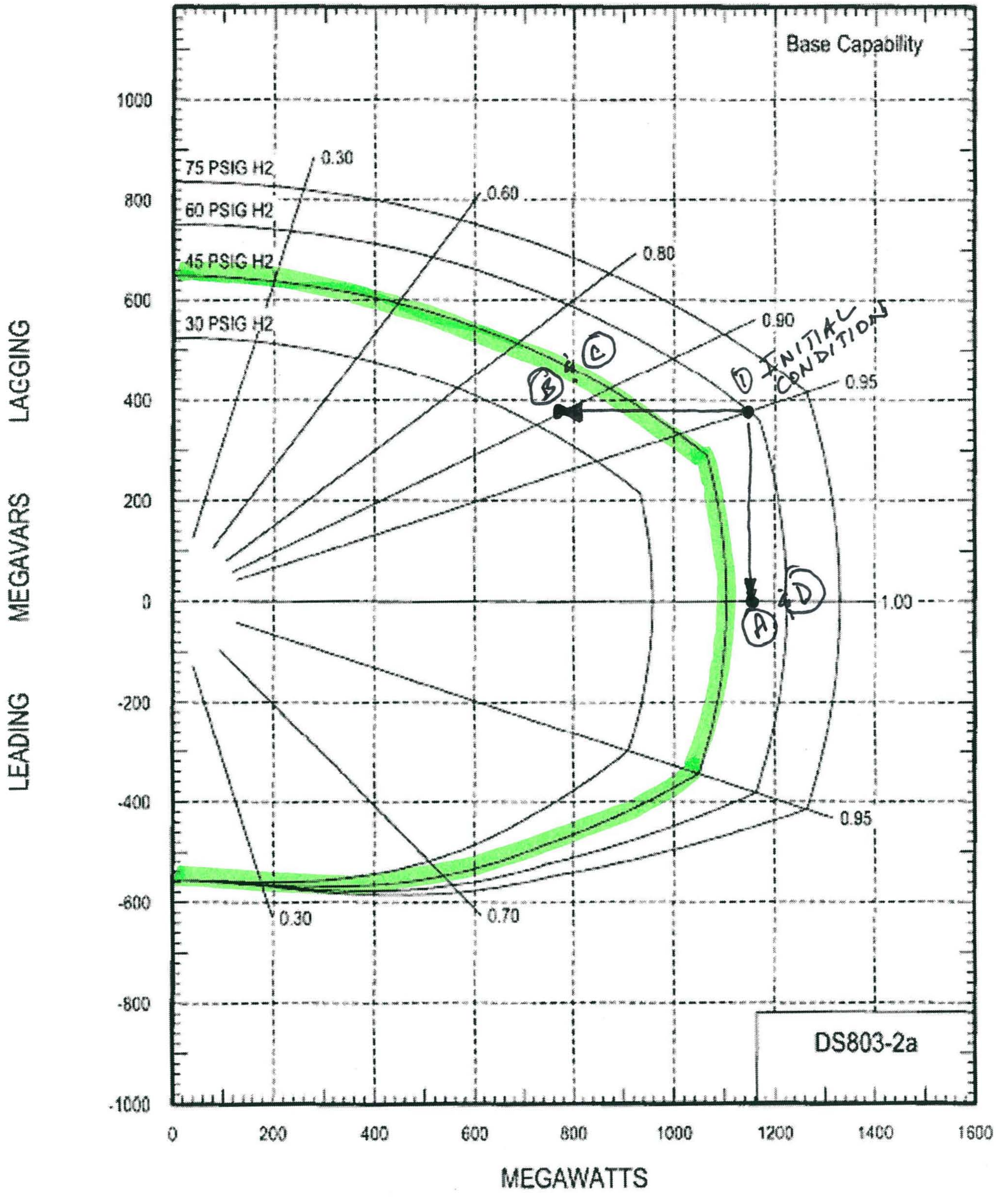
ESTIMATED REACTIVE CAPABILITY CURVES

4 Pole 1800 RPM 1330000 kVA 22000 Volts 0.950 PF

0.580 SCR 75.00 PSIG H2 Pressure 510 Volts Excitation

46 Deg. C Cold Gas 617 Ft. Altitude



EXAMINATION REFERENCE PROVIDED TO CANDIDATE

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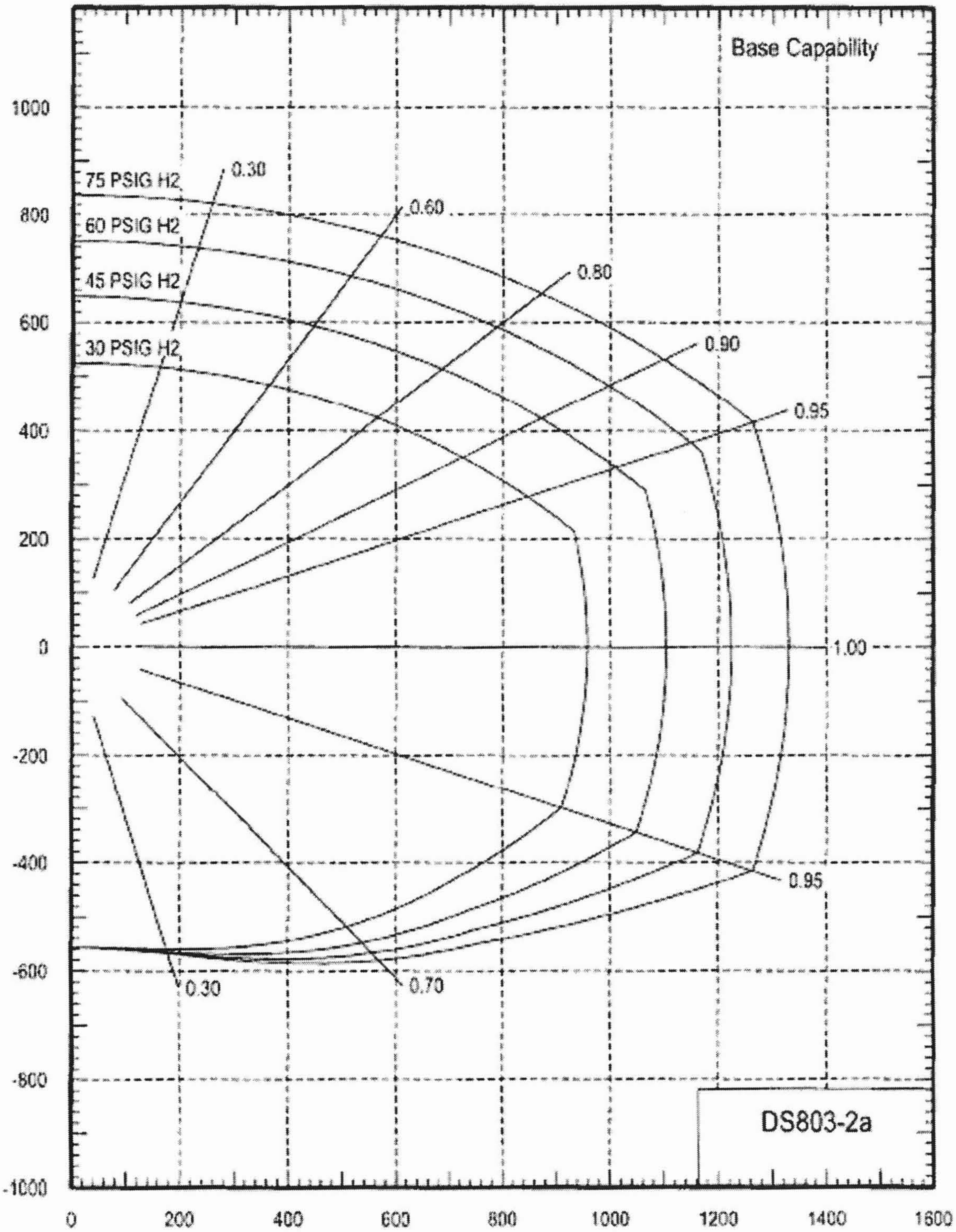

36. RO 268000A2.01 001/MEM/T2G2///268000A2.01//RO/SRO/11/28/07 RMS

Given the following plant conditions:

- Unit 1, 2 and 3 are in operation at Rated power.
- While processing the floor drain sample tank through the Thermx system the following conditions are noted:
  - Floor Drain Collector Tank Level Rising
  - Unit 3 Reactor Building Floor Drain Sump Level High High is received
  - Unit 3 Announces on PA System that Unit 3 has scrammed.

Which ONE of the following is the expected response by the RADWASTE operator?

A.✔ Notify the Unit 3 Unit Supervisor of an EOI-3 Entry condition on Unit 3.

B. Notify the Unit 3 Unit Supervisor that an EOI-3 Entry condition exists, but the affected Unit cannot be determined from RADWASTE.

C. Notify the Shift Manager that an EOI-3 Entry condition exists, but the affected Unit cannot be determined from RADWASTE.

D. Control room notification is not appropriate during a scram transient since redundant alarms are available in the affected control room.

K/A Statement:
268000 Radwaste
A2.01 - Ability to (a) predict the impacts of the following on the RADWASTE ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: System rupture

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions to determine the response of the RADWASTE system due to a rupture of a plant system and the procedures used to mitigate that condition.

References:

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

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<u>REFERENCE PROVIDED:</u> None

<u>Plausibility Analysis:</u>

In order to answer this question correctly the candidate must determine the following:

1. Whether Radwaste can determine which unit has an EOI entry condition.
2. Based on the answer to Item 1, determine the reporting requirements.

<u>A is correct.</u>

<u>B is incorrect.</u> This is plausible because the Unit 3 US is required to be notified, however the affected unit CAN be determined from the RADWASTE control room.

<u>C is incorrect</u>. This is plausible if the candidate fails to recognize that the affected unit CAN be determined from the RADWASTE control room. In addition, the RADWASTE operator is required by procedure to notify the AFFECTED unit control room.

<u>D is incorrect.</u> This is plausible because each control room has a **DRYWELL** FLOOR DRAIN SUMP HIGH LEVEL alarm, but no alarm for the Reactor Building. In addition, control room notification of EOI entry conditions is MANDATORY, especially during transient conditions.

Instructor Notes

(c) Hi-Hi level starts both pumps & brings in Hi Level Alarm.

(2) Alternate relay swaps pump start circuit.

d. Alarms

(1) Low alarms correspond to automatic pump stop. Maintaining adequate sump level is necessary to ensure airborne contamination is minimized in the area containing the sump. Airborne contamination could become significant if the discharges into the sump were not covered by water.

(2) High alarm corresponds to the level at which one of the sump pumps should start.

(3) High-High alarm corresponds to the level at which both sump pumps should have automatically started.

Hi-Hi Alarm @ 66"

(4) The Reactor Building floor drain sumps are monitored for entry into EOI-3 (Secondary Containment Control). The alarms for these sumps are only located in the Radwaste Control Room (panel 25-17).

Obj. V.B.4

Note: RB equip. drain sump is not listed as EOI-3 parameter

(a) There is one common alarm for all three units for Reactor Building floor drain sump level High-High.

(b) There is one common alarm for all three units for Reactor Building equipment drain sump level High-High.

Instructor Notes

(c) The radwaste operator can determine which unit has the high-high sump level by the "REACTOR UNIT ONE", "REACTOR UNIT TWO", or "REACTOR UNIT THREE" alarms. The unit with both alarms in would have the high-high sump level.

(d) The radwaste operator **must** notify the affected unit of the EOI entry condition as directed by the ARP.

4. Sump locations that pump into radwaste.

a. Turbine building floor and equipment drain sumps located north end of condenser room.

b. Condensate pump pit floor and equipment drain sumps located west side of each condensate pump pit.

c. Backwash receiver pit sump located in receiver pit on north wall, and discharges to FDCT.

d. Reactor building equipment drain sump located north-east quadrant basement reactor building.

e. Reactor building floor drain sumps.

(1) 2 per unit - one pump per sump.

(2) Cross connected by 8 inch line at overflow and 6 inch line at normal level.

(3) Located in south-east and south-west quad.

f. Drywell floor and equipment drain sumps located north end of drywell basement.

Instructor Notes

g. Radwaste equipment and floor drain sumps located north wall in basement of Radwaste.

h. Off-Gas sump located northwest end of Radwaste Basement.

i. Standby gas treatment building sumps located in standby gas treatment building.

j. Off-Gas Building sump located in basement of off-gas building.

k. Evaporator Building sump located south wall of the first floor of the evaporator building.

NOTE: Unit station sump pump normally discharges to CCW discharge conduit but may be lined up to pump to radwaste FDCT through normally closed valve at sump.

5. Sump Pump Rating

a. Motors - 480V AC

b. All sumps have two pumps per sump except reactor building floor drain sumps which have one.

37. RO 272000K5.01 001/C/A/SYS/HWC/B9/272000K5.01//RO/SRO/

Given the following plant conditions:

- The HWC System is in the Operator Determined Setpoint mode.
- Hydrogen flow is set at 14 SCFM.

Which ONE of the following describes the plant response if reactor power is reduced?

A.✔ MSL radiation levels will rise in opposition to the lowering of reactor power due to a rise in volitile Ammonia production.

B. MSL radiation levels will lower in response to the lowering of reactor power due to a reduction in Nitrogen concentration.

C. MSL radiation levels will lower in response to the lowering of reactor power due to a reduction in Hydrogen concentration.

D. MSL radiation levels will rise in opposition to the lowering of reactor power due to a rise in Nitrite and Nitrate production.

