

- NRC guidance on this Reportable Event is provided in NUREG 1022, Revision 2, Section 3.2.6. Reporting under § 50.72 and § 50.73 is only required if this Reportable Event occurred within three years of the date of discovery.

### *Related Reportable Events:*

- SAF 1.1, Declaration of Emergency Class
- SAF 1.2, Plant Shutdown Required by Technical Specifications
- SAF 1.5, ECCS Injection/Actuation
- SAF 1.6, RPS Actuation

***Reportable Event: SAF 1.7***



# *REPORTABLE EVENT SAF 1.8:*

## Event or Condition That Could Have Prevented Fulfillment of a Safety Function

***Requirement:*** 10 CFR 50.72(b)(3)(v)
10 CFR 50.72(b)(3)(vi)
10 CFR 50.73(a)(2)(v)
10 CFR 50.73(a)(2)(vi)

**§ 50.72(b)(3)(v):** The licensee shall notify the NRC ... of ... any event or condition that at the time of discovery could have prevented the fulfillment of the safety function of structures or systems that are needed to:
(A) Shut down the reactor and maintain it in a safe shutdown condition;
(B) Remove residual heat;
(C) Control the release of radioactive material; or
(D) Mitigate the consequences of an accident.

**§ 50.72(b)(3)(vi):** Events covered in ... [10 CFR 50.72(b)(3)(v)] ... may include one or more procedural errors, equipment failures, and/or discovery of design, analysis, fabrication, construction, and/or procedural inadequacies. However, individual component failures need not be reported pursuant to ... [10 CFR 50.72(b)(3)(v)] ... if redundant equipment in the same system was operable and available to perform the required safety function.

**§ 50.73(a)(2)(v):** The licensee shall report ... any event or condition that could have prevented the fulfillment of the safety function of structures or systems that are needed to:
(A) Shut down the reactor and maintain it in a safe shutdown condition;
(B) Remove residual heat;
(C) Control the release of radioactive material; or
(D) Mitigate the consequences of an accident.

**§ 50.73(a)(2)(vi):** Events covered in ... [10 CFR 50.73(a)(2)(v)] ... may include one or more procedural errors, equipment failures, and/or discovery of design, analysis, fabrication, construction, and/or procedural inadequacies. However, individual component failures need not be reported pursuant to ... [10 CFR 50.73(a)(2)(v)] ... if redundant equipment in the same system was operable and available to perform the required safety function.

**_REPORTABLE EVENT SAF 1.8 (Cont'd)_**

***References:***

- NUREG 1022, Revision 2
- 10 CFR 50.72
- 10 CFR 50.73

**107** **ID: SR-1000-K26** **Points: 1.00**

Unit 2 has the following conditions:

- Rx Water temperature is 190 degrees F.
- RPV Water level is 90 inches.
- The Mode switch is in SHUTDOWN.

An inadvertent Group 2 isolation occurs that cannot be reset. Rx Water temperature rises to 220 degrees F before being turned.

How did plant conditions change and what procedure should be entered?

A. The plant went from Mode 3 to Mode 4;
QCOA 1000-02, LOSS OF SHUTDOWN COOLING should be entered.

B. The plant went from Mode 4 to Mode 3;
QCOA 1600-02, LOSS OF PRIMARY AND/OR SECONDARY CONTAINMENT should be entered.

C. The plant went from Mode 3 to Mode 4;
QCOA 1600-02, LOSS OF PRIMARY AND/OR SECONDARY CONTAINMENT should be entered.

D. The plant went from Mode 4 to Mode 3;
QCOA 1000-02, LOSS OF SHUTDOWN COOLING should be entered.

Answer: D

**Question 107 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #82 (SRO) |
| System ID: | 9731 |
| User ID: | SR-1000-K26 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 4.00 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | TS Table 1.1-1 |
| User Text: | Generic 2.1.22 |
| User Number 1: | 2.80 |
| User Number 2: | 3.30 |
| Comment: | New question. Higher. When temp exceeds 212 degrees F, the Rx will go from Mode 4 to Mode 3. Due to the Group 2 and temp rise, QCOA 1000-02, Loss of Shutdown Cooling must be entered. There was not a Loss of Containment, but an inadvertant isolation. |

#107



Table 1.1-1 (page 1 of 1)
MODES

| MODE | TITLE | REACTOR MODE SWITCH POSITION | AVERAGE REACTOR COOLANT TEMPERATURE (°F) |
|---|---|---|---|
| 1 | Power Operation | Run | NA |
| 2 | Startup | Refuel[(a)] or Startup/Hot Standby | NA |
| 3 | Hot Shutdown[(a)] | Shutdown | $> 212$ |
| 4 | Cold Shutdown[(a)] | Shutdown | $\leq 212$ |
| 5 | Refueling[(b)] | Shutdown or Refuel | NA |

(a) All reactor vessel head closure bolts fully tensioned.

(b) One or more reactor vessel head closure bolts less than fully tensioned.



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**108** | **ID: S-0001-K12** | **Points: 1.00**

During post LOCA conditions, with drywell temperature at 260°F, and reactor building temperature at 198°F, the following reactor water levels are noted at the same time: 100 psig RPV press

| | |
|---|---|
| Lower Wide Range | -60 inches |
| Fuel Zone | -70 inches |
| Medium Range | -50 inches |
| Upper Wide Range | ~~-60~~ -30 inches |

Which of the above level indicators CANNOT be used in these plant conditions?

A. Lower Wide Range ~~and Fuel Zone~~ ok

B. Medium Range and Upper Wide Range

C. Medium Range ONLY

D. ~~Lower~~ Upper Wide Range ONLY

Given the Table this a direct look up?

Answer: B

why is this SRO only?

**Question 108 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #83 (SRO) |
| System ID: | 106 |
| User ID: | S-0001-K12 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 4.00 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | QGA Detail A |
| User Text: | G.2.1.25 |
| User Number 1: | 2.80 |
| User Number 2: | 3.10 |
| Comment: | STA (75178) Bank question. Higher. Medium range cannot be used because level is - 53 and temp > 195. Upper wide range cannot be used because level is < 73 inches. Lower wide range can be used because level is > -301 inches. Fuel Zone can be used because level is > -303 inches. |

Ref provided in Exam.

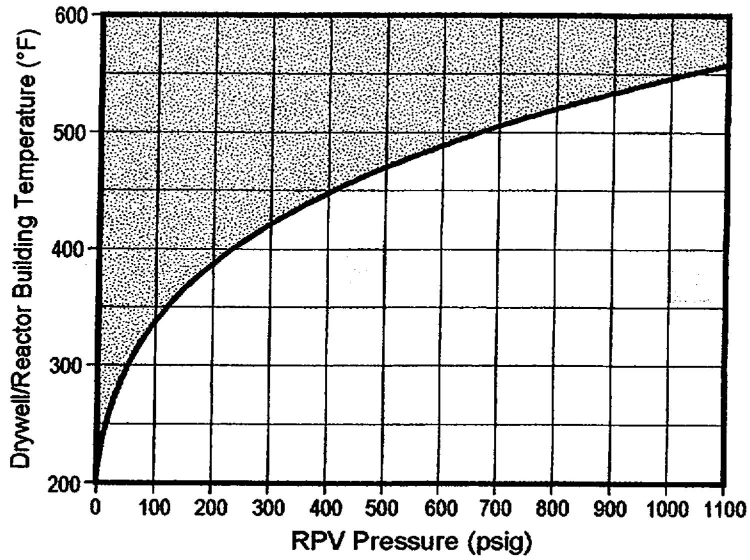
#10?

# A RPV Water Level Instruments

**CAUTION: RPV water level instruments may be unreliable due to boiling in the instrument runs when drywell or reactor building temperature is above Fig B, RPV Saturation Temperature.**

An RPV water level instrument may be used <u>only if</u> the criteria in Table C are satisfied.

## B RPV Saturation Temperature




## C RPV Level Instrument Criteria

| Instrument | Range (in.) | Use only if... |
|---|---|---|
| Fuel Zone | -340 to +60 | Indicated level above -303 in. |
| Lower Wide Range | -344 to +66 | Indicated level above -301 in. |
| Medium Range | -60 to +60 | Indicated level above -43 in. **OR** Reactor building temperature below 195°F |
| Upper Wide Range | -42 to +358 | Indicated level above 73 in. |
| Narrow Range | 0 to +60 | Indicated level on-scale |

NRC COPY #1

**109** **ID: SR-PGTM-K3** **Points: 1.00**

Which one of the following would qualify as a "Temporary Configuration Change" as defined in CC-AA-112, "Temporary Configuration Changes"?

A. An electrical lead is lifted in accordance with a surveillance procedure.

B. A circuit card is pulled to disable an annunciator.

C. A Service Air hose drop is being used for mantenance on a RFP.

D. A hose is installed to drain a heat exchanger under a clearance order.

Answer: B

**Question 109 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #84 (SRO) |
| System ID: | 7487 |
| User ID: | SR-PGTM-K3 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 4.00 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | CC-AA-112 R. 5, pg 23-25 |
| User Text: | G.2.2.11 |
| User Number 1: | 2.50 |
| User Number 2: | 3.40 |
| Comment: | 82605 Bank question. Higher. Disabled annunciators are required to be controlled as a temp change per the table on pg. 23, the others are not. |

**ATTACHMENT 2**
**TCCPs, Exclusions and Associated Administrative Controls (CM 6.1.2.1)**
**Page 2 of 3**

| Controlled and Issued as TCCPs | Pre-Engineered Activities (See Note 1) |
|---|---|
| Temp Heat/Cooling for supplementing equipment heating or cooling requirements | |
| Scaffolding attached to plant system components or appurtenances | |
| Line Stops | |

Note 1: The temporary changes identified in the "Controlled as Procedural Temporary Changes" may not apply to both Regions (Mid-West, and Mid Atlantic). To confirm the availability of procedures that address these topics, refer to the ROG specific Training and Reference Material document for Temporary Configuration Changes.

## Exclusions and associated Administrative Control Requirements

1. **Surveillance and Inservice tests** are repetitive in nature and typically controlled through specific station procedures which call for temporary configuration change (i.e., installation of a jumper to conduct a trip and cal test, would not fall under this procedure).
2. If evolution of a permanent modification includes **temporary changes required to support the implementation of the permanent modification**, and has been evaluated as part of permanent modification process, then temporary changes are exempted.
3. **Maintenance activities, replacements, troubleshooting and surveillance functions** that are conducted in accordance with an approved procedure, or Work Orders developed from the requirements of task specific station approved procedures.
4. **SSCs included <u>within</u> an Operations Clearance**.
5. **M&TE equipment** discussed in 5.a and 5.b, below, shall be tagged per station procedures for implementing the change. The Work Order number used for installing the M&TE shall be entered into the TCCP Tracking Log for Operator awareness. Additionally, the M&TE items shall be tracked in the TCCP monthly report for use in periodically review by the SM. **(CM-6.1.5.11)**

   a. Measurement and Test Equipment (M&TE) installed on equipment with engineered test points do not require a TCCP.

   b. A TCCP is not required for M&TE installed for troubleshooting efforts on equipment without engineered test points that meet the following requirements: **(CM-6.1.2.7)**
   - M&TE does not change the system's design function
   - M&TE are installed and controlled in accordance with an approved procedure or work package instructions provided that the temporary change of the equipment is clearly documented.
   - The system is returned to normal configuration 90 days after installation. (based on Reference 6.5 and Reference 6.12)

XX\09

## ATTACHMENT 2
## TCCPs, Exclusions and Associated Administrative Controls
## Page 3 of 3

- Risk significance has been assessed in accordance with Reference 6.9.

c. For M&TE installed on equipment without engineered test points that do not meet the requirements of 5.b, above, the installation is to be done as a TCCP. **(CM-6.1.2.7)**

6. **120/480 Volt outlets.** Connection of portable equipment to permanently installed plant power feeds (i.e., electrical receptacles, or welding outlets) is not considered a TCCP, provided that the load requirements of the portable equipment (especially in 480v outlets) do not exceed that load included in the Electrical Load Monitoring System for AC power or provided by the auxiliary power system. Otherwise, a TCCP shall be generated.

7. **Service Air hoses and water drops.** Provided that the cross contamination precaution is adhered to, there is no engineering concern for using any of the Service Air or water drops throughout the plant. Similar to power supplies, this does not comment on the devices being used, or general housekeeping, or potential for leakage.

8. **Hoses connected from system drains and vents to floor drains** as part of an approved procedure(s).

9. **Hoses/tubing, and their connecting fittings, connected from non-safety related sample points** for the purpose of obtaining chemistry samples and routed to drains, that do not affect equipment operation either upstream or downstream of the sample point are not considered TCCPs.

10. **Air Movers (fans and eductors).** These generally use Service Air or 480V power as addressed above. Uses may include local application for personnel or general area cooling while work is being done in the area. Consideration before installation of air movers should include possible "masking" of equipment degradation and the need to consider radiological and ventilation (boundary) concerns. **Air movers that are used to replace/augment a design function of permanent HVAC systems <u>require</u> TCCPs to assure complete evaluation of impact and safety significance of the configuration change.**

**110** **ID: SL-0805-K21** **Points: 1.00**

The Fuel Handler Grapple Operator is about to remove the last fuel bundle in a core quadrant. The NSO informs you that the count rate for the adjacent SRM is zero.

Which of the following statements is correct for this condition?

A. Proceed with removing the last bundle, it is too far from the SRM to be detected.

B. ~~The NSO~~ will ~~direct the~~ Instrument Maintenance department to troubleshoot the SRM in question.
Also Add

C. Fuel movements MUST cease until the SRM reads greater than 3 cpm.

D. Operability requirements for SRMs do NOT apply in this case.
Proceed in removing the last bundle

Answer: D

**Question 110 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #85 (SRO) |
| System ID: | 4394 |
| User ID: | SL-0805-K21 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 4.00 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | TS 3.3.1.2 & SR 3.3.1.2.4 |
| User Text: | G2.2.27 |
| User Number 1: | 2.60 |
| User Number 2: | 3.50 |
| Comment: | ILT.08480 (79483) Bank question. Lower. Per TS SR 3.3.1.2.4, 3 cps are not required if there are ≤ 4 fuel assemblies adjacent to the SRM and no fuel assemblies are in the quadrant. This was the last fuel assembly in this quadrant. |

Ref provided in Exam.

why do they need the T/S

If use T/S it is a direct look up

Isn't this a system T/S based question?

#110

## 3.3 INSTRUMENTATION

### 3.3.1.2 Source Range Monitor (SRM) Instrumentation

LCO 3.3.1.2 The SRM instrumentation in Table 3.3.1.2-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1.2-1.

ACTIONS

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| A. One or more required SRMs inoperable in MODE 2 with intermediate range monitors (IRMs) on Range 2 or below. | A.1 | Restore required SRMs to OPERABLE status. | 4 hours |
| B. Three required SRMs inoperable in MODE 2 with IRMs on Range 2 or below. | B.1 | Suspend control rod withdrawal. | Immediately |
| C. Required Action and associated Completion Time of Condition A or B not met. | C.1 | Be in MODE 3. | 12 hours |

(continued)



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#110



ACTIONS

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| D. One or more required SRMs inoperable in MODE 3 or 4. | D.1 | Fully insert all insertable control rods. | 1 hour |
| | AND | | |
| | D.2 | Place reactor mode switch in the shutdown position. | 1 hour |
| E. One or more required SRMs inoperable in MODE 5. | E.1 | Suspend CORE ALTERATIONS except for control rod insertion. | Immediately |
| | AND | | |
| | E.2 | Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies. | Immediately |

#110

SURVEILLANCE REQUIREMENTS

-----NOTE-----
Refer to Table 3.3.1.2-1 to determine which SRs apply for each applicable MODE or other specified condition.

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.3.1.2.1 | Perform CHANNEL CHECK. | 12 hours |
| SR 3.3.1.2.2 | -----NOTES-----<br>1. Only required to be met during CORE ALTERATIONS.<br>2. One SRM may be used to satisfy more than one of the following.<br>----------<br>Verify an OPERABLE SRM detector is located in:<br>a. The fueled region;<br>b. The core quadrant where CORE ALTERATIONS are being performed, when the associated SRM is included in the fueled region; and<br>c. A core quadrant adjacent to where CORE ALTERATIONS are being performed, when the associated SRM is included in the fueled region. | 12 hours |
| SR 3.3.1.2.3 | Perform CHANNEL CHECK. | 24 hours |

(continued)



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#110

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.3.1.2.4 | -----NOTE----- Not required to be met with less than or equal to four fuel assemblies adjacent to the SRM and no other fuel assemblies in the associated core quadrant. ----- <br><br> Verify count rate is: <br><br> a. $\geq$ 3.0 cps; or <br><br> b. $\geq$ 0.7 cps with a signal to noise ratio $\geq$ 20:1. | look up <br><br> 12 hours during CORE ALTERATIONS <br><br> AND <br><br> 24 hours |
| SR 3.3.1.2.5 | -----NOTE----- The determination of signal to noise ratio is not required to be met with less than or equal to four fuel assemblies adjacent to the SRM and no other fuel assemblies in the associated core quadrant. ----- <br><br> Perform CHANNEL FUNCTIONAL TEST and determination of signal to noise ratio. | 7 days |
| SR 3.3.1.2.6 | -----NOTE----- Not required to be performed until 12 hours after IRMs on Range 2 or below. ----- <br><br> Perform CHANNEL FUNCTIONAL TEST and determination of signal to noise ratio. | 31 days |

(continued)

#110



SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.3.1.2.7 | ------------------NOTES------------------<br>1. Neutron detectors are excluded.<br>2. Not required to be performed until 12 hours after IRMs on Range 2 or below.<br>---------------------------------------<br>Perform CHANNEL CALIBRATION. | 24 months |

#110



Table 3.3.1.2-1 (page 1 of 1)
Source Range Monitor Instrumentation

| FUNCTION | APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS | REQUIRED CHANNELS | SURVEILLANCE REQUIREMENTS |
|---|---|---|---|
| 1. Source Range Monitor | 2[a] | 3 | SR 3.3.1.2.1<br>SR 3.3.1.2.4<br>SR 3.3.1.2.6<br>SR 3.3.1.2.7 |
| | 3,4 | 2 | SR 3.3.1.2.3<br>SR 3.3.1.2.4<br>SR 3.3.1.2.6<br>SR 3.3.1.2.7 |
| | 5 | 2[b][c] | SR 3.3.1.2.1<br>SR 3.3.1.2.2<br>SR 3.3.1.2.4<br>SR 3.3.1.2.5<br>SR 3.3.1.2.7 |

(a) With IRMs on Range 2 or below.

(b) Only one SRM channel is required to be OPERABLE during spiral offload or reload when the fueled region includes only that SRM detector.

(c) Special movable detectors may be used in place of SRMs if connected to normal SRM circuits.

# EXAMINATION ANSWER KEY

2002 Quad Cities NRC Exam

**111** **ID: NGET9756** **Points: 1.00**

During a refueling outage, with LPRM detector replacement in progress; an LPRM detector is discovered in a trash barrel in the ~~Auxiliary~~ Reactor Building by a contractor.

RP determined that the contractor received:

- 4 Rem Whole Body
- 16 Rem to the eyes
- 25 Rem shallow dose to his right hand

What is (are) the required notification(s)?

1. A report specifying the exposure issued to the contractor.
2. Notify the NRC Operations Center via the ENS immediately, but no later than 1 hour.
3. Notify the NRC Operations Center within 24 hours.
4. Submit a written report to the NRC within 30 days.

A. 1 ONLY

B. 1 and ~~4~~ 3 ONLY

C. 1, 2 and 4 ONLY

D. 1, 3 and 4 ONLY

Answer: D

How does this got to do with rad control requirements - ie - release ... - limits ... etc ...

It appears on notification requirements - which we previously tested on question Q #106

Ref provided in Exam.

Given the ref - looks like a direct look up?

**Question 111 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #86 (SRO) |
| System ID: | 9756 |
| User ID: | NGET9756 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 4.00 |
| Time to Complete: | 0 |
| Point Value: | 1.00 1120 GMT 10/17/02 |
| Cross Reference: | LS-AA-~~1110~~, R. 0, pg. 17 |
| User Text: | G.2.3.1 |
| User Number 1: | 2.60 |
| User Number 2: | 3.00 |
| Comment: | Bank question. Higher. It is required to Notify the NRC within 24 hours, submit a written report to the contractor and the NRC within 30 dayA 1 hour notification is NOT required. |

RAD control requirements not an SRO only - used rad limits + notify

K/A G.2.4.30

#111



# *REPORTABLE EVENT RAD 1.5:*

Individual Exposure and Radiation Levels
(Unplanned or Planned Special Exposures)

***Requirement:*** 10 CFR 20.2202(a)
10 CFR 20.2202(b)
10 CFR 20.2203(a)
10 CFR 20.1206(g)
10 CFR 20.2204
10 CFR 20.2205
10 CFR 19.13(d)

**Unplanned Exposures**

**§ 20.2202(a):** Each licensee shall immediately report any event involving byproduct, source, or special nuclear material possessed by the licensee that may have caused or threatens to cause any of the following conditions:

(1) An individual to receive -
- (i) A total effective dose equivalent of 25 rems (0.25 Sv) or more; or
- (ii) A lens dose equivalent of 75 rems (0.75 Sv) or more; or
- (iii) A shallow-dose equivalent to the skin or extremities of 250 rads (2.5 Gy) or more; or

(2) The release of radioactive material, inside or outside of a restricted area, so that, had an individual been present for 24 hours, the individual could have received an intake five times the annual limit on intake (the provisions of this paragraph do not apply to locations where personnel are not normally stationed during routine operations, such as hot-cells or process enclosures).

**§ 20.2202(b):** Each licensee shall, within 24 hours of discovery of the event, report any event involving loss of control of licensed material possessed by the licensee that may have caused, or threatens to cause, any of the following conditions:

(1) An individual to receive, in a period of 24 hours -
- (i) A total effective dose equivalent exceeding 5 rems (0.05 Sv); or
- (ii) A lens dose equivalent exceeding 15 rems (0.15 Sv); or
- (iii) A shallow-dose equivalent to the skin or extremities exceeding 50 rems (0.5 Sv); or

(2) The release of radioactive material, inside or outside of a restricted area, so that had an individual been present for 24 hours, the individual could have received an intake in excess of one occupational annual limit on intake (the provisions of this paragraph do not apply to locations where personnel are not normally stationed during routine operations, such as hot-cells or process enclosures).

#111



***REPORTABLE EVENT RAD 1.5 (Cont'd)***

**Planned Special Exposures**

**20.1206(g):** The licensee records the best estimate of the dose resulting from the planned special exposure in the individual's record and informs the individual, in writing, of the dose within 30 days from the date of the planned special exposure.

**20.2204:** The licensee shall submit a written report to the ... NRC ... within 30 days following any planned special exposure conducted in accordance with 10 CFR 20.1206.

| ***Time Limit*** | ***Required Notification(s):*** |
|---|---|

**Unplanned Exposures**

| | |
|---|---|
| 1 HOUR | Notify the NRC Operations Center via the ENS immediately but not later than 1 hour after the occurrence of significant or threatened exposures or releases. [10 CFR 20.2202(a)] [I-10] |
| 24 HOURS | Notify the NRC Operations Center within 24 hours of discovery of any of the events listed above. [10 CFR 20.2202(b)] [P-01] |

**Planned Special Exposures**

| | |
|---|---|
| NONE | No immediate notification is required for Planned Special Exposures. |

| ***Time Limit*** | ***Required Written Report(s):*** |
|---|---|
| 30 DAYS | Submit a written report to the NRC within 30 days of occurrence of the event in accordance with § 20.2203(c). The content of the report is prescribed by § 20.2203(b). [10 CFR 20.2203(a)(1)] [T-14] |
| 30 DAYS | If the event involves exposure of an identified occupationally exposed individual or an identified member of the public, notify the individual involved in writing no later than the transmittal of the follow-up report to the NRC. The information to be provided to the individual is specified in § 19.13(a). [10 CFR 19.13(d) and 10 CFR 20.2205] [T-14] |

***REPORTABLE EVENT RAD 1.5 (Cont'd)***

- The dose from planned special exposures is not to be considered in controlling future occupational dose of the individual but is to be included in evaluations required for other planned special exposures.