K/A Statement:
272000 Radiation Monitoring
K5.01 - Knowledge of the operational implications of the following concepts as they apply to RADIATION MONITORING SYSTEM : Hydrogen injection operation's effect on process radiation indications: Plant-Specific

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use plant conditions to determine the effect on radiation levels due to specific operating conditions of the Hydrogen Injection system.

References:

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. The mechanism by which Hydrogen injection causes MSL radiation levels to rise higher than normal.
2. The effect on hydrogen concentration in the reactor at reduced feedwater flow with Hydrogen Injection flowrate unchanged.
3. The effect on ammonia production due to a reduction of oxygen concentration in the reactor.

A is correct.

B is incorrect. This is plausible because nitrogen concentration DOES decrease with power level. However, due to the reduction in oxygen concentration, the reduction in nitrite and nitrate production allows more nitrogen available to combine with the excess hydrogen and form volitile ammonia.

C is incorrect. This is plausible because of the TYPICAL response of the HWC system when operated in Automatic mode. In addition, understanding that hydrogen injection flowrate is constant may not lead to an understanding of that relationship to a reduction in feedwater flow where hydrogen is injected.. However, in Operator Demand mode the hydrogen concentration increases due to the reduction in feedwater flow.

D is incorrect. This is plausible because MSL radiation levels DO rise in opposition to lowering power. However, the reduction in available oxygen causes a reduction in nitrite and nitrate production, which allows more volitile ammonia formation.

h. O_2 source

(1) Air in-leakage - oxygen in the air leaks into the low pressure parts of the steam cycle

LP turbine blading to discharge of condensate pumps is less than atmospheric pressure

(2) Some air in-leakage is removed by the SJAE's but some dissolves in the condensate

(3) Radiolysis reaction -

$$2\,H_2O \xrightarrow{\text{n and } \gamma \text{ radiation}} 2\,H_2 + O_2$$

One way O_2 is produced

i. O_2 removal

(1) Radiation induced recombination of H_2 and O_2

Reaction is an equilibrium reaction

$$2H_2O \xleftrightarrow{\text{n and gamma radiation}} 2H_2 + O_2$$

(2) Carry-over with the steam

j. Hydrogen addition

(1) When an excess of hydrogen is injected to the feed water, the reaction is driven to the left and less oxygen (and peroxide) is produced

This is the basis of H_2 injection

(2) The chemical environment becomes less oxidizing

(3) Elements exposed to the coolant will assume chemical forms using less oxygen and/or more hydrogen

(4) Solubility and volatility may be affected by the change in "oxidation state" of the element

This is how MSL radiation levels will increase which will be discussed later

G. Operation

Be very careful when selecting functions from different screens. Use self checking.

1. Normally controlled from the HWC Main Control Panel

   a. When using the Operator Interface Unit (OIU) function buttons, be aware that the same function key will cause different actions on different screens

INPO SER 3-05

   b. Operation of the HWC PLC

      (1) Hydrogen controller

Flow controller operates in 2 modes

         (a) Automatic/Power Determined Setpoint Mode - changes hydrogen injection flow in response to changes in reactor power. Used for normal operation of the HWC System and when reducing hydrogen injection related dose rates to support maintenance, chemistry or radcon activities while the plant is operating

Procedure Use

         (b) Automatic/Operator Determined Setpoint Mode - changes hydrogen injection flow in response to the setpoint being manually entered by the operator. Normally used when initially pressurizing, purging and placing the HWC System in service or if Power Determined setpoint is unavailable

Hydrogen flow stays constant, regardless of power changes, until operator manually enters a new setpoint

      (2) Oxygen controller - Automatic/Hydrogen Determined Setpoint

Only mode used for oxygen control

n. Supply Facility Trip - A shutdown signal is generated when either the hydrogen or oxygen gas supply facility trips

o. Hydrogen or Feedwater Flow Signal Failed - A shutdown signal is generated when the hydrogen flow signal or the feedwater flow signal is less than 2 mA or greater than 22 mA

I. Radiological Effects of HWC On MSL's

Obj. V.B.6
Obj. V.B.4
Obj. V.D.6
Obj. V.E.6

1. The primary source of background MSL radiation levels during reactor operation is due to the decay of nitrogen-16 (N^{16})

   a. N^{16} has a half-life of 7.1 seconds

   b. A 6.13 or 7.12 Mev gamma is emitted on N^{16} decay

   6.13 Mev gamma is more common

2. Major source of nitrogen in a BWR is O^{16} (η,ρ) N^{16} reaction

   $O^{16} + \eta \rightarrow N^{16} + \rho$

3. When using normal water chemistry methods, a major portion of the N^{16} present in the reactor coolant combines with the free oxygen to form water soluble nitrites (NO_2) and nitrates (NO_3)

   a. These compounds are circulated through the reactor coolant systems and are ultimately removed by the RWCU System

   b. A smaller fraction of the N^{16} is carried over in the steam in the form of nitrogen gas (N_2) and ammonia (NH_3)

   Predominate contributor to background radiation levels

4. H_2 injection alters the N^{16} carryover ratio

   a. Concentrations of NO_3, NO_2, and NO decrease

   b. Concentration of NH_3 increases

   Ammonia

(1) A gas

(2) High water solubility

5. The net production of N^{16} is not influenced by hydrogen injection

6. The increased dose rates are due to the increased ease with which N^{16} gets out of the reactor and into the steam pipes when in the NH_3 form

We can maintain up to 2.7 ppm injection concentration.

7. The initial U2 run was the first week in Nov. 1999. Up to 90 scfm hydrogen was injected. Average MSL radiation level increased approximately 5 times normal

MSL 'B' was highest at 5.2 times normal

8. Addition of noble metals to reactor water

   a. Noble metals decompose during reactor startup or shutdown

   Rubidium and Iridium

   b. During this time it produces a thin layer of noble metal on wetted surfaces

   c. The ECP on these surfaces are reduced significantly during subsequent operation

   d. This leaves a stoichiometric excess of hydrogen

   e. Now the amount of hydrogen injection can be reduced which will lower MSL radiation levels

This NOTE

c. Consequences of Event

No effects were noted. However there is the potential for rapid recombination in the Offgas Charcoal beds. Additionally excessive hydrogen increases the risk for explosion or fire. .

2. High radiation on reduction in power event at Monticello

a. Event description

On December 13, 1997 with Monticello at approximately 75 percent power, workers entered the main condenser room to repair a leaking root valve and found the dose rate 2.5 times greater than expected

Reactor power had been reduced from 100 percent power to 75 percent power for ALARA purposed and hydrogen water chemistry injection rate had been reduced from the normal 40 scfm to 8 scfm

Dose rate encountered was significantly higher that expected

Encountered - 4,800 mrem/hr
Expected - 2,000 mrem/hr

Job was stopped to evaluate the situation and management decided to have power reduced further

At 60 percent power, dose rates were about 3,200 mrem/hr and the job was completed

b. Cause of event

Lack of understanding of the radiological effect of reducing reactor power under HWC conditions. As reactor power is decreased, less N-16 is produced, and steam line dose rates decrease. However, power level changes also change feedwater flow rate and hydrogen concentration

Questioning Attitude could have prevented this.

Reactor power and feedwater hydrogen concentration both affect steam line dose rates. At a constant hydrogen injection rate, as power is decreased, feedwater flow rate decreases, and hydrogen concentration increases

An increase in hydrogen concentration increases the ammonia concentration although hydrogen injection rates were reduced, the hydrogen concentration increase that occurred when feedwater flow rate decreased with reactor power was not accounted for and resulted in higher than expected dose rates

c. Consequences of Event:

Rx power needed to be lowered to 60% vice 75% planned. This resulted in unplanned lost generation.

Work Planning

Work had to be performed in a 3200mr/hr vice a planned 2000 mr/hr field. This resulted in unplanned exposure.

Potential lost opportunity to plan and perform work that required Rx power to be lowered to 60 % power vice the planned reduction to 75% power

38. RO 290003A3.01 001/MEM/T2G2/HVAC/4/290003A3.01/3.3/3.5/RO/SRO/

Given the following plant conditions:

- High radiation has been detected in the air inlet to the Unit 3 control room.
- Radiation Monitor RE-90-259B is reading 250 cpm.

Which ONE of the following describes the CREV System response?

A. Neither CREVS unit will auto start at that radiation level.

B. Both CREVS units will auto start with suction from the normal outside air path to elevation 3C.

C.✔ The selected CREVS Unit will auto start; the standby CREVS Unit will begin to auto start, but will only run if the selected CREVS Unit fails to develop sufficient flow.

D. The selected CREVS Unit will auto start and will continue to run until Control Bay Ventilation is restarted, then it will automatically stop.

K/A Statement:
290003 Control Room HVAC
A3.01 - Ability to monitor automatic operations of the CONTROL ROOM HVAC including: Initiation/reconfiguration

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions to determine the effect on CREV initiation logic.

References:

Level of Knowledge Justification: This question is rated as MEM due to the requirement to recall or recognize discrete bits of information.

0610 NRC Exam

REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. The initiation setpoint for CREV.
2. The initiation sequence for the selected CREV unit.
3. The method required to secure a CREV unit once started automatically.

A is incorrect. This is plausible because the Tech Spec initiation setpoint is 270 cpm, which is less than the given radiation level. However, the actual CREV initiation setpoint is 221 cpm.

B is incorrect. This is plausible since both CREV units receive a start signal on a valid initiation. However, the CREV unit NOT selected will experience a 30 second time delay on initiation and will only complete its start sequence if the selected CREV unit fails to start.

C is correct.

D is incorrect. This is plausible because the start sequence is correct. However, once initiated, CREV must be manually secured. There is no automatic shutdown capability, only trips.

INSTRUCTOR NOTES

7. Control Room Emergency Ventilation (CREV) is designed to supply and process the outdoor air needed for pressurization during isolated conditions. There are 2 CREV units rated at 3000 cfm each. A CREV unit consists of Motor-driven fan, (power supply is from 480V RMOV Bd 1A for CREV Fan A; RMOV Bd 3B for CREV Fan B), HEPA filter (common), charcoal filter assemblies located in the CREVS Equipment Room, charcoal heater, and inlet isolation damper and a backflow check outlet damper. They are designed to maintain a positive pressurization to 1/8" w.g. minimum to the control room.

> Tech. Spec. 3.7.3
> Obj.V.B.2/ V.B.5/ V.C.6 /V.C.7
> (Old CREV Units abandoned in place as Auxiliary Pressurization Systems)
> TP-4
> 2-47E2865-4

a. A CREV may be started manually from control room Panel 2-9-22 if local control switch is in AUTO position via a 3 position, spring-return to center switch. (STOP-AUTO-START). Actuates only the CREVS unit & associated damper, not the isolation dampers.

> Red indicating lights on panel 3-9-21 to provide indication of CREV Fan A and/or B running on Unit 3. Annunciators are on panel 9-6 for all units.

b. There is also a 2 position maintained contact, one per train, AUTO-INITIATE/ TEST switch which is used to perform system level actions for that train (primarily testing). It provides the same response as auto start.

c. Local start at local control station in relay room is done using a 2 position maintained, one per train, AUTO-TEST switch. Isolation dampers do not operate automatically if started from local panel.

d. Automatic start signals are:

> Obj. V.B.1/V.B.2

(1) High radiation of 221 cpm above background (270 cpm Tech Specs) in air inlet ducts to control room from (Radiation monitor RE 90-259A Units 1 & 2, Radiation monitor RE 90-259B Unit 3). Either monitor starts selected CREV unit.

> Obj. V.C.1
> Obj. V.C.17
> T. S. 3.3.7.1

(2) Reactor zone ventilation systems radiation high $\geq$72 MR/hr

(3) Refuel zone ventilation systems radiation high ≥72 MR/hr

(4) Low reactor water level at +2 inches above instrument zero

(5) High primary containment pressure ≥2.45 psig

e. On receipt of a start signal, normal outside air paths (see below) to elevation 3C are isolated. The selected CREV unit starts once the inlet damper is full open. This supplies pressurizing air to the Unit 1, 2 and 3 control rooms. One CREV unit can supply all three control rooms, so the STBY CREV unit will not normally start. Once started, the CREV unit will continue to run until manually secured by first clearing the high radiation signals and the PCIS signals (otherwise equipment cycling will occur)

f. Control bay (EL 617) isolation is accomplished by five pneumatic and motor-operated low leakage dampers which isolate all normal air intakes and exhausts for EL 617.

(1) FCO-31-150B, fresh air make-up duct to Units 1 and 2 Control Room and Relay Room AHU.

(2) FCO 31-150G, 3C elevation relief vent isolation

(3) FCO-31-150E, exhaust from Unit 1 toilet, locker, and other rooms at elevation 617.

(4) FCO-31-150D, mounted in fresh air makeup duct to Unit 3 Control Room AHU.

(5) FCO-31-150F, exhaust from unit 3 toilet, locker, and other rooms at elevation 617.

INSTRUCTOR NOTES

The inlet damper is normally closed & fails closed. Damper opening takes ~70 seconds. While in the intermediate position both red & green lights will be lit on 2-9-22.

The unit heater will energize 10 sec. after the damper is full open to allow the fan to come up to speed. High Rad or PCIS signal will energize relays in Div I (CR1-A) and Div II (CR1-B). Contacts from the CR1 relays are used to energize solenoids to isolate the M.C.R. normal intake dampers (150B,D,E,F, and G)

INSTRUCTOR NOTES

g. Manual initiation of the emergency mode of operation can be performed from the control room by operation of the AUTO-INITIATE/TEST switch (putting switch in INITIATE/TEST position) at Pnl 2-9-22. This operation results in energizing the CR1 relay for that train/division and isolation of the control room dampers. Only one solenoid must be energized to close these dampers. Therefore, either test switch will initiate a damper isolation.