## *Related Reportable Events:*

- RAD 1.1, Events Involving Byproduct, Source or Special Nuclear Material That Cause or Threaten to Cause Significant Exposure or Release
- RAD 1.2, Events Involving Loss of Control of Licensed Material That Cause or Threaten to Cause Exposure or Release
- RAD 3.1, Events Involving Byproduct, Source, or SNM Causing Significant Exposure or Release
- RAD 3.2, Events Involving Licensed Material Causing Exposure or Release
- RAD 3.3, Individual Exposure and Radiation Levels (Unplanned or Planned Special Exposures)

## *References:*

- 10 CFR 19.13
- 10 CFR 20

**112** **ID: SN-2002-K28** **Points: 1.00**

A transfer of Unit 2 clean-up phase seperator sludge to radwaste is scheduled for your shift

In order to guard against personnel exposure during the transfer:

A. access to Unit 2 reactor building second floor must be restricted.

B. Unit 2 RWCU system must be secured and isolated.

C. both Unit 2 RWCU pumps must be in operation with both filter demins at maximum flow.

D. access to Unit 2 reactor building first floor must be restricted.

Answer: A

**Question 112 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #87 (SRO) |
| System ID: | 9808 |
| User ID: | SN-2002-K28 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 3.00 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | QOP 2050-09, R. 13, pg 2 |
| User Text: | G.2.3.10 |
| User Number 1: | 2.90 |
| User Number 2: | 3.30 |
| Comment: | New question. Lower. Precaution in QOP 2050-09, D.2, restricts access to the Reactor building 2nd Floor during sludge transfer. RWCU is not mentioned. |

# 112
(SR087)



6. Field Supervisor or designee to be in attendance during Cleanup sludge transfer operations. ____

7. HLA prejob brief conducted with applicable Operations and Radiation Protection Personnel in attendance. ____

8. Remote camera(s) and monitor installed to view mixing pump oil level and seal in Cleanup Sludge Mixing Pump Room on unit from which sludge is to be transferred. ____

NOTE

The manual valve lineup does NOT need to be verified if previously completed and documented in the system lineup checklist AND the lineup has NOT been changed.

NOTE

The valves in the valve lineup are NOT required to be positioned in the order listed.

9. Manual valve lineup completed per Attachment A, Transfer of Cleanup Phase Separator Sludge to the Solidification System Mixing Tank. ____

10. Step off pad set up outside Unit 1(2) RWCU Phase Separator Pump Room. ____

11. Lighting verified to be adequate in Unit 1(2) RWCU Phase Separator Pump Room. ____

D. PRECAUTIONS

1. Transfer lines in Radwaste basement in the area of the cross-tie line between Cleanup Phase Separators and feed to A centrifuge must be monitored during transfers for leakage. IF leakage is noted, THEN transfer must be stopped and header isolated.

2. Access to Radwaste Building basement upper levels and Unit 1/Unit 2 Reactor Building second floor, as applicable, must be restricted during the sludge transfer operations.

3. WHEN transfer of sludge is completed, THEN Radiation Protection must survey normally accessible transfer piping and verify dose rates are acceptable for normal access.

**113** **ID: SR-0500-K26** **Points: 1.00**

You are the Unit 2 Supervisor during an ATWS.
ALL RPS scram signals have been bypassed by jumper installation, the scram reset, and the eight scram solenoid group indicating lights are lit.
You have indication on the full core display that the scram valves are still open.

In order to completely close the scram valves and drain the scram discharge volume, you would direct an NSO to: ?

Grammar? The sentence does not make sense.

A. ~~inside~~ the Control Room panels to de-energize ARI in accordance with QCOP 0300-28, ALTERNATE CONTROL ROD INSERTION.

B. the Auxiliary Electric Room to de-energize the scram solenoids in accordance with QCGP 2-3, REACTOR SCRAM.

C. inside the Control Room panels to de-energize the scram solenoids in accordance with QCGP 2-3, REACTOR SCRAM.

D. the Auxiliary Electric Room to de-energize ARI in accordance with QCOP 0300-28, ALTERNATE CONTROL ROD INSERTION.

Answer: D

**Question 113 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #88 (SRO) |
| System ID: | 9809 |
| User ID: | SR-0500-K26 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 4.00 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | QCOP 0300-28, R 19, pg. 9 |
| User Text: | G. 2.4.34 |
| User Number 1: | 3.80 |
| User Number 2: | 3.60 |
| Comment: | New question. Higher. ARI is deenergized in the aux electric room IAW QCOP 0300-28. |

#113
(See #88)



F.5.d. (cont'd)

## CAUTION

**Unit 1 Only:** **IF** the Feedwater Level Control System is in AUTO, **THEN** resetting the Reactor Scram will return Feedwater Level Control to its tape setpoint value, possibly resulting in a sudden reactivity addition, Turbine trip or Reactor Feed Pump trip due to the increase in Feedwater flow and RPV level.

**Unit 2 Only:** Reset of scram > 20 seconds after initiation will have **NO** effect on the FWLC System.

(3) **IF** a RPV level transient is anticipated, **THEN control** the RPV level transient by adjusting the Feedwater Level Control level setpoint **OR** taking manual control of the Feedwater Regulating Valves. ____

(4) **Reset** Reactor Scram. ____

## CAUTION

This next step prevents automatic operation of ARI valves by removing the ARI valve power supply.

e. **IF** scram valves are open, **THEN pull** the following fuses, located in Auxiliary Electrical Room, to allow automatic closure of ARI valves:

(1) At Panel 2201(2)-70A:

(a) Fuse F20A, second fuse from bottom of fuse block. ____

(b) Fuse F21A, bottom fuse of fuse block. ____

**114** **ID: SRNLF-00-K06** **Points: 1.00**

IAW OP-AA-106-101, SIGNIFICANT EVENT REPORTING, which of the following situations would require notification of the Work Week Manager and the Duty Engineering Manager?

A. S.R. 3.8.1.1 NOT completed within one hour.

B. T-12 trips with the Unit - 1 Emergency Diesel Generator OOS.

C. An inadvertant 1/2 scram is received.

D. Failure of SPDS.

Answer: B

**Question 114 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #89 (SRO) |
| System ID: | 9727 |
| User ID: | SRNLF-00-K06 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 4.00 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | OP-AA-106-101 |
| User Text: | 295003 2.1.14 |
| User Number 1: | 2.50 |
| User Number 2: | 3.30 |
| Comment: | New question. Higher. Work Week Manager notification is required upon entry into a 72 hrs or less shutdown LCO. The others do not require notifying the Duty Engineering Manager. |

Ref provided in exam.

#114

# SIGNIFICANT EVENT REPORTING

## 1. PURPOSE

1.1. This procedure describes the protocol to be used for reporting occurrences and significant events to ensure proper response is initiated both onsite and offsite, and to ensure that appropriate management is promptly informed of the event or occurrence. It further delineates the Regional Operating Group (ROG) Duty Officers' responsibilities and qualifications; specifically, those of the Duty Station Manager, Nuclear Duty Officer, and Duty Executive.

## 2. TERMS AND DEFINITIONS

2.1. Station Duty Team is a list of pre-designated on-call personnel.

## 3. RESPONSIBILITIES

3.1. Duty Executive

3.1.1. The Duty Executive is responsible to facilitate rapid event response for significant events.

3.2. Nuclear Duty Officer (NDO)

3.2.1. The Nuclear Duty Officer is the designated representative of the ROG management, responsible for initial notification of management of an operational event or occurrence at one of the sites, as well as initial ROG event response and augmentation as required based on the nature of the event.

3.2.2. The NDO is appointed by the ROG Senior Vice President and/or the Chief Nuclear Officer (CNO).

3.2.3. The Nuclear Duty Officer should be a previously-licensed/certified Senior Reactor Operator. The NDO should have experience as an Operations Manager, Operating Engineer, Shift Manager, or Control Room Supervisor.

3.3. Duty Station Manager

3.3.1. The Duty Station Manager is the designated representative of station management to whom initial notification of an event or occurrence is made by the Shift Manager.

3.5.6. When duty personnel are interfacing with the individual sites on emergent issues, the NDO should be updated as necessary.

4. **MAIN BODY**

4.1. Declaration of Emergency Plan (EP) Classification

4.1.1. For declarations of any EP classification, notifications shall be made in accordance with applicable site emergency plan procedures.

1. Initial notification to the NDO shall be made by the Duty Station Manager or Transmission Operations dispatcher/System Operations dispatcher.
2. The NDO should immediately call the affected station to obtain plant status.
3. The NDO shall promptly report any EP event classification to the Duty Executive and the Chief Nuclear Officer.
4. The CNO shall notify the CEO of an EP emergency declaration in a time-frame consistent with the impact of the event.

4.2. Other Events Requiring Regulatory or Offsite Notification

4.2.1. The Shift Manager will notify the Duty Station Manager for any of the events listed in Attachment 1.

1. **If** the Duty Station Manager cannot be reached, **then** the Shift Manager shall ensure notifications are made in accordance with Attachment 1.

4.2.2. The Duty Station Manager shall use Attachment 1 in determining communication requirements.

4.2.3. The Duty Station Manager is responsible for initial coordination of site response to the event or occurrence, including notification to station senior management, the NDO, and the Chief Nuclear Officer as described in step 4.2.3.6 below.

1. The Duty Station Manager is responsible to ensure that the Nuclear Duty Officer has been notified and has adequate information for communication.
2. The Duty Station Manager will mobilize onsite and offsite personnel to support the needs of the Shift Manager.
3. The Duty Station Manager will mobilize the Station Duty Team personnel upon entry into a 24-hour or less unplanned shutdown LCO. The Duty Station Manager should consider mobilizing the Duty Team upon entry into a 72-hour or less unplanned shutdown LCO.

#11



4.3.3. The CNO shall notify the CEO of events described in step 4.3.1. in a time-frame consistent with the impact of the event.

4.4. A ROG call-out list should be prepared identifying the Duty Executive, NDO, and Duty Station Manager for each site. The call-out list should contain appropriate telephone and pager numbers. This call-out list should be distributed to appropriate personnel.

4.5. Sites, ROG and corporate office should establish and maintain a call-out structure of key personnel on a rotational basis. This structure will serve as the basis for the call-out list described above.

4.6. The on-call Duty Executive, NDO, and Duty Station Manager shall be available at all times during a scheduled tour of duty.

4.6.1. To accomplish this objective, each individual shall carry an operable personal pager at all times and have ready access to telephone communications.

4.6.2. The NDO and Duty Station Manager shall be fit for duty at all times during a scheduled tour of duty.

## 5. **DOCUMENTATION** - None

## 6. **REFERENCES**

6.1. Commitments - None

6.2. Writer's References

6.2.1. ST-SO-101, Notification Procedure for Nuclear Emergencies or Drills

6.2.2. OP-AA-106-101-1001, Event Response Guidelines

## 7. **ATTACHMENTS**

7.1. Attachment 1, Notification Requirements

#114



**ATTACHMENT 1**
**Notification Requirements**
**Page 2 of 2**

| EVENT | NOTIFY |
|---|---|
| Shutdown LCO Entry, forced entry into a 72 hours or less shutdown LCO. | ☐ Site VP<br>☐ Plant Manager<br>☐ Sr. Manager Operations / Operations Manager<br>☐ Nuclear Duty Officer<br>☐ Duty Maintenance Supervisor<br>☐ Duty Engineering Manager<br>☐ Work Week Manager<br>☐ Senior Resident Inspector |
| Unplanned shutdown or load reduction | ☐Site VP<br>☐Plant Manager<br>☐Sr. Manager Operations/Operations Manager<br>☐Nuclear Duty Officer<br>☐Duty/ Shift Maintenance Supervisor<br>☐Duty Engineering Manager<br>☐Work Management Director<br>☐Work Week Manager<br>☐Senior Resident Inspector |
| Significant breakdown of plant radiological or environmental controls<br>Any serious personnel radioactive contamination requiring extensive on-site decontamination or outside assistance. | ☐Site VP<br>☐Plant Manager<br>☐Sr. Manager Operations/Operations Manager<br>☐Nuclear Duty Officer<br>☐Duty/Shift Health Physics Supv<br>☐Senior Resident Inspector |

NRC COPY #1

**115** **ID: SR-6900-K32** **Points: 1.00**

instead of telling when it is inop - where given conditions it can be determined it is inop - or NOT

Based on Sys Knowledge use T/S to determine LCO

Both units are operating at 100% power.
On 12-13-02 at 1200, the Unit 1 125 VDC Battery is declared INOP due to a failed discharge test.
The system engineer has determined that the Unit 1 125 VDC battery needs to be replaced.

How long can the Unit 1 125 VDC Battery be ~~inoperable~~ and the Alternate 125 VDC Battery NOT be in service before the Action statement requiring a Shutdown is entered and what is this based on?

A. 7 days - provides sufficient time to place the alternate power supply in service.

B. 7 days - capability of the battery chargers to supply BOTH the normal system loads and the alternate battery.

C. 72 hours - provides sufficient time to place the alternate power supply in service.

D. 72 hours - capability of the battery chargers to supply BOTH the normal system loads and the alternate battery.

Answer: C

**Question 115 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #90 (SRO) |
| System ID: | 9728 |
| User ID: | SR-6900-K32 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 3.00 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | T.S 3.8.4.C and Bases |
| User Text: | 295004G.2.2.22 |
| User Number 1: | 3.40 |
| User Number 2: | 4.10 |
| Comment: | New question. Application. 72 hours is based on time required to place the alternate battery is service. 7 days is when the battery must be restored and is based on the capacity and capability of remaining DC sources to supply the required loads. |

Ref provided in exam [But not the Bases!]