Adherence to procedures INPO SER 03-05 Obj.V.B.4/V.B.5

h. One switch alone in the INITIATE/TEST position will NOT result in full functionality of the CREVS units. If that train is not the selected unit, then operation of that train will be delayed by approx. 30 seconds, waiting for the selected train. This delay will result in the operator waiting to see the result of his operation of the switch. The operator will see the amber light lit, indicating energization of the CR 1 relay and the solenoid for isolation damper closing, but will see no activity of the CREVS unit until the delay timer has timed out.

Normally "A" is selected unit via switch in CREV room. If "A" is inoperable, switch 0-XSW-031-7214 SYSTEM PRIORITY SELECTOR SWITCH is placed in TRAIN B position to start it without time delay.

i. If the selected train's switch is put in the INITIATE/TEST position that train will immediately enter its initiation sequence, with the damper's red light being lit as well as the green, indicating travel of the damper toward the open position. However, should there be any failure of the selected unit; the standby unit will not start. This is because the CR1 relay for the standby unit was not actuated.

j. Therefore, when manually initiating emergency operation of the new CREVS units, it is important to put the AUTO-INITIATE/TEST switches of BOTH trains to the INITIATE/TEST position.

INSTRUCTOR NOTES

k. Again, to secure operation, the AUTO-INITIATE/TEST switches must both be returned to the AUTO position and then the STOP-AUTO-START switches turned to the STOP position, to reset the CR1 relays in both divisions.

l. Trips for the units, which are effective at all times, are the following:

(1) Fan overload

(2) Unit low flow, less than approx. 2700 cfm -- trip is delayed for 10 seconds after fan start.

(3) High heater discharge temperature, approx. 220°F

(4) Low heater delta temperature(between unit inlet and heater discharge), indicating that the heater is not getting the relative humidity below 70 % -- trip is delayed for approx. 15 seconds after the heater is energized.

m. When any of these trip signals are received, the following will occur:

Obj.V.B.4/ V.B.5

(1) The heater will be immediately deenergized.

(2) The fan will continue to run and the damper will remain open for approx. 30 seconds, to dissipate the heat from the heater. (In the case of fan overload, the fan will trip immediately.)

(3) The inlet damper will be deenergized, and when no longer fully open, the fan will be deenergized. The damper requires approx. 20 seconds to close, while fan coast down is approx. 60-90 seconds.

INSTRUCTOR NOTES

n. In addition to the trips shown above, loss of power to the inlet damper will trip the unit. In this case, the heater is immediately tripped and the fan is deenergized when the damper is no longer fully open. This action results in (slightly) faster tripping of the heater to avoid heat dissipation problems.

o. Flow switches are provided, one for each division/unit, to start the standby unit if the selected unit does not start or trips off. The selected unit not starting is sensed by low differential pressure across the common HEPA filter in the Unit 2 vent tower. Low differential pressure exists when a fan is not operating; this signal will normally be present. The circuit for each unit is such that its initiation sequence is begun upon either of the following:

PDIS 7316 at Unit 2 Vent Tower Intake Plenum

(1) Unit is selected as primary unit and CR1 relay for that division is energized.

(2) Other unit is selected as primary unit, low differential pressure exists across the common HEPA filter, and CR1 relay for that division has been energized for approx. 30 seconds.

p. With this circuit design, when an accident signal is initially received, the selected unit will enter its initiation sequence immediately and the other unit will enter its initiation sequence approx. 30 seconds later. Once the selected unit fan has been started (taking approx. 75 seconds -- 70 for the damper and 5 for the fan), the low differential pressure signal will no longer be present in the standby unit circuitry and its damper will return to the fail-close position.

INSTRUCTOR NOTES

q. If the selected unit fails to start properly, it will itself be turned off by the trips noted above, and the standby unit will continue in its initiation sequence. The time delay for startup of the standby unit will be selected to ensure that regardless of the primary unit failure, both fans will not be running at the same time.

r. If the selected unit starts properly, but then trips at a later time, the standby unit will only be missing the low differential pressure signal to receive its start signal. The standby unit will start when the selected system has completed its shutdown process and the fan has been deenergized.

s. To secure from emergency operation, the high rad signals and the PCIS signals must first be cleared (otherwise equipment cycling will occur). These signals must be cleared on both divisions to not have the standby unit start up when the selected unit is secure. The STOP-AUTO-START switches in the control room should then be moved to the STOP position <u>for both units</u>. This will reset / deenergize the CR1 relays in both divisions, reopen the control room isolation dampers and remove the start signal from the operating CREVS unit. The CREVS unit heater will then be deenergized, with the fan continuing to run and the damper held open for approx. 30 seconds, and the damper closing and the fan turned off as discussed earlier.

Obj. V.B.2.

39. RO 295001AK3.01 001/MEM/T1G1/RECIRC//295001AK3.01//RO/SRO/NEW

Given the following plant conditions:

- Unit 3 is operating at 100% power.
- The following alarm is received.
  - Recirc loop A out of Service

Which ONE of the following describes the Reactor Water Level response?

A. Lower initially due to shrink, then return to normal.

B.✔ Rise initially due to swell, then return to normal.

C. Lower initially due to shrink and remain lower due to the loss of core voids.

D. Rise initially due to swell and remain high due to a lower power level.

K/A Statement:
295001 Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4
AK3.01 - Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION : Reactor water level response

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions to determine the effect on reactor water level due to a partial loss of Recirculation flow.

References:

Level of Knowledge Justification: This question is rated as MEM due to the requirement to recall or recognize discrete bits of information.

0610 NRC Exam

REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. The internal response of the Reactor Vessel due to a Recirc Pump trip.
2. The RPV level instrument response to the conditions determined from Item 1 above.

The RPV response occurs in three parts.

First, the trip of the recirc pump causes a sudden reduction of coolant flow up through the fuel bundles while power level remains approximately 100%. This causes a sudden increase in core void fraction. The large voiding in the active fuel region coupled with the reduction in inventory being removed from the downcomer by the tripped recirc pump results in a rapid rise in RPV level outside the core shroud while water level inside the core shroud lowers. Since RPV level is measured outside the shroud, indication rises.

Next, the large void content in the active fuel region responds quickly to insert negative reactivity, causing a large reduction in reactor power and therefore, steam (void) production. This reduction in void fraction draws water from the downcomer region outside the shroud into the active fuel region inside the shroud. Even though reactor power will drop to approximately 65% with a 100% rod pattern, core void fraction at 65% is actually greater than at 100% due to the effect of recirc flow. Therefore, RPV level indication does not immediately return to its original value.

Finally, the Feedwater Control System responds to the transient by reducing feedwater flow below steam flow to enable RPV level to slowly return to the original setpoint at a lower reactor power.

A is incorrect. This is plausible because level inside the core shroud initially lowers, then returns to normal, however RPV level is not measured inside the core shroud.

B s correct.

C is incorrect. This is plausible because level inside the core shroud initially lowers, then returns to normal. In addition, the reduction of core voids is temporary. Final void fraction is actually higher. However RPV level is not measured inside the core shroud.

D is incorrect. This is plausible because RPV level initially rises in response to the recirc pump trip. However, the lower power level is compensated by automatic adjustments of Feedwater Control.

40. RO 295001G2.1.14 001/MEM/T1G1/68 - RECIRC/2/295001G2.1.14//RO/SRO/0606S NEW6/24/2007

Given the following plant conditions:

- You are the At-The-Controls (ATC) operator on Unit-1
- Unit 1 is operating at full power when 1A Recirculation pump tripped.
- The Unit Supervisor has directed you to carry out the actions of 1-AOI-68-1A, Recirc Pump Trip/Core Flow Decrease OPRMs Operable.

Which ONE of the following describes the required operator action(s) that CANNOT be carried out from your watch station?

A. IMMEDIATELY take actions to insert control rods to less than 95.2% loadline AND REFER TO 0-TI-464, Reactivity Control Plan Development and Implementation.

B.✓ Perform 1-SR-3.4.1(SLO), Reactor Recirculation System Single Loop Operation.

C. CHECK parameters associated with the Recirc Drive and Recirc Pump/Motor 1A(1B) on ICS and RECIRC PMP MTR 1A & 1B WINDING & BRG TEMPS, 1-TR-68-71 to determine the cause of trip.

D. REFER TO ICS screens VFDPMPA(VFDPMPB) and VFDAAL(VFDBAL) to help determine the cause of the recirc pump trip/core flow decrease.

K/A Statement:
295001 Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4
2.1.14 - Conduct of Operations Knowledge of system status criteria which require the notification of plant personnel

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions to determine the required actions which require notification of plant personnel outside of the control room due to a partial loss of Recirculation flow.

References: 1-AOI-68-1

Level of Knowledge Justification: This question is rated as MEM due to the requirement to recall or recognize discrete bits of information.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. Which actions are required to be performed by 1-AOI-68-1.
2. Which of the actions determined from Item 1 above CANNOT be carried out by the ATC operator.

A is incorrect. This is plausible because the BOP operator typically inserts control rods while the ATC operator executes 1-AOI-68-1 and acts as Peer Checker if possible. However, manipulating control rods is part of the ATCwatch station duties.

B is correct. This duty is carried out by Reactor Engineering.

C is incorrect. This is plausible because the BOP operator or STA typically carry out this action. However, utilizing ICS screens is available at the ATC watch station and within his required duties.

D is incorrect. This is plausible because the BOP operator or STA typically carry out this action. However, utilizing ICS screens is available at the ATC watch station and within his required duties.

4.2 Subsequent Actions (continued)

NOTE

1) Step 4.2[3] through Step 4.2[18.3] apply to any core flow lowering event.
2) Power To Flow Map is maintained in 0-TI-248, Station Reactor Engineer and on ICS.

[3] **IF** Region I or II of the Power to Flow Map is entered, **THEN** (Otherwise N/A)

IMMEDIATELY take actions to insert control rods to less than 95.2% loadline AND **REFER TO** 0-TI-464, Reactivity Control Plan Development and Implementation. ☐

[4] **RAISE** core flow to greater than 45% in accordance with 1-OI-68. ☐

[5] **INSERT** control rods to exit regions if **NOT** already exited AND **REFER TO** 0-TI-464**,** Reactivity Control Plan Development and Implementation. ☐

NOTE

The remaining subsequent action steps apply to a single Reactor Recirc Pump trip.

[6] **CLOSE** tripped Recirc Pump discharge valve. ☐

[7] **MAINTAIN** operating Recirc pump flow less than 46,600 gpm in accordance with 1-OI-68. ☐

[8] [NER/C] **WHEN** plant conditions allow, **THEN**, (Otherwise N/A)

MAINTAIN operating jet pump loop flow greater than 41×10^6 lbm/hr (1-FI-68-46 or 1-FI-68-48). [GE SIL 517] ☐

4.2 Subsequent Actions (continued)

CAUTION

The temperature of the coolant between the dome and the idle Recirc loop should be maintained within 75°F of each other. If this limit cannot be maintained, a plant cool down should be initiated. Failure to maintain this limit and **NOT** cool down could result in hangers and/or shock suppressers exceeding their maximum travel range. [GE SIL 251, 430 and 517]

[9] **IF** Recirc Pump was tripped due to dual seal failure, **THEN** (Otherwise N/A)

[9.1] **VERIFY TRIPPED**, RECIRC DRIVE 1A(1B) NORMAL FEEDER, 1-HS-57-17(14). ☐

[9.2] **VERIFY TRIPPED**, RECIRC DRIVE 1A(1B) ALTERNATE FEEDER, 1-HS-57-15(12). ☐

[9.3] **CLOSE** tripped recirc pump suction valve using, RECIRC PUMP 1A(1B) SUCTION VALVE, 1-HS-68-1(77). ☐

[9.4] **IF** it is evident that 75°F between the dome **AND** the idle Recirc loop cannot be maintained, **THEN**

COMMENCE plant shut down and cool down in accordance with 1-GOI-100-12A. ☐

[10] **NOTIFY** Reactor Engineer to perform Reactor Recirculation System Single Loop Operation, 1-SR-3.4.1(SLO) AND to refer to Station Reactor Engineer, 0-TI-248 and Tech Specs 3.4.1 as necessary. ☐

[11] [NER/C] **WHEN** the Recirc Pump discharge valve has been closed for at least five minutes (to prevent reverse rotation of the pump) [GE SIL-517], **THEN** (N/A if Recirc Pump was isolated in Step 4.2[9])

OPEN Recirc Pump discharge valve as necessary to maintain Recirc Loop in thermal equilibrium. ☐

4.2 Subsequent Actions (continued)

[12] **REFER TO** the following ICS screens to help determine the cause of recirc pump trip/core flow decrease.

- VFDPMPA(VFDPMPB) ☐
- VFDAAL(VFDBAL) ☐

[13] **CHECK** parameters associated with Recirc Drive and Recirc Pump/Motor 1A(1B) on ICS and RECIRC PMP MTR 1A & 1B WINDING & BRG TEMPS, 1-TR-68-71 to determine cause of trip. ☐

[14] **PERFORM** visual inspection of tripped Reactor Recirc Drive. ☐

[15] **PERFORM** visual inspection of Reactor Recirc Pump Drive relay boards for relay targets. ☐

[16] **IF** necessary, **THEN** (Otherwise N/A)

REFER TO 1-OI-68 for Reactor Recirc Pump trips. ☐

[17] **INITIATE** actions required to make the necessary repairs. ☐

NOTE
Restarting a Recirc Pump while in Region 1 is **NOT** allowed. Tech Spec 3.4.1.A requires that the Reactor Mode Switch be immediately placed in SHUTDOWN upon entry into Region 1

[18] **PERFORM** the following for Single Loop Operation:

[18.1] **REFER TO** 1-OI-68 for guidance on single loop operation. ☐

[18.2] **REFER TO** Tech Specs 3.4.1. ☐

[18.3] **WHEN** available, **THEN**

RETURN tripped Recirc Pump to service in accordance with 1-OI-68. ☐

41. RO 295003AA2.01 001/MEM/T1G1/0-GOI-100-4//RO 295003AA2.01//RO/SRO/12/18/2007

Given the following plant conditions:

- All three units were at 100% rated power when 500KV PCB 5234 (Trinity 1 feed to Bus 1 Section 1) tripped and failed to auto close.
- The signal which caused the PCB trip cannot be reset.
- The Chattanooga Load Coordinator has issued a Switching Order directing BFN to open Motor Operated Disconnect (MOD) 5233 and 5235 to isolate 500KV PCB 5234 for troubleshooting.