#115



## 3.8 ELECTRICAL POWER SYSTEMS

### 3.8.4 DC Sources — Operating

LCO 3.8.4 The following DC electrical power subsystems shall be OPERABLE:

a. Two 250 VDC electrical power subsystems;

b. Division 1 and Division 2 125 VDC electrical power subsystems; and

c. The opposite unit's 125 VDC electrical power subsystem capable of supporting equipment required to be OPERABLE by LCO 3.6.4.3, "Standby Gas Treatment (SGT) System," LCO 3.7.4, "Control Room Emergency Ventilation (CREV) System" (Unit 2 only), LCO 3.7.5 "Control Room Emergency Ventilation Air Conditioning (AC) System" (Unit 2 only), and LCO 3.8.1, "AC Sources — Operating."

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| A. One 250 VDC electrical power subsystem inoperable. | A.1 | Restore the 250 VDC electrical power subsystem to OPERABLE status. | 72 hours |

(continued)



NRC COPY #1

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| B. ---------NOTE--------- Only applicable if the opposite unit is in MODE 1, 2, or 3. --------------------- | B.1 Place associated OPERABLE alternate 125 VDC electrical power subsystem in service. | 72 hours |
| Division 1 or 2 125 VDC battery inoperable as a result of maintenance or testing. | AND | |
| | B.2 Restore Division 1 or 2 125 VDC battery to OPERABLE status. | Prior to exceeding 7 cumulative days per operating cycle of battery inoperability, on a per battery basis, as a result of maintenance or testing |
| C. ---------NOTE--------- Only applicable if the opposite unit is in MODE 1, 2, or 3. --------------------- | C.1 Place associated OPERABLE alternate 125 VDC electrical power subsystem in service. | 72 hours |
| Division 1 or 2 125 VDC battery inoperable, due to the need to replace the battery, as determined by maintenance or testing. | AND | |
| | C.2 Restore Division 1 or 2 125 VDC battery to OPERABLE status. | 7 days |

(continued)

ACTIONS

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| D. Division 1 or 2 125 VDC electrical power subsystem inoperable for reasons other than Conditions B or C. | D.1 | Restore Division 1 or 2 125 VDC electrical power subsystem to OPERABLE status. | 72 hours |
| | OR | | |
| | D.2 | --------NOTE--------<br>Only applicable if the opposite unit is not in MODE 1, 2, or 3.<br>--------------------<br>Place associated OPERABLE alternate 125 VDC electrical power subsystem in service. | 72 hours |
| E. Opposite unit 125 VDC electrical power subsystem inoperable. | E.1 | Restore the opposite unit 125 VDC electrical power subsystem to OPERABLE status. | 7 days |
| F. Required Action and associated Completion Time not met. | F.1 | Be in MODE 3. | 12 hours |
| | AND | | |
| | F.2 | Be in MODE 4. | 36 hours |

#115

SURVEILLANCE REQUIREMENTS

| | SURVEILLANCE | FREQUENCY |
|---|---|---|
| SR 3.8.4.1 | Verify battery terminal voltage on float charge is:<br>a. ≥ 260.4 VDC for each 250 VDC subsystem; and<br>b. ≥ 125.9 VDC for each 125 VDC subsystem. | 7 days |
| SR 3.8.4.2 | Verify no visible corrosion at battery terminals and connectors.<br>OR<br>Verify battery connection resistance is ≤ 1.5E-4 ohm for inter-cell connections and ≤ 1.5E-4 ohm for terminal connections. | 92 days |
| SR 3.8.4.3 | Verify battery cells, cell plates, and racks show no visual indication of physical damage or abnormal deterioration that could degrade battery performance. | 24 months |
| SR 3.8.4.4 | Remove visible corrosion and verify battery cell to cell and terminal connections are coated with anti-corrosion material. | 24 months |
| SR 3.8.4.5 | Verify battery connection resistance is ≤ 1.5E-4 ohm for inter-cell connections and ≤ 1.5E-4 ohm for terminal connections. | 24 months |

(continued)

SURVEILLANCE REQUIREMENTS

| | SURVEILLANCE | FREQUENCY |
|---|---|---|
| SR 3.8.4.6 | Verify each required battery charger supplies:<br><br>a. ≥ 250 amps at ≥ 250 VDC for ≥ 4 hours for the 250 VDC subsystems; and<br><br>b. ≥ 200 amps at ≥ 125 VDC for ≥ 4 hours for the 125 VDC subsystems. | 24 months |
| SR 3.8.4.7 | ------------------NOTE------------------<br>The modified performance discharge test in SR 3.8.4.8 may be performed in lieu of the service test in SR 3.8.4.7 provided the modified performance discharge test completely envelopes the service test.<br>----------------------------------------<br><br>Verify battery capacity is adequate to supply, and maintain in OPERABLE status, the required emergency loads for the design duty cycle when subjected to a battery service test. | 24 months |

(continued)

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.8.4.8 | Verify battery capacity is $\geq$ 80% of the manufacturer's rating for the 125 VDC batteries or the minimum acceptable battery capacity from the load profile for the 250 VDC batteries when subjected to a performance discharge test or a modified performance discharge test. | 60 months<br><br><u>AND</u><br><br>12 months when battery shows degradation or has reached 85% of expected life with capacity < 100% of manufacturer's rating<br><br><u>AND</u><br><br>24 months when battery has reached 85% of the expected life with capacity $\geq$ 100% of manufacturer's rating |

#115

BASES

APPLICABLE SAFETY ANALYSES (continued)

b. A worst case single failure.

The DC sources satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The DC electrical power subsystems — with: a) each 250 VDC subsystem consisting of one 250 VDC battery, one battery charger and the corresponding control equipment and interconnecting cabling supplying power to the associated unit bus, b) the Division 1 125 VDC subsystem consisting of the unit 125 VDC battery, one full capacity battery charger, a unit bus, and the corresponding control equipment and interconnecting cabling to the associated unit 125 VDC Division 1 bus, c) the Division 2 125 VDC subsystem consisting of the opposite unit 125 VDC battery, one full capacity battery charger, opposite unit buses, and all the corresponding control equipment, interconnecting cabling, and bus ties up to the unit 125 VDC Division 2 bus, and d) the opposite unit Division 2 125 VDC subsystem consisting of the unit 125 VDC battery, one full capacity battery charger, unit buses, and the corresponding control equipment, interconnecting cabling, and bus ties up to the associated opposite unit 125 VDC Division 2 bus are required to be OPERABLE to ensure the availability of the required power to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated DBA. Loss of any DC electrical power subsystem does not prevent the minimum safety function from being performed (Ref. 1).

APPLICABILITY

The DC electrical power sources are required to be OPERABLE in MODES 1, 2, and 3 to ensure safe unit operation and to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and

b. Adequate core cooling is provided, and containment integrity and other vital functions are maintained in the event of a postulated DBA.

(continued)

BASES

BACKGROUND (continued)

The DC batteries associated with each unit are housed in a ventilated room apart from its charger and distribution buses. This arrangement ensures redundant subsystems are located in an area separated physically and electrically from the other subsystems to ensure that a single failure in one subsystem does not cause a failure in a redundant subsystem. There is no sharing between redundant Class 1E subsystems such as batteries, battery chargers, or distribution buses.

The 125 VDC batteries for DC electrical power subsystems are sized to produce required capacity at 80% of nameplate rating, corresponding to warranted capacity at end of life cycles and the 100% design demand. The minimum design voltage limit is 105 V. For the 250 VDC batteries, the minimum allowable battery capacity is based on the capacity margin calculated for the design load profile. The minimum design voltage limit is 210 V.

Each DC electrical power subsystem battery charger has ample power output capacity for the steady state operation of connected loads required during normal operation, while at the same time maintaining its battery bank fully charged. Each station service battery charger has sufficient capacity to restore the battery from the design minimum charge to its fully charged state within 24 hours while supplying normal steady state loads (Ref. 1).

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the UFSAR, Chapter 6 (Ref. 4) and Chapter 15 (Ref. 5), assume that Engineered Safety Feature (ESF) systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the DGs, emergency auxiliaries, and control and switching during all MODES of operation.

The OPERABILITY of the DC subsystems is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining DC sources OPERABLE during accident conditions in the event of:

a. An assumed loss of all offsite AC power or all onsite AC power; and

(continued)

ACTIONS (continued)

C.1 and C.2

Condition C, Division 1 or 2 125 VDC battery inoperable due to the need to replace the battery as determined by maintenance or testing, represents one division with a loss of ability to completely respond to an event. It is therefore imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for complete loss of DC power to the affected division. Operation in this Condition may be needed during the operating cycle to completely replace a battery to maintain the Division 1 or 2 VDC subsystem OPERABLE for the remainder of the cycle. Condition C is modified by a Note indicating that the Condition is only applicable when the opposite unit is in MODE 1, 2, or 3.

If one of the 125 VDC batteries is inoperable, the remaining 125 VDC electrical power subsystem has the capacity to support a safe shutdown of one unit and to mitigate an accident condition in the other unit. Since a subsequent worst case single failure could, however, result in the loss of minimum necessary DC electrical subsystems to mitigate a worst case accident, continued power operation is limited. Required Action C.2 limits the time the unit can operate in this condition to 7 days. Therefore, each 125 VDC battery can be removed from service to completely replace a battery. In addition, Required Action C.1 requires the associated OPERABLE alternate 125 VDC electrical power subsystem to be placed in service. An OPERABLE alternate 125 VDC electrical power subsystem consists of the alternate 125 VDC battery and one full capacity battery charger. For the alternate 125 VDC battery to be considered OPERABLE, all SR requirements associated with the alternate 125 VDC battery must be met. (The full capacity battery charger is the same battery charger (normal or spare) associated with the normal 125 VDC electrical power subsystem.) Therefore, placement of the OPERABLE alternate 125 VDC electrical power subsystem in service will help ensure that the design basis can be met. However, the design configuration of the alternate battery is susceptible to single failure and hence, is not as reliable as the normal battery. Therefore, only a limited time of operation is allowed in this condition.

(continued)

#115

BASES

ACTIONS

C.1 and C.2 (continued)

The 72 hour Completion Time to place the associated OPERABLE alternate 125 VDC electrical power subsystem in service provides sufficient time to safely remove the Division 1 or 2 125 VDC electrical power subsystem from service and place the alternate supply in service. The 7 day Completion Time to restore the 125 VDC battery is based on the capacity and capability of the remaining DC Sources, including the enhanced capability afforded by the capability of the alternate 125 VDC electrical power subsystem to supply the required loads.

D.1 and D.2

With one Division 1 or 2 125 VDC electrical power subsystem inoperable for reasons other than Condition B or C represents one division with a loss of ability to completely respond to an event, and a potential loss of ability to remain energized during normal operation. It is therefore imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for complete loss of 125 VDC power to the affected division.

If one 125 VDC electrical power subsystem is inoperable (e.g., inoperable battery, inoperable required battery charger, or inoperable required battery charger and associated inoperable battery), the remaining 125 VDC electrical power subsystem has the capacity to support a safe shutdown and to mitigate an accident condition. Since a subsequent worst case single failure could, however, result in the loss of minimum necessary DC electrical subsystems to mitigate a worst case accident, continued power operation should not exceed 72 hours. The Completion Time of Required Action D.1 to restore the 125 VDC electrical power subsystem to OPERABLE status is based on the capacity, reliability and capability of the remaining 125 VDC subsystem.

Required Action D.2 is modified by a Note indicating that the action is only applicable if the opposite unit is not in MODE 1, 2, or 3. In this condition, the shutdown unit is under maintenance and a complete test of at least one 125 VDC subsystem may be necessary. Required Action D.2

(continued)

**116** **ID: S-0001-K18** **Points: 1.00**

Drywell pressure is 3.0 psig.
Reactor pressure is 950 psig.
Reactor water level is 24 inches lowering at 1 inch per minute with all available systems operating.
RWCU recirculation and blowdown modes are being used to control reactor pressure.
RWCU blowdown rate is 100 gpm.

What direction should the Unit Supervisor provide to the panel operators?

A. Secure RWCU recirculation mode and install jumpers to maintain RWCU blowdown mode.

B. Secure RWCU blowdown and recirculation modes.

C. Install jumpers to maintain RWCU recirculation AND blowdown modes.

D. Secure RWCU blowdown mode and install jumpers to maintain RWCU recirculation mode.

Answer: D

**Question 116 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #91 (SRO) |
| System ID: | 9729 |
| User ID: | S-0001-K18 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 4.00 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | QGA 100, Pressure Control |
| User Text: | 295009 AA2.03 |
| User Number 1: | 2.90 |
| User Number 2: | 2.90 |
| Comment: | New question. Higher. QGA 100 allows RWCU recirc mode bypassed. RWCU mode CANNOT be bypassed in this situation. |

?

Are we giving them just the specific area needed to answer question? If so, it is an easy lookup!

Ref provided in Exam - QGA-100.

If so, give all of QGA-100 with info blacked or cut out!

ok.

#116

Stabilize RPV pressure below 1060 psig using main turbine bypass valves.

- Use Alternate Pressure Control Systems (listed below) if needed.

Alternate Pressure Control Systems

- ADS valves... <u>only if</u> torus level is above 5 ft (QCOP 0203-01)
  - Use preferred sequence if you can.
  - If Drywell Pneumatics lost, place switch for "A" ADS valve in AUTO.
- HPCI (QCOP 2300-06)
  - **CAUTION: Exceeding NPSH/Vortex Limits may cause system damage.**
  - Use CCST suction if you can.
- RCIC (QCOP 1300-02)
  - **CAUTION: Exceeding NPSH/Vortex Limits may cause system damage.**
  - Use CCST suction if you can.
- RWCU, blowdown mode (QCOP 1200-07)
- RWCU, recirculation mode (QCOP 1200-11)
  - Bypass filter/demins.
  - OK to defeat isolations, including SBLC (QCOP 1200-02).
- Main steam line drains (QCOP 0250-05)

**117** | **ID: S-0001-K12** | **Points: 1.00**

Which of the following set of conditions would result in the Reactor Building to torus Vacuum Breakers opening when Drywell Sprays were initiated?

| | Drywell Pressure | Drywell Temperature |
|---|---|---|
| A. | 8 psig | ~~250~~ 300 degrees F |
| B. | 8 psig | 200 degrees F |
| C. | 6 psig | 200 degrees F |
| D. | 6 psig | ~~250~~ 300 degrees F |

Answer: D

Instead of just asking what cond. they can initiate DW sprays — Give a set of conditions in the stem which requires certain action based on procedure —

**Question 117 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #92 (SRO) |
| System ID: | 9730 |
| User ID: | S-0001-K12 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 2.00 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | QGA 200, Detail K |
| User Text: | 295012 AA2.02 |
| User Number 1: | 3.90 |
| User Number 2: | 4.10 |
| Comment: | New question. Application. 6 psig and 250 degrees F are the only conditions left of the DSIL curve, all others are to the right. |

Ref provided in Exam — If only provide the specific Graph DSIL curve too easy look up? why is it discriminating?

#117



E. Drywell sprays

1. Drywell sprays are initiated when torus pressure exceeds 5 psig to prevent chugging (discussed earlier).

2. Initiate sprays only if torus level is below 17 ft.

   a. Corresponds to the elevation of the torus-to-drywell vacuum breakers less the vacuum breaker opening pressure in feet of water.

   b. The vacuum breakers may not function properly if any part is submerged.

   c. Operating drywell sprays with the vacuum breakers submerged could cause the containment differential pressure capability to be exceeded.

   d. The opening pressure is subtracted from the elevation since the water level in the downcomers (on the drywell side of the vacuum breakers) will rise as drywell pressure decreases below torus pressure.

3. Initiate sprays only if drywell pressure and temperature are inside the Drywell Spray Limit.

   a. Limits the evaporative cooling pressure drop to ensure that drywell pressure will not drop below:

      - The drywell pressure scram setpoint.

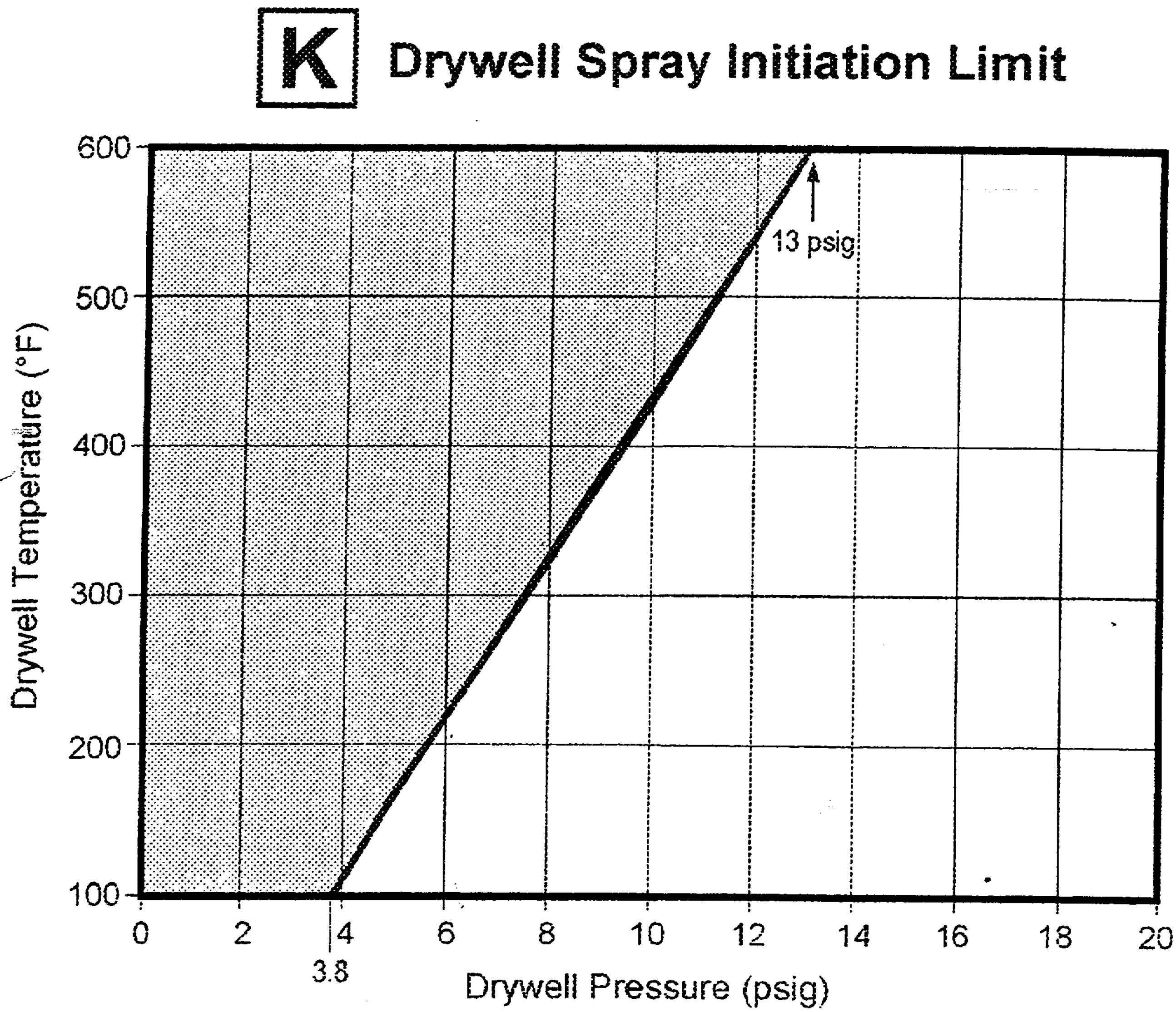
      1) Initiating sprays outside the curve could challenge containment integrity or draw in air through the reactor building-to-torus vacuum breakers.

      2) Limiting the final pressure to the scram setpoint provides sufficient margin to permit manual action before pressure drops below atmospheric.

   b. The derivation of the Drywell Spray Limit is discussed in the *Calculations* lesson plan.

   c. Applies only to spray initiation, not operation.

#117




NRC COPY #1

**118** **ID: S-1000-K33** **Points: 1.00**

Which of the following combinations of equipment will satisfy Tech Spec definition of two operable RHR Shutdown Cooling subsystems WITHOUT reliance on either RHR or RHRSW Cross-tie Valve. (Assume closed and inoperable).

| | RHR Pumps | RHR/RHRSW Heat Exchanger | RHRSW Pumps |
|---|---|---|---|
| A. | B & D | B | B & D |
| B. | B & C | B | B & C |
| C. | A & B | A | A & B |
| D. | A & C | A | A & C |

Answer: C

**Question 118 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #93 (SRO) |
| System ID: | 9810 |
| User ID: | S-1000-K33 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 4.00 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | TS 3.4.7, 3.4.8 & Bases |
| User Text: | 295021G.2.2.25 |
| User Number 1: | 2.50 |
| User Number 2: | 3.70 |
| Comment: | New question. Higher. Bases requires 1 RHR pump, 1 HX and 1 RHRSW pump for a subsystem. The answer is the only example that contains 2 subsystems. |

Ref provided in exam (TS)

But not the Bases

#118

## 3.4 REACTOR COOLANT SYSTEM (RCS)

### 3.4.7 Residual Heat Removal (RHR) Shutdown Cooling System—Hot Shutdown

LCO 3.4.7 Two RHR shutdown cooling subsystems shall be OPERABLE.

-----NOTE-----
One RHR shutdown cooling subsystem may be inoperable for up to 2 hours for the performance of Surveillances.
--------------

APPLICABILITY: MODE 3, with reactor steam dome pressure less than the RHR cut-in permissive pressure.

ACTIONS

-----NOTES-----
1. LCO 3.0.4 is not applicable.

2. Separate Condition entry is allowed for each RHR shutdown cooling subsystem.
---------------

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One or two RHR shutdown cooling subsystems inoperable.<br>need basis to identify operable subsystem | A.1 Initiate action to restore RHR shutdown cooling subsystem(s) to OPERABLE status.<br><br>AND | Immediately<br><br>(continued) |



NRC COPY #1

#118

## B 3.4 REACTOR COOLANT SYSTEM (RCS)

### B 3.4.8 Residual Heat Removal (RHR) Shutdown Cooling System—Cold Shutdown

BASES

BACKGROUND

Irradiated fuel in the shutdown reactor core generates heat during the decay of fission products and increases the temperature of the reactor coolant. This decay heat must be removed to maintain the temperature of the reactor coolant ≤ 212°F in preparation for performing Refueling maintenance operations, or the decay heat must be removed for maintaining the reactor in the Cold Shutdown condition.

The two redundant, manually controlled shutdown cooling subsystems (loops) of the RHR System provide decay heat removal. Each loop consists of two motor driven pumps, a heat exchanger, and associated piping and valves. Both loops have a common suction from the same recirculation loop. Each pump discharges the reactor coolant, after circulation through the respective heat exchanger, to the reactor via the associated recirculation loop. The RHR heat exchangers transfer heat to the RHR Service Water System.

APPLICABLE SAFETY ANALYSES

Decay heat removal by operation of the RHR System in the shutdown cooling mode is not required for mitigation of any event or accident evaluated in the safety analyses. Decay heat removal is, however, an important safety function that must be accomplished or core damage could result.

The RHR Shutdown Cooling System meets Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

Two RHR shutdown cooling subsystems are required to be OPERABLE. An OPERABLE RHR shutdown cooling subsystem consists of one OPERABLE RHR pump, one heat exchanger, the associated piping and valves, and the necessary portions of the RHR Service Water System capable of providing cooling water to the heat exchanger. The two subsystems have a common suction source and are allowed to have a common heat exchanger and common discharge piping. Thus, to meet the LCO, both RHR pumps in one loop (and two RHR service water pumps) or one RHR pump (and one RHR service water pump) in

(continued)



NRC COPY #1

#118

BASES

LCO (continued)

each of the two loops must be OPERABLE. Since the piping and heat exchangers are passive components that are assumed not to fail, they are allowed to be common to both subsystems. Each shutdown cooling subsystem is considered OPERABLE if it can be manually aligned (remote or local) in the shutdown cooling mode for removal of decay heat. In MODE 4, one RHR shutdown cooling subsystem can provide the required cooling, but two subsystems are required to be OPERABLE to provide redundancy.

Note 1 allows both RHR shutdown cooling subsystems to be inoperable during hydrostatic testing. This is necessary since the RHR Shutdown Cooling System is not designed to operate at the Reactor Coolant System pressures achieved during hydrostatic testing. This is acceptable since adequate reactor coolant circulation will be achieved by operation of a reactor recirculation pump and since systems are available to control reactor coolant temperature. Note 2 allows one RHR shutdown cooling subsystem to be inoperable for up to 2 hours for the performance of Surveillance tests. These tests may be on the affected RHR System or on some other plant system or component that necessitates placing the RHR shutdown cooling subsystem in an inoperable status during the performance. This is permitted because the core heat generation can be low enough and the heatup rate slow enough to allow some changes to the RHR shutdown cooling subsystems or other operations requiring RHR flow interruption and loss of redundancy.