Which ONE of the following describes your response to this Switching Order and the basis for that response?

A. Ensure the PK block for PCB 5234 is installed to facilitate testing PCB 5234 by TPS personnel assigned to troubleshoot the breaker trip.

B. Ensure the PK block for PCB 5234 is installed to prevent actuating the breaker failure logic and tripping the remainder of the PCBs on Bus 1.

C✓ Remove the PK block from PCB 5234 to prevent actuating the breaker failure logic and tripping the remainder of the PCBs on Bus 1.

D. Remove the PK block from PCB 5234 to prevent electrical arching across the MOD contacts while being opened.

K/A Statement:
295003 Partial or Complete Loss of AC / 6
AA2.01 - Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER : Cause of partial or complete loss of A.C. power

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions to determine the potential cause of a partial or complete loss of AC power.

References: 0-GOI-300-4

Level of Knowledge Justification: This question is rated as MEM due to the requirement to recall or recognize discrete bits of information.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. The function of the PK block and its relationship to the breaker failure logic.
2. Recognize the CAUTION in 0-GOI-300-4 related to PK block removal.

A is incorrect. This is plausible since the PK block must be installed to troubleshoot the breaker, however it is not re-installed until AFTER the MODs are opened.

B is incorrect. This is plausible since the wording is ALMOST identical to the CAUTION, however the PK block must be removed.

C is correct.

D is incorrect. This is plausible because electrical arching across the MOD contacts is actually what causes the trip signal to be generated if the PK block is installed. The electrical arching will occur with or without the PK block installed. However, it will not trip the 500KV breakers on the bus when it happens without the PK block installed.

8.2 Response to a Breaker Trip on 161kV or 500kV Breaker

CAUTION

Breaker reclosure times on opposite ends of the transmission lines leaving BFN are 15 to 17 seconds after a trip. The breakers at BFN should reclose immediately thereafter.

[1] **IF** a line trips, **THEN**

WAIT 30 seconds before resetting the disagreement to ensure adequate time for automatic reclosure. ☐

NOTE

1. 161kV breakers have high speed and standard speed reclosure.
2. For PCBs equipped with digital relays, the PCB will lockout from AUTO closure if the affected line does not reclose from the other end within approximately 1.5 seconds. Only the AUTO closure is prevented, the breaker can be manually closed with Dispatcher concurrence.

CAUTION

Induced currents in the current transformers of a 500KV PCB during cycling of the associated MOD's, in conjunction with an existing PCB trip signal, may actuate the breaker failure logic and trip all PCB's on the associated 500KV bus. Thus the MOD's associated with a tripped PCB should **NOT** be operated until the trip has been reset; or, if the trip cannot be reset, the breaker failure PK block has been removed for the associated tripped PCB during MOD operation. Contact Dispatcher for instruction or assistance to reset the tripped relay.

42. RO 295004AK1.03 001/C/A/T1G1/24VDC/VB9/295004AK1.03//RO/SRO/

Given the following plant conditions:

- A reactor startup is in progress and reactor power is on IRM Range 7.
- The operator observes the following annunciators/indications:
  - SRM Channels A and C fail downscale
  - IRM Channels A, C, E, and G HI-Hi INOP

Which ONE of the following power sources, if lost, would cause these failures?

A.✔ +/-24V DC Power Distribution Panel

B. 48V DC Power Distribution Panel

C. 120V AC Instrument and Control Power Distribution Panel

D. 120V AC RPS Power Supply Distribution Panel

K/A Statement:
295004 Partial or Total Loss of DC Pwr / 6
AK1.03 - Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER : Electrical bus divisional separation

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions to determine the effect on a division of IRM instruments due to a loss of DC power.

References:

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. Which of th listed power supplies input to the IRM system.
2. Which power supply, if lost, would provide only those indications listed.

A is correct.

B is incorrect. This is plausible because 48V DC supplies power to annunciator panels in the control room including the two annunciators listed in the stem. However, other annunciators would also be affected by the loss that are not included on the list.

C is incorrect. This is plausible because 120V AC I&C Buses supply power to IRM detectors and drives. A loss of that power supply would also affect the entire division. However, the indications given in the stm are not indicative of loads supplied by I&C buses.

D is incorrect. This is plausible because RPS supplies trip units associated with IRMs. However, the given annunciators are not indicative of the loads supplied by RPS.

IRM CHAN B.D.F.H HI-HI/INOP	34

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Sensor/Trip Point:

Relay K16

A. Hi-Hi
   1. 116.4 on 125 scale.
B. INOP
   1. Hi voltage low.
   2. Module unplugged.
   3. Function switch **NOT** in operate.
   4. Loss of ± 24 VDC to monitor.

Sensor Location: Control Room Panel 1-9-12.

Probable Cause:

A. Flux level at or above setpoint.
B. One or more inoperable conditions exist.
C. Testing in progress.
D. Malfunction of sensor.
E. Control rod drop accident.

Automatic Action:

A. Half-scram if one sensor actuates (except with Rx Mode Switch in RUN).
B. Reactor scram if one sensor per channel actuates (except with Rx Mode Switch in RUN).

Operator Action:

A. **STOP** any reactivity changes. ☐
B. **VERIFY** alarm by multiple indications. ☐
C. **RANGE** initiating channel or **BYPASS** initiating channel. **REFER TO** 1-OI-92A. ☐
D. With SRO permission, **RESET** Half Scram. **REFER TO** 1-OI-99
E. **IF** alarm is from a control rod drop, **THEN** **REFER TO** 1-AOI-85-1. ☐
F. [NRC/C] **IF** one or more IRM recorder reading is downscale, **THEN** **CHECK** for loss of ± 24 VDC power. ☐
G. **NOTIFY** Instrument Maintenance that functional tests of any monitors indicating an INOP condition, including a downscale reading, are required before the instrument can be considered operable. [NRC IE item 86-40-03] ☐
H. **NOTIFY** Reactor Engineer. ☐
I. **REFER TO** Tech Spec Table 3.3.1.1-1, TRM Tables 3.3.4-1 and 3.3.5-1. ☐

References:

1-45E620-6	1-730E237-6, -10	1-730E915-10
1-730E915RF-12	1-SIMI-92B	

SRM DOWNSCALE 6

(Page 1 of 1)

Sensor/Trip Point:

Relay K-19 Count rate 5 cps.

Sensor Location: Panel 1-9-12, MCR.

Probable Cause:
A. An un-bypassed SRM channel having a count rate ≤ 3 counts per second.
B. SI (or SR) in progress.
C. Malfunction of sensor.

Automatic Action: Rod block below range 3 on IRM and Rx Mode Sw. **NOT** in Run.

Operator Action:
A. **VALIDATE** SRM downscale. ☐
B. **IF** alarm valid, **THEN** **REFER TO** 1-OI-92 during startup (Mode 2) operation or 0-GOI-100-3A, -3C during refuel (Mode 5) operation. ☐
C. **NOTIFY** Unit Supervisor. ☐
D. **REFER TO** Tech. Spec. Sect. 3.3.1.2, Table 3.3.1.2-1, TRM Tables 3.3.4-1 and 3.3.5-1. ☐

References: 1-45E620-6-1 1-730E237-8

2.0 SYMPTOMS (continued)

H. Loss of Main Steam Relief Valve position indication.

I. Loss of power to RCIC and HPCI Turbine Vibration circuitry and position indication for testable check valves, (Panel 9-3).

J. Loss of RHRSW and EECW Division I instrumentation, (Panel 9-3, 9-20).

K. Loss of SBGT A flow and differential pressure indication, (Panel 9-25).

L. Loss of SLC A and B amber ready lights and valve position indication, (Panel 9-5).

M. Loss of LPRM meter lights and APRM alarm lights, (Panel 9-5).

N. Loss of Condensate - Feedwater and Heater Drains instrumentation, (Panel 9-6).

O. Loss of SRM/IRM detector drive power and position indication, (Panel 9-5).

P. Loss of one-half the blue scram lights and accumulator low pressure-high level light indications (Panel 9-5, 25-04).

Q. Loss of Control Bay Emergency Ventilation System Division I.

R. Loss of AC Supply to ± 24V NEUTRON MONITORING BATT CHGR A1-1 NEG SIDE, 1-CHGD-283-0000A1-1 and ± 24V NEUTRON MONITORING BATT CHGR A2-1 POS SIDE, 1-CHGD-283-0000A2-1. (The Neutron Monitoring Battery System is rated to carry loads for 3 hours. STACK GAS CH1 RAD MON RTMR, 0-RM-090-0147B will be lost after this time period)

S. Loss of Main Steam Line B, D and Feedwater Line B flow indicators and inputs to 3 Element Control and Rod Worth Minimizer.

T. "B" Fuel Pool Demin valves:

1. 1-FCV-078-0063, FPC F/D OUTBD ISOL VLV Closes,
2. 1-FCV-078-0068, RX WELL INFL INBD VLV Closes,
3. 1-FCV-078-0066, FPC F/D 1A BYP VLV Opens.

These actions result from loss of power to A and C skimmer surge tank low-low level switches.

U. I&C BUS A VOLTAGE ABNORMAL (1-XA-55-8C, Window 21).

V. Short Cycle valves 1-FCV-002-0029A and1- FCV-002-0029B fail open due to loss of power to 1-FC-2-29.

43. RO 295005AA1.04 001/C/A/T1G1///295005AA1.04//RO/SRO/11/28/07 RMS

Given the following plant conditions:

- Unit-1 is at 100% rated power when the Desk Unit Operator notices that the number 3 MTSV position indication is reading 0%.
- The number 1, 2, and 4 MTSV position indications all read 100%.
- Maintenance investigation determines that the cause of the MTSV position indication failure is due to a mechanical failure of the LVDT.
- The Unit-1 Main Turbine receives a trip signal

Which ONE of the following describes the effect on Main Turbine operation and any required action?

A. Main Turbine operation is unaffected. The RPS logic contact is already open for the #3 MTSV so a turbine trip will still initiate a scram.

B. Main Turbine operation is affected. The RPS logic contact for the #3 MTSV will not function so a turbine trip may not initiate a scram.

C. Main Turbine operation is unaffected. The Generator output breaker will still open on a turbine trip due to a 2-out-of-4 logic arragement.

D✓ Main Turbine operation is affected. The Generator output breaker will not open on a turbine trip due to a 4-out-of-4 logic arragement.

K/A Statement:
295005 Main Turbine Generator Trip / 3
AA1.04 - Ability to operate and/or monitor the following as they apply to MAIN TURBINE GENERATOR TRIP : Main generator controls

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions to determine the required action following a Main Turbine Generator trip.

References: OPL171.228

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. Whether the LVDT position indicator feeds the RPS logic, the Turbine Trip logic, or both.
2. Based on the above answer, the affect of a turbine trip with the failure active.

A is incorrect. This is plausible because the position of #3 MTSV is supplied to RPS logic. However, the position indication supplied to RPS is a limit switch, not the LVDT position. Therefore, RPS logic "sees" #3 MTSV as open until the turbine trips.

B is incorrect. This is plausible because the Main Turbine operation is affected. However, the position indication supplied to RPS is a limit switch, not the LVDT position. Therefore, a turbine trip WILL initiate a scram signal to RPS.

C is incorrect. This is plausible based on the different logic associated with the CIVs and RPS on the main turbine. However, the logic for MTSV inputs to open the generator output breaker is a 4-out-of-4 logic. Therefore, the generator outpu breaker will not automatically open.

D is correct.

INSTRUCTOR NOTES

c. Consequences of Event

This caused the Bypass valves to start opening. Due to the short duration of the error signal the bypass valves did not reach full open and subsequently closed.

Operation of the bypass valves would impact Rx pressure, Rx power and Generator load.

Corrective Action - EHC logic software was modified to eliminate the possibility of this type response to a communications glitch.

2. At BFN on 1/15/2006, the Unit 3 generator breaker failed to trip as expected on a turbine trip.

PER 95370

a. Description of Event

At BFN on 1/15/2006, the Unit 3 generator breaker failed to trip as expected on a turbine trip. The logic for the generator breaker needs to see all the stop valves closed and the CIV's closed (either intercept or stop). The LVDT for S.V#1 was failed such that the generator breaker would not open on a turbine trip. Operator action was taken to manually trip the generator breaker

The metal rod moves to alter the magnetic coupling of 2 opposing transformer secondary windings to make an LVDT provide an output proportional to the position of the metal rod.

INSTRUCTOR NOTES

b. Cause of Event

The LVDT transformer coupling rod became disconnected from the valve and fell to a position which gave indication of ~ 50% valve position. The affect on the logic for tripping the generator PCB on a turbine trip was not recognized.

Work practices
Monitor all parameters during a transient and ensure automatic actions have occurred

c. Consequences of Event

Tripping of the generator breaker on a turbine trip prevents a reverse power situation where the generator and turbine could attempt to rotate backwards, causing equipment damage. The unit operator's quick recognition and response to the breaker failure to trip prevented damage.

44. RO 295006AK3.05 001/C/A/T1G1/RPS/1/295006AK3.05//RO/SRO/

Given the following plant conditions:

- Power ascension is in progress on Unit 3 with the main turbine on line.
- Control rods are being withdrawn to increase power.
- As reactor power approaches 35%, the STA notes that 2 turbine bypass valves are open.

Which ONE of the following describes the effect on the plant?

Regarding the FSAR Chapter 14 analyses for a turbine trip, the above condition ________.

A. is more conservative than the assumptions used in the FSAR because it lowers the actual power level at which the RPS reactor scram on turbine trip is enabled.

B.✔ is less conservative than the assumptions used in the FSAR because it raises the actual power level at which the RPS reactor scram on turbine trip is enabled.

C. is less conservative than the assumptions used in the FSAR because it raises the actual power level for a design basis transient in regard to peak cladding temperature.

D. is more conservative than the assumptions used in the FSAR because it lowers the peak vessel pressure for a design basis transient in regard to transition boiling.

K/A Statement:
295006 SCRAM / 1
AK3.05 - Knowledge of the reasons for the following responses as they apply to SCRAM : Direct turbine generator trip: Plant-Specific

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions to determine the response to a Main Turbine trip and the basis for that response related to a reactor scram.

References:

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. Which assumptions are of concern regarding the FSAR analysis for a turbine trip.
2. What affect the above conditions have on that analysis.
3. Which thermal limit is of concern regarding the analyzed transient.

A is incorrect. This is plausible because the initial power level prior to a scram is assumed in the analysis. However, the given conditions raise the initial power level which is less conservative.

B is correct.

C is incorrect. This is plausible because the condition is less conservative based on initil power. However, the limit of concern is not PCT, but MCPR.

D is incorrect. This is plausible because RPV pressure affects transition boiling. However, this is not the limit of concern during this analysis and the initial conditions are LESS conservative with regard to MCPR.

45. RO 295016AA2.04 001/C/A/T1G1/AOI-100-2//295016AA2.04//RO/SRO/12/17/2007 RMS

Given the following plant conditions:

- Unit-3 control room was abandoned due to a fire.
- Control has been established at Panel 25-32 and actions are being carried out in accordance with 3-AOI-100-2, Control Room Abandonment.
- RCIC is injecting with RPV level at +20 inches and steady.
- A cooldown has begun using MSRVs. Pressure is 850 psig and lowering.
- RHR Loop I is in Suppression Pool Cooling.

In accordance with 3-AOI-100-2, Control Room Abandonment, a Suppression Pool Temperature limit of _____ ^{O}F has been established. The basis for this limit is _______?

A. $\leq$ 95^{O}F, to prevent exceeding the Technical Specification LCO before reaching Mode 4 (Cold Shutdown).

B. $\leq$ 110^{O}F, to prevent exceeding the Heat Capacity Temperature Limit before the reactor can be verified to be shutdown.

C. $\leq$ 120^{O}F, to prevent damage to the RCIC turbine from over-heated lube oil which is cooled by the Suppression Pool water.

D✓ $\leq$ 120^{O}F, to prevent exceeding the design basis maximum allowable values for primary containment temperature or pressure.

K/A Statement:

295016 Control Room Abandonment
AA2.04 - Ability to determine and/or interpret the following as they apply to CONTROL ROOM ABANDONMENT : Suppression pool temperature

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions to determine the limitation and basis for Suppression Pool Temperature during a Control Room Abandonment.

References: 3-AOI-100-2, Tech Spec Bases 3.6, EOIPM 0-V-B

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. The SP Temperature limit established by 3-AOI-100-2.
2. The basis for the limit.

A is incorrect. This is plausible because $\leq 95^{O}F$ is the normal operating limit imposed by Technical Specification. However, this limit is not expected to be maintained during Control Room Abandonment.

B is incorrect. This is plausible because the basis for $\leq 110^{O}F$ is correct. However, this limit is not expected to be maintained during Control Room Abandonment and the reactor is assumed to be shutdown.

C is incorrect. This is plausible because elevated SP temperatures can result in over-heating the lube oil used for lubricating RCIC. This constitues the basis for Caution #6 in the EOIs, but does not apply during Control Room Abandonment.

D is correct.

Date ________

4.2 Unit 3 Subsequent Actions (continued)

[15.7] **ESTABLISH** RHR system flow between 7,000 and 10,000 gpm as follows: ☐

[15.7.1] **MONITOR** RHR SYS I TOTAL FLOW, 3-FI-74-79 at Panel 3-25-32. ☐

[15.7.2] **THROTTLE OPEN** 3-HS-074-0059C, RHR SYSTEM I TEST VLV at 480V RMOV Bd 3A, Compt. 12C, ☐

[15.7.3] **WHEN** RHR SYS I TOTAL FLOW, 3-FI-74-79 indicates between 7,000 and 10,000 gpm, **THEN**

DIRECT the operator to stop throttling 3-HS-074-0059C. ☐

[15.7.4] **VERIFY CLOSED** RHR SYSTEM I MINIMUM FLOW VALVE, 3-FCV-74-7, at either of the following: ☐

- 480V RMOV Bd 3D, Compt. 4E, 3-BKR-074-0007 RHR SYSTEM I MINIMUM FLOW VLV FCV-74-7 (MO10-16A), OR (Otherwise N/A) ☐
- Rx Bldg - SW Quad - E1 541' local control switch RHR SYSTEM I MINIMUM FLOW VALVE, 3-HS-074-0007B. (Otherwise N/A) ☐

[15.8] **MONITOR** SUPPR POOL TEMPERATURE, 3-TI-64-55B, at Panel 3-25-32 **and MAINTAIN** temperature less than 120°F , ☐

BASES

ACTIONS

D.1, D.2, and D.3 (continued)

Additionally, when suppression pool temperature is > 110°F, increased monitoring of pool temperature is required to ensure that it remains ≤ 120°F. The once per 30 minute Completion Time is adequate, based on operating experience. Given the high suppression pool average temperature in this Condition, the monitoring Frequency is increased to twice that of Condition A. Furthermore, the 30 minute Completion Time is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal suppression pool average temperature condition.

E.1 and E.2

If suppression pool average temperature cannot be maintained at ≤ 120°F, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the reactor pressure must be reduced to < 200 psig within 12 hours, and the plant must be brought to at least MODE 4 within 36 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

Continued addition of heat to the suppression pool with suppression pool temperature > 120°F could result in exceeding the design basis maximum allowable values for primary containment temperature or pressure. Furthermore, if a blowdown were to occur when the temperature was > 120°F, the maximum allowable bulk and local temperatures could be exceeded very quickly.

(continued)

BASES

LCO (continued)

b. Average temperature ≤ 105°F when any OPERABLE IRM channel is > 70/125 divisions of full scale on Range 7 and testing that adds heat to the suppression pool is being performed. This required value ensures that the unit has testing flexibility, and was selected to provide margin below the 110°F limit at which reactor shutdown is required. When testing ends, temperature must be restored to ≤ 95°F within 24 hours according to Required Action A.2. Therefore, the time period that the temperature is > 95°F is short enough not to cause a significant increase in unit risk.

c. Average temperature ≤ 110°F when all OPERABLE IRM channels are ≤ 70/125 divisions of full scale on Range 7. This requirement ensures that the unit will be shut down at > 110°F. The pool is designed to absorb decay heat and sensible heat but could be heated beyond design limits by the steam generated if the reactor is not shut down.

Note that 70/125 divisions of full scale on IRM Range 7 is a convenient measure of when the reactor is producing power essentially equivalent to 1% RTP. At this power level, heat input is approximately equal to normal system heat losses.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause significant heatup of the suppression pool. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining suppression pool average temperature within limits is not required in MODE 4 or 5.

(continued)

DISCUSSION: CAUTION #5 and CAUTION #6

CAUTION #5, this warns the operator of the potential plant response if injection of cold, unborated water into the core is too rapid under conditions where little or no margin to subcriticality may exist. This may result in a large increase in positive reactivity with a subsequent reactor power excursion large enough to substantially damage the core.

CAUTION #6, the HPCI and RCIC Lube Oil Coolers are cooled by routing part of the pump discharge fluid to the cooler. At elevated temperatures in the suppression pool, the turbine lube oil may get too hot to provide adequate lubrication. Only during EOI operations will the system be needed at such an extreme suppression pool temperature. Therefore, the EOIs are an appropriate location for this caution.

46. RO 295018AK2.01 001/MEM/T1G1/RBCCW/3/295018AK2.01///

Which ONE of the following components would lose cooling upon isolation of the RBCCW non-essential loop isolation valve (2-FCV-70-48)?

A. Drywell atmospheric coolers

B.✔ Fuel pool cooling heat exchanger

C. Recirculation pump seals

D. Drywell Equipment Drain Sump Heat Exchanger

K/A Statement:
295018 Partial or Total Loss of CCW / 8
AK2.01 - Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER and the following: System loads

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific system knowledge to determine the effect on RBCCW loads due to a partial loss of RBCCW.

References: 1/2/3-AOI-70-1, OPL171.047

Level of Knowledge Justification: This question is rated as MEM due to the requirement to recall or recognize discrete bits of information.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. Which of the loads listed are part of the "Essential Loop".
2. Which of the loads listed are part of the "Non-essential Loop".

A is incorrect. This is an "Essential Loop" load.

B is correct.

C is incorrect. This is an "Essential Loop" load.

D is incorrect. This is an "Essential Loop" load.

Attachment 1
(Page 1 of 1)

Components Cooled by RBCCW During Normal Plant Operation

SYSTEM	COMPONENTS COOLED
Reactor Recirculation	Pump Seals Pump Motor Bearings Pump Motor Windings Pump Discharge Sample Cooler
Primary Containment	Drywell Atmosphere Cooling Coils
Reactor Water Cleanup	Non-Regenerative Heat Exchangers Pump Seals Pump Bearings
Fuel Pool Cooling and Cleanup	Fuel Pool Heat Exchangers
Equipment Drains	Reactor Building Equipment Drain Sump Heat Exchanger Drywell Equipment Drain Sump Heat Exchanger

d. Proper system flow operation is assured by monitoring the system DP (pump discharge minus pump suction). — Done Each Shift

2. RBCCW Heat Loads

a. Essential loop loads — Obj. V.B.2

- Drywell Blowers(10) — Obj. V.D.2
- Reactor recirculation pump motor coolers (2)
- Reactor recirculation pump seal coolers (2)
- Drywell equipment drain sump heat exchanger (1)

b. Non-essential loop loads — Obj. V.B.3

- Reactor Building equipment drain sump heat exchanger (1) — Obj. V.D.3
- Reactor water cleanup pump seal water coolers and bearing oil coolers (2)
- RWCU Non-regenerative heat exchangers (2)
- Fuel pool cooling heat exchangers (2)
- Reactor recirculation pump discharge sample cooler (1)

3. RBCCW Heat Exchangers

a. These provide the means for heat removal from RBCCW by RCW with Emergency Equipment Cooling Water (EECW) as a backup. — DCN 51195, replaced HX1A & 1B, HX 1C NOT replaced. OPL171.051

b. They are counter-flow type, 50% capacity each.

- RBCCW flow makes one pass through the shell side.
- RCW makes one pass through the tube side.

47. RO 295019AA2.02 001/C/A/SYS/CA//295019AA2.02///MODIFIED 11/17/07

Given the following plant conditions:

- Unit 2 was at 100% power when a transient occurred which resulted in a reactor scram.
- The unit is stabilized, and the scram signal is reset.
- All 8 scram solenoid group lights are on.
- Ten minutes later, the following conditions are present:

  - RCW pressure low alarm
  - CRD charging water pressure high alarm
  - Outboard MSIVs closed, Inboard MSIVs open
  - SDV vents and drain valves closed
  - Scram solenoid air valves open

Which ONE of the following describes the cause for the event?

A.✓ Loss of Control Air.

B. Loss of both RPS busses.

C. Loss of 9-9 cabinet 5, Unit Non-Preferred.

D. Loss of Drywell Control Air.

K/A Statement:
295019 Partial or Total Loss of Inst. Air / 8
AA2.02 - Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR : Status of safety-related instrument air system loads (see AK2.1 - AK2.19)

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions to determine the effect of a loss of Control Air on safety related loads.

References: 2-AOI-32-2

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. Which indications given are indicative of the possible causes listed.

A is correct.

B is incorrect. This is plausible because the scram would occur as well as scram valves open and SDV vents and drains closed, however these indications would NOT be appropriate AFTER the scram was reset. In fact, the scram could NOT be reset without RPS available.

C is incorrect. This is plausible because the only indications given that would not apply would be outboard MSIVs closing and SDV vents and drains failure to re-open.

D is incorrect. This is plausible because a loff of Drywell Control Air would cause MSIVs to close, however the INBOARD valves would close.

1.0 PURPOSE

This Abnormal Operating Instruction provides symptoms, automatic action, operator actions and expected system responses for loss of control air.

2.0 SYMPTOMS

A. AIR COMPRESSOR ABNORMAL annunciator, (1-XA-55-20B, Window 29) is in alarm.

B. CONTROL AIR COMP G BKR ENERGIZED (0-XA-55-23B, Window 38) will reset (extinguish) when panel reset pushbutton is depressed.