APPLICABILITY

In MODE 4, the RHR Shutdown Cooling System must be OPERABLE to ensure it can be operated in the shutdown cooling mode to remove decay heat to maintain coolant temperature below 212°F.

In MODES 1 and 2, and in MODE 3 with reactor steam dome pressure greater than or equal to the RHR cut-in permissive pressure, this LCO is not applicable. Operation of the RHR System in the shutdown cooling mode is not allowed above this pressure because the RCS pressure may exceed the design pressure of the shutdown cooling piping. Decay heat removal at reactor pressures greater than or equal to the RHR cut-in permissive pressure is typically accomplished by condensing the steam in the main condenser. Additionally, in MODE 2

(continued)

**119** **ID: S-0001-K24** **Points: 1.00**

The reactor was operating at 100% power when a RPV leak occurred in the Drywell.
The reactor was manually scrammed at 2.0 psig.
All rods fully inserted.

what's normal?

Current plant conditions include the following:

- RPV pressure is 920 psig and lowering slowly.
- RPV level is 0" and rising slowly.
- Drywell pressure is 5 psig and rising.
- Drywell temperature is 225 degrees F and rising.
- Torus pressure is 16 psig and rising.
- Torus level is ~~normal.~~ [illegible] 1.13 ft
- Torus sprays are operating.
- ~~Drywell sprays have NOT been attempted.~~ ok leave with

WHICH of the following actions is ~~appropriate~~ required per the QGAs?

A. Perform an RPV Blowdown.

B. Vent the Primary Containment.

C. Spray the Drywell ~~due to high Drywell Temp~~.

D. Do Not Spray the Drywell ~~due to high Torus Pressure.~~ but continue spraying Torus

Answer: A

Ref provided QGA 200

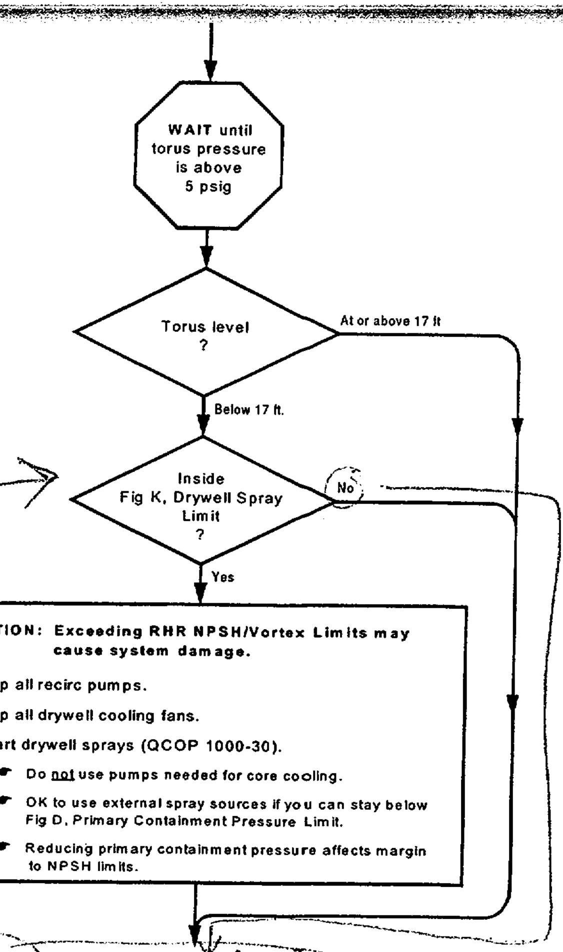
should only Give the set of Graphs DSIL, PSP etc. - not the whole QGA

Question does not exactly meet K/A However need Torus level (what's normal?)

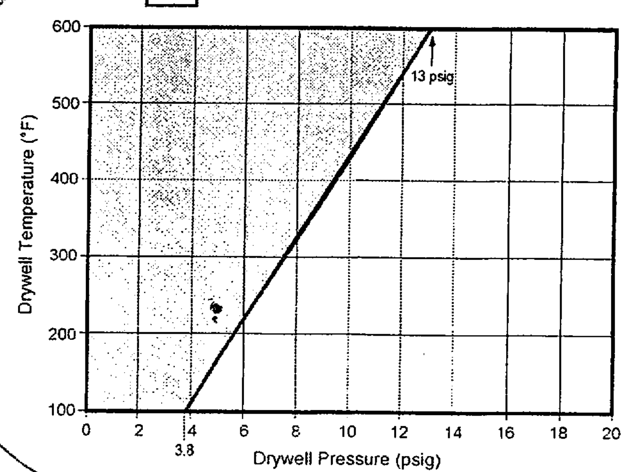
**Question 119 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #94 (SRO) |
| System ID: | 9754 |
| User ID: | S-0001-K24 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 4.00 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | QGA 200 |
| User Text: | 295028EA2.05 |
| User Number 1: | 3.60 |
| User Number 2: | 3.80 |
| Comment: | New question. Higher. RPV blowdown required due to about to exceed PSP. CANNOT spray the Drywell due to being to the left of the DSIL curve. Cannot vent because at this point there is no direction to exceed off-site release rates. |

#119




**K** Drywell Spray Initiation Limit






NRC COPY #1

#119

from pg 22

Continue to control drywell and torus pressures.

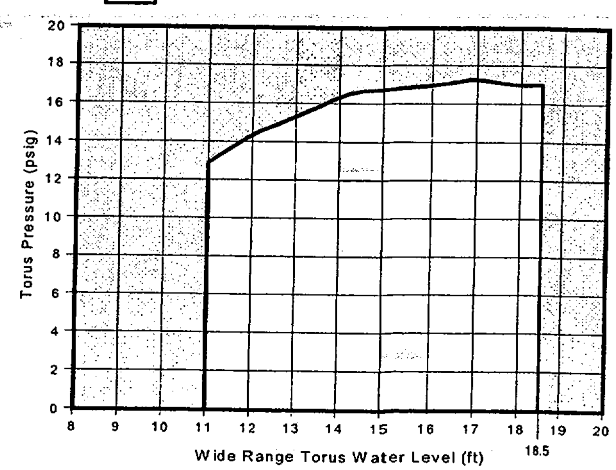
| IF | THEN |
| --- | --- |
| Cannot stay inside Fig L, Pressure Suppression Pressure | Go to ⑯. |

⑯

**BLOW DOWN:**

Enter QGA 500-1 while continuing here.

**L** Pressure Suppression Pressure






NRC COPY #1

# 119



E. Drywell sprays

1. Drywell sprays are initiated when torus pressure exceeds 5 psig to prevent chugging (discussed earlier).

2. Initiate sprays only if torus level is below 17 ft.

   a. Corresponds to the elevation of the torus-to-drywell vacuum breakers less the vacuum breaker opening pressure in feet of water.

   b. The vacuum breakers may not function properly if any part is submerged.

   c. Operating drywell sprays with the vacuum breakers submerged could cause the containment differential pressure capability to be exceeded.

   d. The opening pressure is subtracted from the elevation since the water level in the downcomers (on the drywell side of the vacuum breakers) will rise as drywell pressure decreases below torus pressure.

3. Initiate sprays only if drywell pressure and temperature are inside the Drywell Spray Limit.

   a. Limits the evaporative cooling pressure drop to ensure that drywell pressure will not drop below:

      - The drywell pressure scram setpoint.

      1) Initiating sprays outside the curve could challenge containment integrity or draw in air through the reactor building-to-torus vacuum breakers.

      2) Limiting the final pressure to the scram setpoint provides sufficient margin to permit manual action before pressure drops below atmospheric.

   b. The derivation of the Drywell Spray Limit is discussed in the *Calculations* lesson plan.

   c. Applies only to spray initiation, not operation.

G. Blow down

1. Continue to control drywell and torus pressures.

   a. Continue to control pressures with sprays. All restrictions and operating details continue to apply.

   b. No other action is required as long as pressure can be held below the Pressure Suppression Pressure.

2. Pressure Suppression Pressure

   a. The Pressure Suppression Pressure is the lower of:

      1) The highest torus pressure that can occur without steam in the torus airspace.

      2) The highest torus pressure at which a blowdown will not result in exceeding the Primary Containment Pressure Limit before RPV pressure drops to the Minimum RPV Flooding Pressure.

      3) The highest torus pressure at which ADS valves can be opened without exceeding the torus boundary design load.

   b. The highest torus pressure that can occur without steam in the torus airspace is limiting at QCNPS.

   c. The curve defines a maximum torus pressure as a function of torus water level.

   d. If pressure is above the curve, pressure suppression capability may not be available.

      1) To the left of the curve, the downcomers are uncovered.

      2) To the right of the curve, the downcomers may not clear properly following a LOCA.

      3) Above the curve, bypass leakage may exist.

   e. The derivation of the Pressure Suppression Pressure is discussed in the *Calculations* lesson plan.

3. Blow down

a. If pressure suppression capability is not available, the RPV must be depressurized.

b. Depressurization minimizes further release of energy from the RPV to the primary containment.

c. Perform the blowdown in accordance with QGA 500-1.

d. QGA 100 entry will already be required due to high drywell pressure. The reactor will already be scrammed.

# EXAMINATION ANSWER KEY



**120** **ID: S-0001-K24** **Points: 1.00**

Do Not need OK QGA-200

The following post LOCA conditions are present:

- All Rods in.
- The only operable ECCS pump is 1B Core Spray.
- Reactor pressure is 275 psig and slowly lowering.
- Reactor level is being maintained at -150 inches by:
  - 2 Condensate/Condensate Booster pumps with a combined flow of 0.5 MLBM/hr.
  - 1B Core Spray pump with a flow of 4600 GPM.
- Hotwell level is being maintained by Standby Coolant injection.
- All Torus water level indication has failed, however LI1-1640-21 "Primary Containment Water Level" indicates 16.75 feet and rising.

clean up margin condition

What is the next required action and why?

A. Secure the condensate pumps because adequate core cooling is assured.

B. Start both remaining condensate pumps to raise reactor water level above the top of active fuel.

C. Secure all operating injection pumps because adequate core cooling is assured.

D. ~~Start a third condensate pump~~ Initiate SBLC to raise reactor water level above the top of active fuel.

This is understanding in the why of performing mitigating actions EOP Basis quest

Answer: A

Basically similar answer over or two cond pump start Basic premise is to raise wtr level - Also why aren't those acceptable? (TAF? level -142)

**Question 120 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #95 (SRO) |
| System ID: | 9811 |
| User ID: | S-0001-K24 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 4.00 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | QGA 100 / 200 |
| User Text: | 295029EA2.03 |
| User Number 1: | 3.40 |
| User Number 2: | 3.50 |
| Comment: | New question. Higher. Per the EOP basis, one core spray pump injecting at 4600 gpm ~~is adequate~~ core cooling. |

QGA-200 ref has no support to the answer

#120

| Content/Skills | Activities/Notes |
|---|---|

C. High Level (#21)

1. Hold torus level below 17 ft.

   a. Corresponds to the elevation of the bottom of the torus-to-drywell vacuum breakers, less vacuum breaker opening pressure in feet of water.

   b. Preserves vacuum breaker operability and permits continued operation of drywell sprays.

   c. The vacuum breakers may not function properly if any part is submerged.

   d. The opening pressure is subtracted from the elevation since the water level in the downcomers (on the drywell side of the vacuum breakers) will rise as drywell pressure decreases below torus pressure.

2. Cannot hold level below 17 ft.

   a. Stop drywell sprays.

      1) Operating sprays with the vacuum breakers submerged could cause the containment differential pressure capability to be exceeded.

      2) The Primary Containment Pressure and Drywell Temperature branches permit spray initiation only if torus water level is below 17 ft.

   b. Terminate injection from sources outside the primary containment not needed for core cooling.

      1) If a primary system break exists, continuing RPV injection from sources outside the primary containment will increase torus water level.

      2) Terminating injection prevents further level increase.

      3) Do *not* terminate:

         a) Injection needed for core cooling. Core cooling takes precedence and terminating drywell sprays minimizes the risk of containment damage.

         b) Injection from the torus, since it will not increase primary containment water level.

         c) Injection from systems being used to shut down the reactor (CRD and SBLC).

Q# 120



**CAUTION: Exceeding NPSH/Vortex Limits may cause system damage.**

Control RPV water level above -142 in. (TAF) using any of the Preferred Injection Systems (listed above).