C. CONTROL AIR COMP G MOTOR AMPS, 0-EI-32-2901, on Panel 1-9-20 indicates approximately zero amps.

D. Air Compressor G ICS Display shows Compressor G in an unloaded or shutdown condition.

E. Air Compressor G ICS Display shows lowering Control Air Header Pressure.

F. Control Air Compressor G breaker tripped.

G. SERVICE AIR XTIE VLV OPEN (0-FCV-33-1 Open) annunciator, 0-PA-33-1A/1(3) (Unit 1 and Unit 3) on Panel 1(3)-9-20 is in alarm at (1(3)-XA-55-20B, Window 30).

H. CONTROL AIR PRESS LOW annunciator, 0-PA-32-88 is in alarm (2-XA-55-20B, Window 32).

I. SCRAM PILOT AIR HEADER PRESS LOW annunciator, 2-PA-85-38B on Panel 9-5 is in alarm (2-XA-55-5B, Window 28).

J. MAIN STEAM LINE ISOL VLV POSN HALF SCRAM annunciator is in alarm (2-XA-55-4A, Window 30).

K. DRYWELL CONTROL AIR PRESSURE LOW 2-PA-32-70 annunciator is in alarm (2-XA-55-3E, Window 35).

L. CONDENSER A, B OR C VACUUM LOW 2-PA-47-125 annunciator is in alarm (2-XA-55-7B, Window 17).

M. OG HOLDUP LINE INLET FLOW LOW 2-FA-66-111A annunciator is in alarm (2-XA-55-53, Window 4).

N. HOTWELL A(B)(C) LEVEL ABNORMAL 2-LA-2-3(2-LA-2-6)(2-LA-2-9) is in alarm (2-XA-55-6A, Window 5(6)(7)).

2.0 SYMPTOMS (continued)

O. REACTOR WATER LEVEL ABNORMAL 2-LA-3-53 annunciator is in alarm (2-XA-55-5A, Window 8).

P. REACTOR PRESS HIGH 2-PA-3-53 annunciator is in alarm (2-XA-55-5A, Window 1).

Q. REACTOR CHANNEL A(B) AUTO SCRAM annunciator in alarm if any scram setpoint is exceeded (2-XA-55-5B, Window 1(2)).

R. MAIN STEAM RELIEF VLV AIR ACCUM PRESS LOW 2-PA-32-31 annunciator in alarm (2-XA-55-3D, Window 18).

3.0 AUTOMATIC ACTIONS

A. Unit 2 to Unit 3 Control Air Crosstie, 2-PCV-032-3901, will close when Control Air Header pressure reaches 65 psig and lowering at the valve.

B. U-1 TO U-2 CONT AIR CROSSTIE, 1-PCV-032-3901, will close to separate Units 1 and 2 when Control Air Header pressure reaches 65 psig and lowering at the valve.

C. 2-PCV-84-0654, CAD/CA FLOW SEL, will select nitrogen from CAD tank A to supply 2-FSV-64-20, 2-FSV-64-21, 2-FSV-64-221, and 2-FSV-64-222 at ≤75 psig.

D. 2-PCV-84-0033, will select nitrogen from CAD tank A to supply 2-FSV-84-19, 2-FSV-64-29, and 2-FSV-64-32.

E. 2-PCV-84-0034, will select nitrogen from CAD tank B to supply 2-FSV-84-20, 2-FSV-64-31, and 2-FSV-64-34.

4.0 OPERATOR ACTIONS

NOTE

[NER/C] Attachment 1 provides expected system responses, critical components that do not fail in intended positions should be placed in the required positions. [INPO SOER 88-001]

4.1 Immediate Actions

None

4.2 Subsequent Actions

[1] **IF** a RFP Minimum Flow Valve failed open and flow is required from the condensate/feedwater system to reactor vessel or to prevent pump overload, **THEN**

ISOLATE the associated RFP minimum flow lines in the appropriate RFPT Room as follows: (N/A any RFP valves not affected.)

- RFP 2A MIN FLOW SHUTOFF, 2-SHV-003-0508 ☐
- RFP 2B MIN FLOW SHUTOFF, 2-SHV-003-0517 ☐
- RFP 2C MIN FLOW SHUTOFF, 2-SHV-003-0526 ☐

[2] **IF** CNDS BSTR PUMPS DISCH BYPASS TO COND B, 2-FCV-2-29A

and

CNDS BSTR PUMPS DISCH BYPASS TO COND C, 2-FCV-2-29B fail CLOSED, **THEN** (Otherwise N/A)

- **VERIFY** a flow path for condensate system

  OR

- **STOP** the condensate pumps/booster pumps using 2-OI-2. ☐

[3] **IF** any outboard MSIVs fails closed, **THEN:**

PLACE associated hand-switch on Panel 2-9-3 to close position. (Otherwise N/A) ☐

4.2 Subsequent Actions (continued)

[4] **IF** RSW STRG TNK ISOLATION VALVE, 0-FCV-025-0032 FAILS CLOSED, **THEN**

START a high pressure fire pump using 0-OI-26 . ☐

[5] **OPEN** CAD SYSTEM A N2 SHUTOFF VALVE, 0-FCV-084-0005, at Panel 9-54. ☐

[6] **OPEN** CAD SYSTEM B N2 SHUTOFF VALVE, 0-FCV-084-0016, at Panel 9-55. ☐

NOTES

1) All RCW temperature control valves fail open except for 2-TCV-24-80B and 2-TCV-24-85B on 2A and 2B RBCCW heat exchangers and 2-TCV-024-0075B on the Main Turbine Oil Coolers (4" line) which fail closed.

2) The appropriate computer points may be used for monitoring for the following lube oil temperatures, or any local temperature monitoring device that may be available, as necessary

[7] **IF** RCW pump motor amps indicate that RCW System flow reduction is required, **THEN**

REDUCE RCW flows as required: (Otherwise N/A). ☐

[7.1] **CLOSE** main turbine lube oil cooler TCV isolation valve 2-SHV-024-0583 or 2-SHV-024-0584, **THEN**

ESTABLISH lube oil temperature between 80°F and 90°F using TCV BYPASS VALVE 2-BYV-024-0585 or 2-BYV-024-0586. ☐

[7.2] **CLOSE** the following RFP turbine oil cooler TCV isolation valves

- A RFP 2-24-624A or 2-24-625A ☐
- B RFP 2-24-624B or 2-24-625B ☐
- C RFP 2-24-624C or 2-24-625C ☐

1.0 PURPOSE

This abnormal operating instruction provides symptoms, automatic actions and operator actions for the loss of Drywell Control Air System for causes other than Group 6 Isolation. The loss of Drywell Control Air caused by a Group 6 Isolation is addressed in 2-AOI-64-2d.

2.0 SYMPTOMS

A. DRYWELL CONTROL AIR PRESS LOW (2-XA-55-3E, Window 35) at $\leq$ 87 psig.

B. MAIN STEAM RELIEF VLV AIR ACCUM PRESS LOW (2-XA-55-3D, Window 18) at $\leq$ 82 psig.

C. Inboard MSIV's close or start to close.

D. Drywell cooler dampers close.

3.0 AUTOMATIC ACTIONS

None

48. RO 295021G2.4.50 001/C/A/T1G1/74-1//2950212.4.50///7

Given the following plant conditions:

- Unit 2 is aligned with RHR Loop I in shutdown cooling with Loop II in standby readiness.
- A leak occurs which results in the following conditions:

  - RPV level at '0' and slowly lowering
  - DWP at 3.0 psig and slowly rising
  - RHR pumps 'A' and 'C' tripped

Which ONE of the following describes the minimum actions required to align RHR Loop II for injection to the RPV?

A. After FCV-74-47 and FCV-74-48 are closed; reset PCIS; push the RHR SYS II SD CLG INBD INJECT ISOL RESET 2-XS-74-132; and open both injection valves.

B.✔ After FCV-74-47 or FCV-74-48 is closed; push the RHR SYS II SD CLG INBD INJECT ISOL RESET 2-XS-74-132.

C. After FCV-74-47 or FCV-74-48 is closed; reset PCIS; push the RHR SYS II SD CLG INBD INJECT ISOL RESET 2-XS-74-132; and open the inboard injection valve.

D. After FCV-74-47 and FCV-74-48 are closed; start Loop 2 pumps;reset PCIS; and open the inboard injection valve.

K/A Statement:
295021 Loss of Shutdown Cooling / 4
2.4.50 - Emergency Procedures / Plan Ability to verify system alarm setpoints and operate controls identified in the alarm response manual

K/A Justification: This question satisfies the K/A statement by requiring the candidate to analyze plant conditions and determine the required actions during an emergency which have resulted in a loss of shutdown cooling.

References: 2-AOI-74-1

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. Current RHR Loop II status given the initial conditions.
2. Based on the RHR Loop II status, determine the minimum actions to align Loop II for injection to the RPV.

A is incorrect. This is plausible because the valve alignment is correct and resetting 2-XS-74-132 is correct. However, resetting PCIS and re-opening FCV 74-47 & 48 are NOT required.

B is correct.

C is incorrect. This is plausible because the valve alignment is correct and resetting 2-XS-74-132 is correct. However, resetting PCIS and re-opening FCV 74-47 is NOT required.

D is incorrect. This is plausible because the valve alignment is correct and resetting 2-XS-74-132 is correct. However, resetting PCIS, re-opening FCV 74-47 and re-starting RHR pumps are NOT required.

4.2 Subsequent Actions (continued)

[5] **IF** Shutdown Cooling isolates on low RPV water level or high Drywell press (GROUP 2 ISOL) AND RPV water level needs restoring using LPCI, **THEN** (Otherwise N/A)

PERFORM the following before reaching -122 inches RPV water level: ☐

NOTE
The LPCI inboard injection valve that is aligned per 2-POI-74-2 will already be in the required accident position with the breakers open and will **NOT** isolate.

[5.1] **PERFORM** the following on a group 2 isolation: ☐

[5.1.1] **IF** 2-POI-74-2 is in effect, **THEN**

VERIFY CLOSED one of the following valves: (Otherwise N/A) ☐

- RHR SHUTDOWN COOLING SUCT OUTBD ISOL VLV, 2-FCV-74-47. ☐
- RHR SHUTDOWN COOLING SUCT INBD ISOL VLV, 2-FCV-74-48.

AND ☐

- **VERIFY CLOSED** the LPCI inboard injection valve **NOT** aligned for 2-POI-74-2, (RHR SYS I LPCI INBD INJECT VALVE, 2-FCV-74-53 OR RHR SYS II LPCI INBD INJECT VALVE, 2-FCV-74-67) ☐

4.2 Subsequent Actions (continued)

[5.1.2] **IF** 2-POI-74-2 is **NOT** in effect, **THEN**

VERIFY CLOSED the following valves on a Group 2 isolation: ☐

- RHR SHUTDOWN COOLING SUCT OUTBD ISOL VLV, 2-FCV-74-47. ☐
- RHR SHUTDOWN COOLING SUCT INBD ISOL VLV, 2-FCV-74-48. ☐
- RHR SYS I LPCI INBD INJECT VALVE, 2-FCV-74-53. ☐
- RHR SYS II LPCI INBD INJECT VALVE, 2-FCV-74-67. ☐

[5.2] **DEPRESS** RHR SYS I(II) SD CLG INBD INJECT ISOL RESET, 2-XS-74-126 and 2-XS-74-132 AND **VERIFY** 2-IL-74-126 and 2-IL-74-132 extinguished. ☐

49. RO 295023AK1.02 001/C/A/T1G1/79-2/V.B.3.B/295023AK1.02///

Fuel loading is in progress on Unit 1 when you notice an unexplained rise in SRM count rate and an indicated reactor period; you suspect that an inadvertant criticality event is taking place.

Select which ONE of the following actions is an appropriate response to Inadvertant Criticality During Incore Fuel Movements?

A. If unexpected criticality is observed following control rod withdrawal, manually SCRAM the reactor.

B. If all rods are not inserted/cannot be inserted, verify the fuel grapple is latched onto the fuel assembly handle and immediately remove the fuel assembly from the reactor core.

C.✔ If the reactor cannot be determined to be subcritical, traverse the refueling bridge and fuel assembly away from the reactor core, preferably to the area of the cattle chute..

D. Immediately EVACUATE all personnel from the refuel floor.

K/A Statement:
295023 Refueling Acc Cooling Mode / 8
AK1.02 - Knowledge of the operational implications of the following concepts as they apply to REFUELING ACCIDENTS : Shutdown margin

K/A Justification: This question satisfies the K/A statement by requiring the candidate to analyze specific plant conditions to determine a reduction in Shutdown Margin has occurred and the actions required to address that condition.

References: 1-AOI-79-2

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. The appropriate condition and Immediate Action required by 1-AOI-79-2.

A is incorrect. This is plausible because the condition is correct, but the action to scram is incorrect. Reinserting the control rod is required.

B is incorrect. This is plausible because the required action is correct, but the condition is NOT correct. This action is based on unexplained criticality following insertion of a fuel assembly.

C is correct.

D is incorrect. This is plausible because the evacuation of the Refuel Floor MAY be directed, but other actions to mitigate the problem take precedence until personnel safety is compromised.