- Use Alternate Injection Systems if needed (Detail E).

| IF | THEN |
|---|---|
| Cannot restore level above -166 in. and hold it there **AND** Neither Core Spray loop flow is at or above 4500 gpm | 1. Maximize Injection using Preferred and Alternate Injection Systems (Detail E). |
| Cannot restore level at or above -191 in. and hold it there | 2. IF ........you still cannot restore and hold level above -166 in., AND ....TSC prepared to provide SAMG decision-making, THEN..**FLOOD CONTAINMENT**: Exit all QGAs → Enter all SAMGs |



NRC COPY #1

Content/Skills | Activities/Notes

#120

H. Enter SAMGs

1. Post-blowdown strategies (in order of preference):

   S-0001-K18

   a. Evaluate the effectiveness of preferred injection systems at low pressure.

   b. Establish and maintain spray cooling conditions.

   **Change in criteria for transfer to SAMGs and containment flooding.**

   c. Maximize injection with preferred and alternate injection systems.

   d. Flood the primary containment.

2. No further action is required following the blowdown if either:

   - Operating injection systems can hold RPV water level above the MSCRWL (-166 in.) at the lower pressure, or
   - Spray cooling can be established (at least one Core Spray loop injecting at or above design flow with RPV water level at or above the top of the jet pumps).

3. If adequate core cooling *still* cannot be restored, enter the SAMGs.

4. The SAMGs provide integrated guidance for extremely degraded conditions beyond the scope of the QGAs.

5. If a break exists, flooding the primary containment will backfill the RPV through the break.

6. The SAMGs are used by the TSC. Continue efforts to restore additional injection sources and increase RPV injection until the TSC is manned and assumes command and is prepared to provide SAMG decision-making.

7. Do not align direct injection into the primary containment until directed to do so by the TSC.

8. Exit *all* QGAs.

**121** **ID: SR-1601-K29** **Points: 1.00**

Torus level is lowering as indicated on the narrow range torus level indication.

Which of the following list the HIGHEST Torus level during operation in Mode 1 which would require entering a Tech Spec LCO and why?

A. -1" due to excessive suppression pool TEMPERATURE during a DBA LOCA.

B. -3" due to excessive suppression pool TEMPERATURE during a DBA LOCA.

C. -3" due to excessive suppression pool SWELL LOADS during a DBA LOCA.

D. -1" due to excessive suppression pool SWELL LOADS during a DBA LOCA.

Answer: B

**Question 121 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #96 (SRO) |
| System ID: | 9762 |
| User ID: | SR-1601-K29 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 3.00 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | TS 3.6.2.2 and bases |
| User Text: | 295030G2.1.33 |
| User Number 1: | 3.40 |
| User Number 2: | 4.00 |
| Comment: | New question. Higher. TS bases 3.6.2.2, pg. 1 states that lower volume in the Torus would result in less steam energy being absorbed before heating up excessively. Higher level results in excessive swell loads. Torus TS level is + or - 2 inches. |

SRO 96

121

#121



B 3.6 CONTAINMENT SYSTEMS

B 3.6.2.2 Suppression Pool Water Level

BASES

BACKGROUND

The suppression chamber is a toroidal shaped, steel pressure vessel containing a volume of water called the suppression pool. The suppression pool is designed to absorb the energy associated with decay heat and sensible heat released during a reactor blowdown from relief valve discharges or from a Design Basis Accident (DBA). The suppression pool must quench all the steam released through the downcomer lines during a loss of coolant accident (LOCA). This is the essential mitigative feature of a pressure suppression containment, which ensures that the peak containment pressure is maintained below the maximum allowable pressure for DBAs (62 psig). The suppression pool must also condense steam from the steam exhaust lines in the turbine driven systems (i.e., High Pressure Coolant Injection (HPCI) System and Reactor Core Isolation Cooling (RCIC) System) and provides the main emergency water supply source for the reactor vessel. The suppression pool volume ranges between approximately 111,500 $ft^3$ at the low water level limit of 14 ft 1 inch and approximately 115,000 $ft^3$ at the high water level limit of 14 ft 5 inches.

If the suppression pool water level is too low, an insufficient amount of water would be available to adequately condense the steam from the relief valve quenchers, downcomer lines, or HPCI and RCIC turbine exhaust lines. Low suppression pool water level could also result in an inadequate emergency makeup water source to the Emergency Core Cooling System. The lower volume would also absorb less steam energy before heating up excessively. Therefore, a minimum suppression pool water level is specified.

If the suppression pool water level is too high, it could result in excessive clearing loads from relief valve discharges and excessive pool swell loads during a DBA LOCA. Therefore, a maximum pool water level is specified. This LCO specifies an acceptable range to prevent the suppression pool water level from being either too high or too low.

(continued)



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121

SC096

#121



## 3.6 CONTAINMENT SYSTEMS

### 3.6.2.2 Suppression Pool Water Level

LCO 3.6.2.2 Suppression pool water level shall be ≥ 14 ft 1 inch and ≤ 14 ft 5 inches.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| A. Suppression pool water level not within limits. | A.1 | Restore suppression pool water level to within limits. | 2 hours |
| B. Required Action and associated Completion Time not met. | B.1 | Be in MODE 3. | 12 hours |
| | <u>AND</u> | | |
| | B.2 | Be in MODE 4. | 36 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.6.2.2.1 | Verify suppression pool water level is within limits. | 24 hours |

**122** **ID: S-1601-K33** **Points: 1.00**

Torus level was normal when a large unisolable leak developed on the bottom of the Torus

Which of the following describes the expected change in Drywell to Torus delta-P and the validity of the safety analysis?

Drywell to Torus Delta-P will increase until Torus level reaches _(1)_ feet and then equalize.

Safety analysis assumptions are valid until Torus level reaches _(2)_ feet.

| | (1) | (2) |
|---|---|---|
| ~~A~~. | 11' | 14'1" |
| ~~B~~. | 5' | 11' |
| ~~C~~. | 5' | 14'1" |
| ~~D~~. | 11' | 11' |

Answer: ~~A~~ B

## Question 122 Details

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #97 (SRO) |
| System ID: | 9761 |
| User ID: | S-1601-K33 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 4.00 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | T.S. 3.6.2.2 & Bases |
| User Text: | 295030EA2.04 |
| User Number 1: | 3.50 |
| User Number 2: | 3.70 |
| Comment: | New question. Lower. Downcomers are 3.67 to 4 feet below the surface of the Torus. This means the downcomers will be uncovered at ~ 10 feet, at which point Drywell to Torus DP will stop increasing. TS 3.6.2.2, pg 2 states that safety analysis conditions are not met if Torus water level is outside of the limits. |

100
97

122

BASES (continued)

APPLICABLE SAFETY ANALYSES

Initial suppression pool water level affects suppression pool temperature response calculations, calculated drywell pressure for a DBA, calculated pool swell loads for a DBA LOCA, and calculated loads due to relief valve discharges. Suppression pool water level must be maintained within the limits specified so that the safety analysis of Reference 1 remains valid.

Suppression pool water level satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

LCO

A limit that suppression pool water level be ≥ 14 ft 1 inch and ≤ 14 ft 5 inches above the bottom of the suppression chamber is required to ensure that the primary containment conditions assumed for the safety analyses are met. Either the high or low water level limits were used in the safety analyses, depending upon which is more conservative for a particular calculation.

APPLICABILITY

In MODES 1, 2, and 3, a DBA would cause significant loads on the primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. The requirements for maintaining suppression pool water level within limits in MODE 4 or 5 is addressed in LCO 3.5.2, "ECCS-Shutdown."

ACTIONS

A.1

With suppression pool water level outside the limits, the conditions assumed for the safety analyses are not met. If water level is below the minimum level, the pressure suppression function still exists as long as the downcomers are covered, HPCI and RCIC turbine exhausts are covered, and relief valve quenchers are covered. If suppression pool water level is above the maximum level, protection against overpressurization still exists due to the margin in the peak containment pressure analysis and the capability of the RHR Suppression Pool Spray System. Therefore, continued operation for a limited time is allowed. The 2 hour Completion Time is sufficient to restore suppression pool water level to within limits. Also, it takes into account the low probability of an event impacting the suppression pool water level occurring during this interval.

(continued)



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# EXAMINATION ANSWER KEY

2002 Quad Cities NRC Exam

**123** | **ID: S-GSEP-P01** | **Points: 1.00**

(maybe shorten the answer. obvious ?!

Field teams have been dispatched due to a Radioactivity Release.
The field teams are located as follows:
Field Team #1 is at the Cribhouse.
Field Team #2 is ~~15 feet East of Highway 84 and Site~~ Access Road intersection.
Field Team #3 is at the Meteorological Tower.
Field Team #4 is at the Hydrogen Farm.

only one that is off-site

Which of the following describes which Field Team(s) is (are) OFF-SITE for the purposes of Emergency Classification?

A. Field Team #2 only

B. Field Teams #2 & #4.

C. Field Teams #3 & #4.

D. Field Team #2 & #3
~~All Field Teams.~~
~~Field Team #3 only~~

lets pick another location

Why high level? Is this memory? recognizing what is outside the site.

Answer: ~~A~~ D

## Question 123 Details

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #98 (SRO) |
| System ID: | 9815 |
| User ID: | S-GSEP-P01 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 3.00 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | ODCM, Figure 1 |
| User Text: | 295038EA2.01 |
| User Number 1: | 3.30 |
| User Number 2: | 4.30 |
| Comment: | Bank question (Clinton 2002 exam #125) Higher. The Cribhouse, Met Tower and Hydrogen Farm ~~are~~ all considered on site. The East side of Rte. 84 is off-site. |

Show where the choices are located

Ref provided See site map ODCM

#123

## 12.4 GASEOUS EFFLUENTS

### 12.4.1 Dose Rate

Operability Requirements

12.4.1.A The dose rate in unrestricted areas (at or beyond the site boundary, see Quad Cities Station ODCM Annex, Appendix F, Figure F-1) due to radioactive materials released in gaseous effluents from the site shall be limited to the following:

1. For Noble Gases:

   (a) Less than 500 mrem/year to the whole body.
   (b) Less than 3000 mrem/year to the skin.

2. For iodine-131, for iodine 133, and for all radionuclides in particulate form with half-lives greater than 8 days less than 1500 mrem/year.

Action:

If the dose rates exceed the above limits, without delay decrease the release rates to bring the dose rates within the limits, and to provide prompt notification to the Commission (12.6)

Surveillance Requirements

12.4.1.B The dose rates due to radioactive materials released in gaseous effluents from the site shall be determined to be within the prescribed limits by obtaining representative samples in accordance with the sampling and analysis program specified in Table 12.4-1. The dose rates are calculated using methods prescribed in the Offsite Dose Calculation Manual (ODCM).

Bases

12.4.1.C This specification provides reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a Member of the Public in an Unrestricted Area, either at or beyond the Site Boundary in excess of the design objectives of appendix I to 10 CFR part 50. This specification is provided to ensure that gaseous effluents from all units on the site will be appropriately controlled. It provides operational flexibility for releasing gaseous effluents to satisfy the Section II.A design objectives of appendix I to 10 CFR part 50. For Members of the Public who may at times be within the Site Boundary, the occupancy will usually be sufficiently low to compensate for the reduced atmospheric dispersion of gaseous effluents relative to that for the Site Boundary. Examples of calculations for such Members of the Public, w ith appropriate occupancy factors, shall be given in the ODCM. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to an individual at or beyond the unrestricted area boundary to less than or equal to a dose rate of 500 mrem/year to the total body or to not less than or equal to a dose rate of 3000 mrem/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to a child via the inhalation pathway to not less than or equal to a dose rate of 1500 mrem/year. For purposes of calculating doses resulting from airborne releases the main chimney is considered to be an elevated release point, and the reactor vent stack is considered to be a mixed mode release point.



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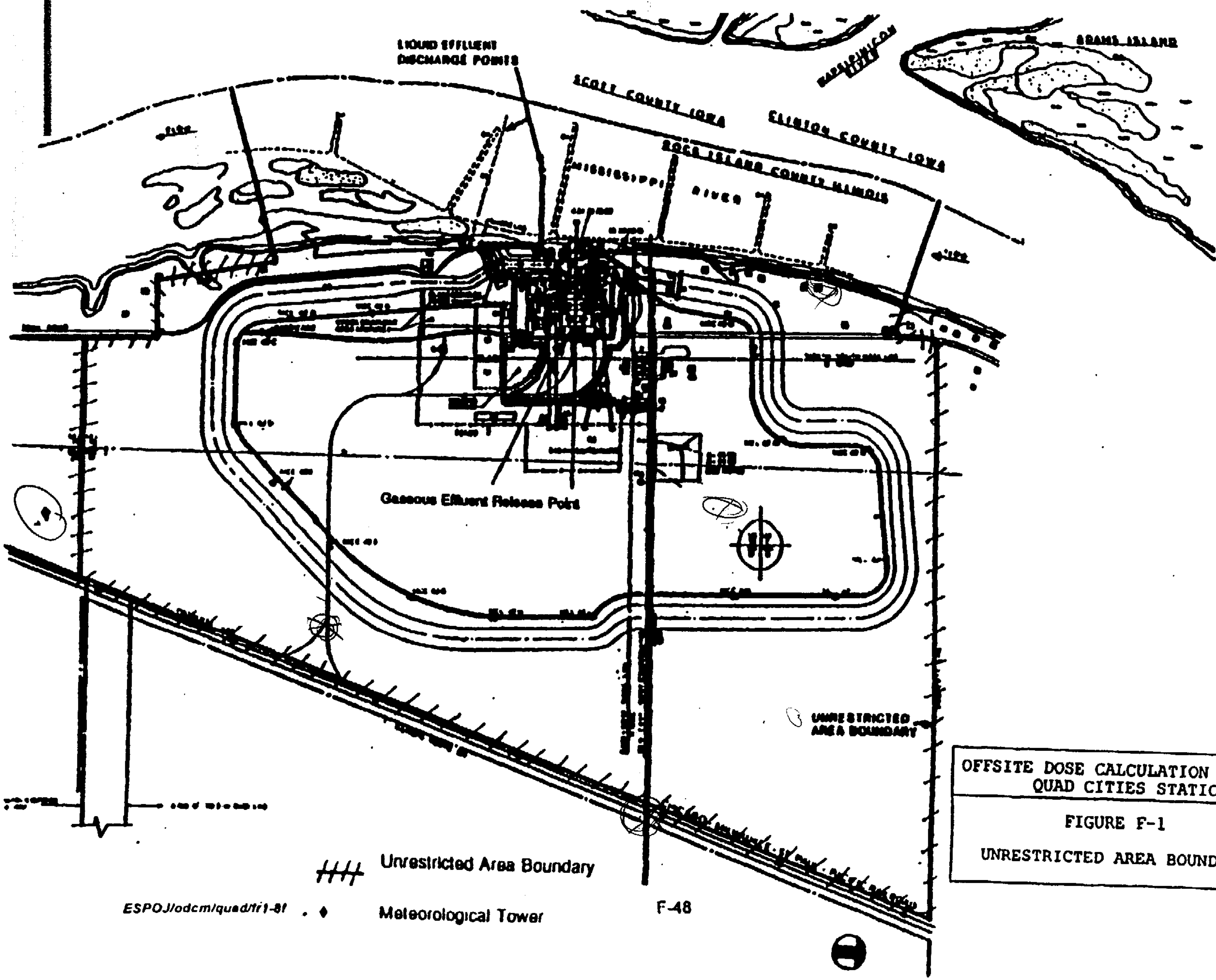


OFFSITE DOSE CALCULATION MANUAL
QUAD CITIES STATION

FIGURE F-1

UNRESTRICTED AREA BOUNDARY

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#123

#123



2.25 Site Boundary: Defined as a circle with a radius of ¼ or ½ mile (depending on the site specific ODCM) and the containment building as it's center.

2.26 Station Vent: That part of the plant's ventilation system through which the containment building and auxiliary building air may be processed to the outside atmosphere. The discharge of the station vent is continuously monitored for abnormal amounts of radiation and would be isolated long before radiation levels approach federal limits.

2.27 Subareas: Pre-designated areas offsite in which Protective Actions such as evacuation of sheltering will be performed.

2.28 Total Effective Dose Equivalent (TEDE): A method of converting exposure to radiation to the biological effects that it will cause to the human body. It combines the external and internal ionizing radiation exposure. The TEDE is the sum of Deep Dose Equivalent and Committed Effective Dose Equivalent.

## 3. RESPONSIBILITIES

3.1.1 The **Shift Manager**, or designated on-shift individual, shall perform required dose assessments prior to responsibility being transferred to either the Technical Support Center (TSC) or Emergency Operations Facility (EOF).

3.1.2 The **TSC Radiological Controls Coordinator** shall relieve the Control Room and perform required assessments if the transfer of PAR / dose assessment responsibilities to the EOF is delayed.

3.1.3 The **EOF Dose Assessor** shall relieve the TSC Radiological Controls Coordinator when directed by the EOF Dose Assessment Coordinator, and perform required dose assessments. Responsibility for dose assessments can be assumed directly from the Control Room.

## 4. MAIN BODY

4.1 Dose Assessment and Protective Action Recommendation (DAPAR)

- **REFER** to Attachment 1 for user guidelines.

4.2 MESOREM 99 (B-Model)

- **REFER** to Attachment 2 for user guidelines.

4.3 C-Model, Corrected Containment Radiation Level and Radioactivity

- **REFER** to Attachment 3 for user guidelines.

**124** **ID: SL-801-K33** **Points: 1.00**

According to Tech Specs Bases, fuel handling is restricted with low fuel pool levels because during a refueling accident __________ could not be assumed.

A. adequate cooling of irradiated fuel bundles seated in the reactor vessel

B. net positive suction head for fuel pool cooling pumps

C. adequate cooling of irradiated fuel bundles seated in the spent fuel pool

D. absorbtion of water soluable fission product gasses

Answer: D

**Question 124 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #99 (SRO) |
| System ID: | 9812 |
| User ID: | SL-801-K33 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 4.00 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | TS Bases 3.7.8 R. 0 pg. 1 |
| User Text: | 295023AA2.02 |
| User Number 1: | 3.40 |
| User Number 2: | 3.70 |
| Comment: | New question. Fundamental. Absorbtion of water soluable fission product gasses per TS 3.7.8 pg. 1. |

124



B 3.7 PLANT SYSTEMS

B 3.7.8 Spent Fuel Storage Pool Water Level

BASES

BACKGROUND

The minimum water level in the spent fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident.

A general description of the spent fuel storage pool design is found in the UFSAR, Section 9.1.2 (Ref. 1). The assumptions of the fuel handling accident are found in Reference 2.

APPLICABLE SAFETY ANALYSES

The water level above the irradiated fuel assemblies is an explicit assumption of the fuel handling accident. A fuel handling accident is evaluated to ensure that the radiological consequences (calculated whole body and thyroid doses at the exclusion area and low population zone boundaries) are ≤ 25% of 10 CFR 100 (Ref. 3) exposure guidelines NUREG-0800 (Refs. 4 and 5) and less than the 10 CFR 50, Appendix A, GDC 19 limits (Ref. 6). A fuel handling accident could release a fraction of the fission product inventory by breaching the fuel rod cladding as discussed in Regulatory Guide 1.25 (Ref. 7).

The fuel handling accident is evaluated for the dropping of an irradiated fuel assembly onto the reactor core. The water level in the spent fuel storage pool provides for absorption of water soluble fission product gases and transport delays of soluble and insoluble gases that must pass through the water before being released to the secondary containment atmosphere. This absorption and transport delay reduces the potential radioactivity of the release during a fuel handling accident.

The spent fuel storage pool water level satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The specified water level preserves the assumptions of the fuel handling accident analysis (Ref. 2). As such, it is the minimum required for fuel movement within the spent fuel storage pool.

(continued)



NRC COPY #1

124

#124



BASES (continued)

APPLICABILITY

This LCO applies during movement of irradiated fuel assemblies in the spent fuel storage pool or whenever movement of new fuel assemblies occurs in the spent fuel storage pool with irradiated fuel assemblies seated in the spent fuel storage pool, since the potential for a release of fission products exists.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply. If moving fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of fuel assemblies is not a sufficient reason to require a reactor shutdown.

When the initial conditions for an accident cannot be met, action must be taken to preclude the accident from occurring. If the spent fuel storage pool level is less than required, the movement of fuel assemblies in the spent fuel storage pool is suspended immediately. Suspension of this activity shall not preclude completion of movement of a fuel assembly to a safe position. This effectively precludes a spent fuel handling accident from occurring.

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1

This SR verifies that sufficient water is available in the event of a fuel handling accident. The water level in the spent fuel storage pool must be checked periodically. The 7 day Frequency is acceptable, based on operating experience, considering that the water volume in the pool is normally stable, and all water level changes are controlled by unit procedures.

(continued)

**125** **ID: S-4100-K32** **Points: 1.00**

In order for the Fire Diesels to be considered operable, __(1)__ Diesel Engine must be capable of auto starting and driving its associated fire pump.

__(2)__ Diesel Driven Fire Pump(s) must be operable in order to consider the Fire Protection Water Supply System operable.

(1) (2)

A. One; One

B. Each; Each

C. One; Each

D. Each; One

Answer: D

**Question 125 Details**

| | |
|---|---|
| Question Type: | Multiple Choice |
| Topic: | Question #100 (SRO) |
| System ID: | 9753 |
| User ID: | S-4100-K32 |
| Status: | Active |
| Must Appear: | No |
| Difficulty: | 4.00 |
| Time to Complete: | 0 |
| Point Value: | 1.00 |
| Cross Reference: | QCAP 1500-01, R17, D.5,6 |
| User Text: | 600000G2.2.25 |
| User Number 1: | 2.50 |
| User Number 2: | 3.70 |
| Comment: | New question. Lower. This is per QCAP 1500-01 and not a TS bases because fire prot. was removed from TS. When discussed at outline submittal, permission was given to keep this KA and ask a question on the admin procedure. In order for both Fire Diesels to be considered operable, they both must be operable. One Fire Diesel is required to be operable in order for fire suppression to be operable. |

a better K/A - also ~~for the~~ quest in generic. ~~K/A~~ 600000 concerning fire - would be G 2.2.2x (operability)

Adjust SRO outline & replace K/A choice & note in

#125



## D.5. Diesel Driven Fire Pumps:

a. Operability Requirements:

(1) Both diesel driven fire pumps and engine auxiliaries **SHALL** be operable at all times.

(2) Each diesel engine **SHALL** be capable of automatically starting when required by the automatic initiation logic and driving its associated fire pump.

(3) **IF** a single diesel driven fire pump is declared inoperable, **THEN** compensatory and reporting requirements **SHALL** be followed per steps D.5.c and D.5.d.

b. Surveillance Requirements:

(1) Fire pump diesel starting 24 volt battery bank and charger shall be demonstrated OPERABLE:

| Description | Frequency |
|---|---|
| Verify electrolyte level of each battery is above plates and that overall battery voltage is $\geq$ 24 volts. | Monthly |
| Verify batteries, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration, and the battery-to-battery and terminal connections are clean, tight, free of corrosion, and coated with anti-corrosion material. | 18 months |

(2) **IF** batteries fail any of above tests, **THEN** batteries and associated diesel engine must be considered inoperable until an evaluation of operability is performed. Compensatory and reporting requirements shall be followed per steps D.5.c and D.5.d.

#125



D.5.d. (cont'd)

(2) **IF** diesel driven fire pump is found to be inoperable by unplanned circumstances, **THEN** a CR (nonreportable) **SHALL** be generated describing inoperable condition.

## D.6. Fire Protection Water Supply System:

a. Operability Requirements:

(1) Fire protection water supply system **SHALL** be operable at all times. Fire protection water supply system contains following major components:

(a) At least one (1) diesel driven fire pump and control circuitry.

(b) A flow path capable of taking suction from the Mississippi River and transferring water through distribution piping.

(c) One flow path **SHALL** be operable and capable of delivering water to both Unit 1 and 2 Reactor Building and Turbine Building main loops for the fire protection water suppression system to be considered operable.

(2) **IF** system declared inoperable, **THEN** compensatory and reporting requirements **SHALL** be followed per steps D.6.c and D.6.d.

#175



D.6. (cont'd)

d. Reporting requirements:

(1) **IF** system is found to be inoperable by unplanned circumstances, **THEN** a CR (nonreportable) **SHALL** be generated describing inoperable condition.

(2) **IF** system is inoperable **AND** backup water supply is **NOT** established within 24 hours, **THEN** a CR (reportable under SAF 1.1--4 hour **AND** SAF 1.14--LER) **SHALL** be generated.

## D.7. Fire Hose:

a. EFP and NFP operability requirements:

(1) All fire hose stations listed in Attachment L **SHALL** be operable at all times.

(2) The following components are required to be operable:

(a) Hose station valve.

(b) Fire hose.

(c) Nozzle.

b. Fire Truck operability requirements:

(1) Staged within:

(a) The Protected Area and available.

**OR**

(b) Within the Owner Controlled Area **AND** continuously manned.

(2) Equipped with 1000 feet of Fire Hose of varying sizes.

(3) Fittings, as appropriate, to use/connect fire hydrant.