4.0 OPERATOR ACTIONS

4.1 Immediate Actions

[1] **IF** unexpected criticality is observed following control rod withdrawal, **THEN**

REINSERT the control rod. ☐

[2] **IF** all control rods can NOT be fully inserted, **THEN**

MANUALLY SCRAM the Reactor. ☐

[3] **IF** unexpected criticality is observed following the insertion of a fuel assembly, **THEN**

PERFORM the following:

[3.1] **VERIFY** fuel grapple latched onto the fuel assembly handle AND **IMMEDIATELY REMOVE** the fuel assembly from the Reactor core. ☐

[3.2] **IF** the Reactor can be determined to be subcritical **AND** no radiological hazard is apparent, **THEN**

PLACE the fuel assembly in a spent fuel storage pool location with the least possible number of surrounding fuel assemblies and **LEAVE** the fuel grapple latched to the fuel assembly handle. ☐

[3.3] **IF** the Reactor can NOT be determined to be subcritical **OR** adverse radiological conditions exist, **THEN**

TRAVERSE the Refueling Bridge and fuel assembly away from the Reactor core, preferably to the area of the cattle chute and **CONTINUE** at Step 4.1[4]. ☐

[4] **IF** the Reactor can NOT be determined to be subcritical **OR** adverse radiological conditions exist, **THEN**

EVACUATE the refuel floor. ☐

50. RO 295024G2.1.33 001/C/A/T1G1/CONT/PRI/B10/295024G2.1.33//RO/SRO/

During operation at 100% power a gross failure of both seals on recirculation pump "B" increases drywell pressure to 2.0 psig.

Which ONE of the following is the approximate amount and type of RCS leakage?

A.✓ 60 gpm of Unidentified leakage

B. 60 gpm of Identified leakage

C. 30 gpm of Unidentified leakage

D. 30 gpm of Identified leakage

K/A Statement:
295024 High Drywell Pressure / 5
2.1.33 - Conduct of Operations Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications

K/A Justification: This question satisfies the K/A statement by requiring the candidate to determine that entry into Technical Specifications is required based on conditions which have resulted in high drywell pressure.

References: U2 TSR Sections 1 & 3.4.4, 2-AOI-68-1, OPL171.007

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. Whether the leakage is IDENTIFIED or UNIDENTIFIED leakage.
2. The amount of leakage assicoated with a gross failure of both seals on a single recirc pump.

A is correct.

B is incorrect. This is plausible because the amount of leakage is correct. However, the leakage is not IDENTIFIED because the leakage is not intentionally captured and directed to a sump and is not expected.

C is incorrect. This is plausible because the leakage is UNIDENTIFIED and equal to the Tech Spec value for total leakage, but insufficient for the conditions given.

D is incorrect. This is plausible because the leakage is equal to the Tech Spec value for total leakage, but insufficient for the conditions given. In addition, the leakage is not IDENTIFIED because the leakage is not intentionally captured and directed to a sump and is not expected.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 RCS Operational LEAKAGE

LCO 3.4.4 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. ≤ 5 gpm unidentified LEAKAGE; and

c. ≤ 30 gpm total LEAKAGE averaged over the previous 24 hour period; and

d. ≤ 2 gpm increase in unidentified LEAKAGE within the previous 24 hour period in MODE 1.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Unidentified LEAKAGE not within limit. OR Total LEAKAGE not within limit.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Unidentified LEAKAGE increase not within limit.	B.1 Reduce LEAKAGE increase to within limits. OR	4 hours (continued)

1.1 Definitions (continued)

LEAKAGE	LEAKAGE shall be: a. Identified LEAKAGE 1. LEAKAGE into the drywell, such as that from pump seals or valve packing, that is captured and conducted to a sump or collecting tank; or 2. LEAKAGE into the drywell atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; b. Unidentified LEAKAGE All LEAKAGE into the drywell that is not identified LEAKAGE; c. Total LEAKAGE Sum of the identified and unidentified LEAKAGE; d. Pressure Boundary LEAKAGE LEAKAGE through a nonisolable fault in a Reactor Coolant System (RCS) component body, pipe wall, or vessel wall.
LINEAR HEAT GENERATION RATE (LHGR)	The LHGR shall be the heat generation rate per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.

(continued)

RECIRC PUMP A NO. 2 SEAL LEAKAGE HIGH 2-FA-68-55 18	Sensor/Trip Point: 2-FIS-068-0055 0.1-0.2 gpm after second seal.

(Page 1 of 2)

Sensor Location: Recirculation Pump 2A Drywell

Probable Cause:
A. Recirculation Pump 2A No. 2 (outer) seal failure.
B. Sensor malfunction.

Automatic Action: None

Operator Action:

A. **COMPARE** No. 2 cavity pressure indicator (2-PI-68-63A) to No. 1 cavity pressure indicator (2-PI-68-64A), on Panel 2-9-4 or ICS. No. 2 seal degradation is indicated if the pressure at No. 2 seal is less than 50% of the pressure at No. 1 seal. ☐

B. **IF** seal failure is indicated, **THEN**
INITIATE seal replacement as soon as possible. Continued operation is permissible if Drywell leakage is within T.S. limits. ☐

NOTE

1) Possible indications of dual seal failure include:
- Window 25 on this panel alarming in conjunction with this window.
- Rising drywell pressure and/or temperature.
- Increased leakage into the drywell sump.
- Increased vibration of the recirc pump.

C. **IF** dual seal failure is indicated, **THEN** ☐
1. **SHUTDOWN** Recirc Pump 2A by DEPRESSING RECIRC DRIVE 2A SHUTDOWN, 2-HS-96-19.. ☐
2. **VERIFY TRIPPED**, RECIRC DRIVE 2A NORMAL FEEDER, 2-HS-57-17. ☐
3. **VERIFY TRIPPED**, RECIRC DRIVE 2A ALTERNATE FEEDER, 2-HS-57-15. ☐
4. **CLOSE** RECIRC PUMP 2A SUCTION VALVE, 2-HS-68-1. ☐

Continued on Next Page

(b) The flow keeps number 1 seal cavity clean and cool by flowing out of the seal area, along the pump shaft, and into the recirculation system.

(c) This purge flow reduces the possibility of seal damage due to foreign material entering the seal from an unclean piping system.

(8) Seal Failures — Obj. V.B.8; Obj. V.C.3; TP-6

(a) Seal failure may be assessed by the resulting changes in flows and pressures.

(b) Failure of the number 1 seal assembly would allow a higher flow to the number 2 seal cavity, forcing the number 2 seal to operate at a higher pressure (i.e., greater than 500 psig). — ARPs provide useful info/analysis. Obj. V.D.2c; Obj. V.E.3c

(c) This failure of the number 1 seal will cause leakage through the controlled seal leak-off line to rise to approximately 1.1 gpm. A flow element in this line causes a common alarm on high flow at 0.9 gpm or on low flow at 0.5 gpm.

(d) Failure of the number 2 seal assembly would cause its seal pressure to drop (depending upon the magnitude of the failure).

(i) This failure would also cause a higher leakage through the seal leak detection line downstream from the number 2 seal.

(ii) Normally there is no flow through this line and flow switches are set to alarm at 0.1-0.2 gpm flow.

(e) Failure of both mechanical seals would result in a total seal assembly leakage of 60 gpm as limited by the seal breakdown bushings.

Would cause elevated drywell temp and pressure, and would exceed Tech Spec and EPIP limits for RCS leakage.

(f) Should the number 1 seal restricting orifice become plugged, the RECIRC PUMP A(B) NO. 1 SEAL LEAKAGE ABN annunciator will alarm on low flow (less than or equal to 0.5 gpm). Additionally, a reduction in number 2 seal pressure would be seen.

(g) Should the number 2 restricting orifice become plugged, the RECIRC PUMP A(B) NO. 1 SEAL LEAKAGE ABN annunciator would also alarm on low flow; however, number 2 seal pressure would rise to near the pressure of number 1 seal.

(9) Seal Cooling — Obj. V.B.9

(a) Cooling for the recirculation pump seals is required due to the heat generated by the friction of the sealing surfaces and the leakage of reactor water through the seal assembly. — Obj. V.C.3, Obj. V.B.19c

(b) This cooling is provided by a combination of supplied Reactor Building Closed Cooling Water (RBCCW) and the leakage of primary coolant past the seals.

51. RO 295025EK2.08 001/C/A/T1G1/EHC LOGIC//295025EK2.08//RO/SRO/

Unit 2 has experienced an inadvertant MSIV closure and subsequent reactor scram. Consequently, RCIC was placed in level control and is also maintaining reactor pressure 900 to 1000 psig with the MSIVs still isolated.

Given these plant conditions, the digital EHC system is in __________ pressure control with the pressure setpoint set at _____ psig.

A. Reactor, 970

B. Reactor, 700

C. Header, 970

D.✔ Header, 700

K/A Statement:
295025 High Reactor Pressure / 3
EK2.08 - Knowledge of the interrelations between HIGH REACTOR PRESSURE and the following: Reactor/turbine pressure regulating system: Plant-Specific

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions to determine the response of the digital EHC system to a transient resulting in a high reactor pressure.

References: OPL171.228

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. Whether the header pressure dropped sufficiently low enough to cause an automatic transfer to Header Pressure Control.
2. Whether current plant conditions have allowed EHC logic to automatically transfer back to Reactor Pressure Control.

A is incorrect. This is plausible because this condition is typical for post-scram EHC conditions if the MSIVs are open.

B is incorrect. This is plausible because EHC automatically transfers back to Header pressure control if HEADER pressure returns above 725 psig. However, the MSIVs are still closed so the given readctor pressure is not being sensed by the header pressure instruments.

C is incorrect. This is plausible because EHC will swap to Header pressure control, but the setpoint will drop from 970 psig to 700 psig.

D is correct.

INSTRUCTOR NOTES

2. During turbine start-up and for a brief time following synchronization, the bypass valve control also maintains the reactor steam pressure. Once all the bypass valves are closed, then the turbine control maintains reactor steam pressure either in Header Pressure or Reactor Pressure Control depending on which operating mode is selected.

   Monitor Plant parameters for expected response

3. Steam pressure control is selectable from either panel 9-7 or the EHC Workstation by selecting HEADER PRESSURE CONTROL or REACTOR PRESSURE CONTROL.

   Obj.V.B.9.a

4. Header Pressure Control Input Signal

   a. Two redundant pressure transmitters sense header pressure at the main steam throttle just upstream of the main turbine stop valves.

      Powered from within the EHC system

   b. Both signals are monitored for low, high, difference, and hardware failures.

   c. The higher of the two signals when no failures are detected is selected as the input.

   d. A maximum difference setpoint of 10-PSI is also established to detect a fault and/or transmitter drift from either of the inputs.

   e. In the event a fault is detected, the channel is prohibited from being used in the signal processing and the appropriate BYPASS pushbutton light will illuminate on 9-7 and on the HMI operator interface.

      Obj.V.B.9.c

   f. Once the failed signal is corrected, depressing the BYPASS pushbutton will reset the BYPASS logic and both input signals will then be processed.

INSTRUCTOR NOTES

g. This mode IS NOT single failure proof - one of the two pressure sensors failing upscale can, and generally will be selected by the logic to control. This will open the TCV's and BPV's to depressurize the header to the MSIV isolation setpoint of 852 psig in RUN Mode.

h. In the unlikely event that both inputs signals are detected as failed, the control logic will automatically switch to reactor pressure control.

i. If header pressure drops below 700-PSI, and reactor pressure control is the controlling mode of operation, the control logic will automatically transfer to header pressure control. If desired, the operator may re-select reactor pressure control after the transfer has been made even though header pressure is below 700-psi. The automatic transfer logic will re-engage if header pressure rises above 725-psi.

5. Reactor Pressure Control Input Signal — TP-3

a. Four (4) redundant pressure transmitters (PT- 204a-d) grouped in pairs with "A" and "B" constituting one pair and "C" and "D" the other pair.

b. A pressure-biasing algorithm determines the lagged high-median value of the four (4) inputs and biases the remaining three (3) input signals to that high median value.

c. The high-median signal is then averaged with the other three signals and is used as "Actual Rx Pressure". — Four biased signals are averaged.

52. RO 295026EA2.01 001/C/A/T1/G1///295026EA2.01//RO/SRO/RWM

Given the following plant conditions:

- Unit-2 is in a transient condition with current conditions as follows.

  - Suppression pool level: 13.5 feet
  - Reactor pressure: 900 psig
  - Suppression pool temperature: 105°F

Which ONE of the following describes the required action?

REFERENCE PROVIDED

A.✓ Operate all available suppression pool cooling.

B. Emergency Depressurize the RPV by opening all six ADS valves.

C. Rapidly depressurize via the Main Turbine bypass valves.

D. Lower Reactor Pressure to stay within the Safe Area of the Heat Capacity Temperature Limit Curve and maintain cooldown rate below 100 deg. F/hr.

<u>K/A Statement:</u>

<u>295026 Suppression Pool High Water Temp. / 5</u>

EA2.01 - Ability to determine and/or interpret the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: Suppression pool water temperature

<u>K/A Justification:</u> This question satisfies the K/A statement by requiring the candidate to correctly identify an adverse condition related to Suppression Pool High Temperature and then determine the action required to correct the adverse condition.

<u>Reference:</u> 2-EOI-2 Flowchart, EOIPM Section 0-V-D Page 85

<u>Level of Knowledge Justification:</u> This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to solve a problem. This requires mentally using this knowledge and its meaning to resolve the problem.

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REFERENCE PROVIDED- HCTL Curve only

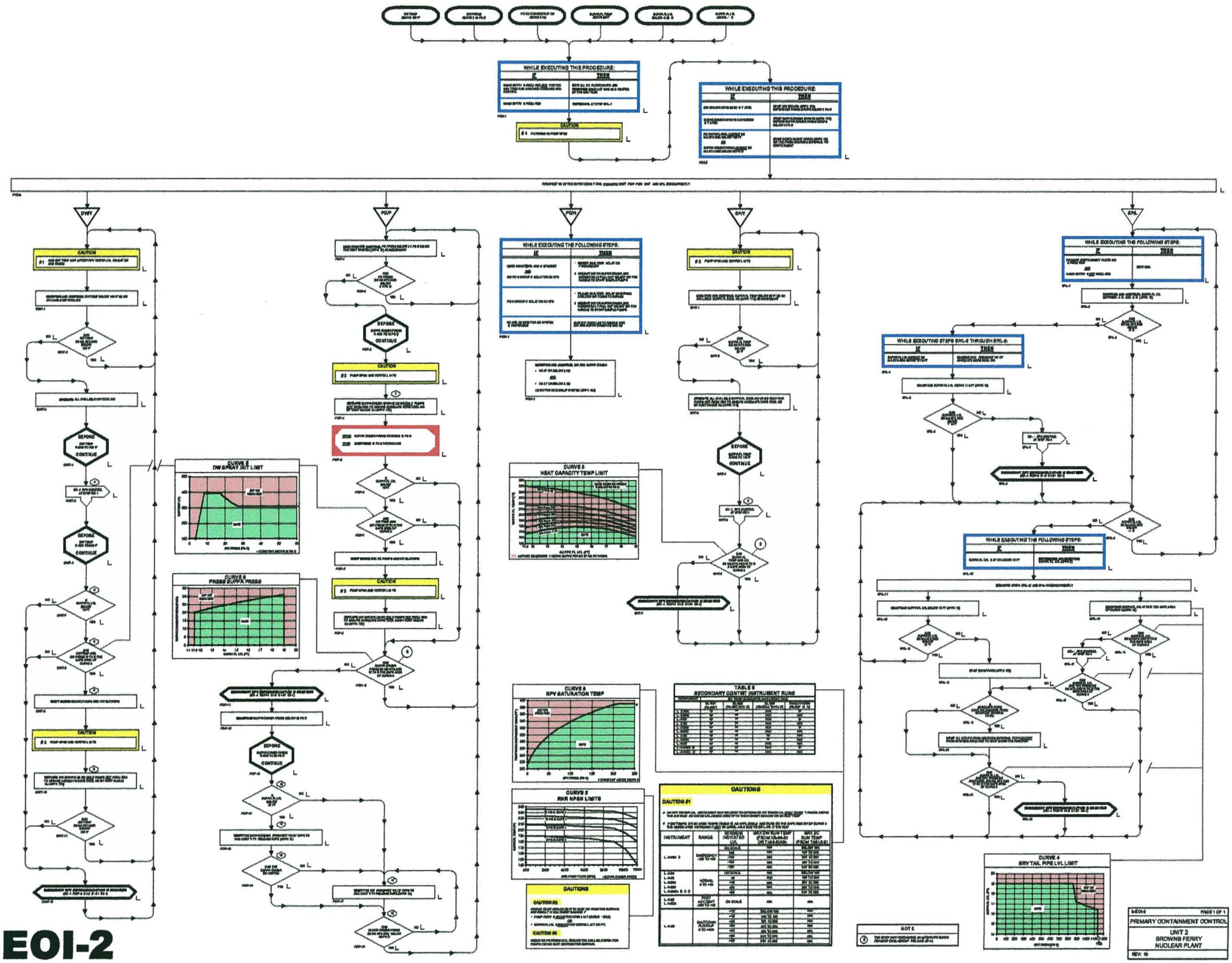
<u>Plausibility Analysis:</u>

<u>A is correct.</u>

<u>B is incorrect.</u> No condition has been met requiring Emergency Depressurization at this temperature. It is plausible if the candidate focuses on the SP level, which is approaching the limit of 11.5 feet for Emergency Depressurization. If this is the case, other actions on SP/L take priority over ED.

<u>C is incorrect.</u> ED would not be anticipated under these conditions unless the candidate focuses on SP level which is approaching the limit of 11.5 feet for Emergency Depressurization. If this is the case, other actions on SP/L take priority over ED.

<u>D is incorrect.</u> It is plausible if the candidate continues down the SP/T leg of EOI-2 and determines that exceeding the HCTL is possible. Since EOI-1 is used to lower pressure and cooldown, and no EOI-1 entry condition has been met, it is unacceptable to assume that exceeding the HCTL is possible.

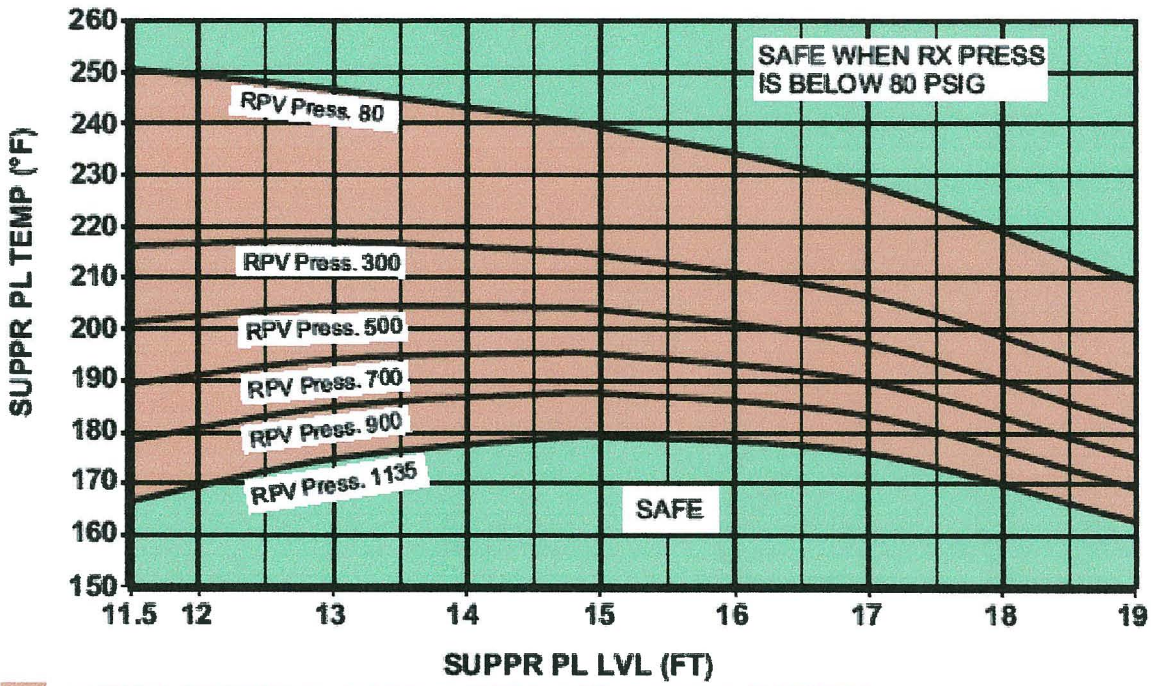


EOI-2

2-EOI-2 PAGE 1 OF 1
PRIMARY CONTAINMENT CONTROL
UNIT 2
BROWNS FERRY
NUCLEAR PLANT
REV:

EXAMINATION REFERENCE PROVIDED TO CANDIDATE

CURVE 3
HEAT CAPACITY TEMP LIMIT



ACTION REQUIRED IF ABOVE CURVE FOR EXISTING RX PRESS

53. RO 295028EK3.04 001/C/A/T1G1/480VLS/B5/295028EK3.04//RO/SRO/

Given the following plant conditions:

- A Loss of Off-site power has occurred in conjunction with a LOCA on Unit-2.
- Plant conditions are as follows:

- Reactor Water Level	+20 inches, steady
- Average Drywell Temperature	230°F, rising
- Suppression Chamber Pressure	11 psig, rising
- EDGs	Tied and loaded to 4 KV Sd Bds
- Reactor pressure	Remains > 800 psig

Which ONE of the following describes the final status of Unit 2 Drywell cooling?

A.✔ Drywell coolers are operating with RBCCW available.

B. Drywell coolers are operating but no RBCCW is available.

C. Drywell coolers must be manually restarted, RBCCW is available.

D. Drywell coolers must be manually restarted, RBCCW is unavailable.

K/A Statement:
295028 High Drywell Temperature / 5
EK3.04 - Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL TEMPERATURE : Increased drywell cooling

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions to determine the status of drywell cooling following a transient which results in high drywell temperature.

References:

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. Whether a CAS 480V Load Shed has been initiated based on the given conditions.
2. The status of RBCCW and DW Blowers based on the answer to Item #1 above.

A is correct.

B is incorrect. This is plausible because the DW blowers would be operating. However, RBCCW does not receive a trip signal because a 480V Load Shed signal has not yet been iitiated. RPV level and pressure are too high.

C is incorrect. This is plausible because following a 480V Load Shed, the DW blowers on the accident unit must be manually started and RBCCW would be available. However, a 480V Load Shed signal has not yet been iitiated. RPV level and pressure are too high.

D is incorrect. This is plausible because following a 480V Load Shed, the DW blowers on the accident unit must be manually started. However, RBCCW does not receive a trip signal because a 480V Load Shed signal has not yet been iitiated.

X. Lesson Body

A. The 480V Load Shedding Logic System removes selected loads from 480V boards which are powered from the 4kV Shutdown Boards — Obj. V.B.1/V.D.1; TP-1, 2

1. The load shedding is initiated by an accident signal on Unit 1 or 2 with a **diesel generator supplying one 4kV Shutdown Board as its only source of power** — Obj. V.B.3/ V.D.3; Obj. V.C.2

AND

2. The accident signal is generated in the Core Spray System logic — TP-3; Obj. V.B.2./V.D.2; Obj. V.C.1

   a. Low-low-low reactor water level (-122"/Level 1)

   OR — CASA signal

   b. High drywell pressure (2.45 psig) with low reactor pressure (450 psig)

   c. For load shed signal on U1 or U2, the accident is for either unit

   d. Unit 3 accident signal won't cause Unit 1 or 2 load shed or vice versa

3. The signal representing "diesel generator supplying a 4KV shutdown board" is called "DGVA" — TP-4

4. For DGVA logic to be satisfied, both conditions must be present:

   a. The DG output breaker or the U2 tie breaker to U3 being closed

   b. The normal and alternate feeder breaker must be open

5. All Unit 1-2 DGVA contacts are in parallel

6. Any D/G tied to its Shutdown Board with an accident signal present will initiate U1-2 load shed logic

3.0 AUTOMATIC ACTIONS (continued)

V. Unit 1/2 480V Load Shed occurs on a loss of offsite power in conjunction with a LOCA signal:

1. One RBCCW pump auto restarts (after 40 seconds on U1 and U2).

2. Drywell Blowers auto restart on non-accident unit (after 40 seconds). Drywell Blowers with their respective auto restart inhibit switches in the INHIBIT position will not auto restart.

3. Drywell coolers are manually restarted on the accident unit. A Drywell Blower with its auto restart inhibit switch in the INHIBIT position can be manually restarted after a ten minute time delay.

4. SGT TRAINS A & B trip, but will AUTO RESTART in 40 seconds when an initiation signal is present.

5. Loss of Control Bay Chilled Water Pumps A & B. (may be restarted after 10 minutes with use of bypass switch).

W. Unit 3 480V load shedding occurs as follows:

1. Division I 480V load shedding will occur when an accident signal is present and diesel generator voltage is available on the 4160V shutdown board supplying the 480V shutdown board 3A as follows:

   a. RBCCW pump 3A trips

   b. Drywell blowers 3A1 & 3A2 trip

   c. After a 40 second time delay, with the control switch in Normal After Start, RBCCW pump 3A restarts

   d. After a 40 second time delay, Drywell blowers 3A1 and 3A2 can be manually restarted

   e. Drywell blowers 3A3, 3A4 and 3A5 cannot be restarted until the load shed signal is corrected

54. RO 295030EA1.06 001/C/A/T1G1/3.5/3.5//295030EA1.06//RO/SRO/11/20/07 RMS

Given the following plant conditions:

- A LOCA has caused gross fuel failure on Unit 3.
- The SED/SRO has approved implementation of EOI Appendix 18, Suppression Pool Water Inventory Removal and Makeup.
- The control room crew has just closed 3-FCV-74-63, RHR RADWASTE SYS FLUSH VALVE
- Suppression Pool level is -3.5 inches and steady.

Which ONE of the following describes the next appropriate action(s)?

A. Open 3-FCV-74-62, RHR MAIN CNDR FLUSH VALVE and direct suppression pool water to the Main Condenser ONLY.

B. Re-open 3-FCV-74-63, RHR RADWASTE SYS FLUSH VALVE and direct suppression pool water to Radwaste ONLY.

C. Verify open the 3-FCV-73-40, HPCI CST SUCTION VALVE, and open 3-FCV-73-30, HPCI PUMP MIN FLOW VALVE

D✓ Appendix 18 is complete; the Suppression Pool level is acceptable.

K/A Statement:
295030 Low Suppression Pool Water Level / 5
EA1.06 - Ability to operate and/or monitor the following as they apply to LOW SUPPRESSION POOL WATER LEVEL: Condensate storage and transfer (make-up to the suppression pool): Plant-Specific

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions to determine the effectiveness of actions to control Suppression Pool level using the Condensate storage and transfer system.

References: 2-EOI Appendix 18

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. Which actions are required based on the given conditions.

NOTE: Each distractor is plausible because they are all actions directed by 2-EOI Appendix 18 to control Suppression Pool level.

A is incorrect. The given SP level is sufficiently low enough that additional inventory removal is not necessary. In addition, following a gross fuel failure, rejecting water to the main condenser is also inappropriate.

B is incorrect. The given SP level is sufficiently low enough that additional inventory removal is not necessary. However, following a gross fuel failure, rejecting water to Radwaste is more appropriate than to the main condenser.

C is incorrect. The given SP level is sufficiently high enough that additional inventory makeup is not necessary.

D is correct.

3-EOI APPENDIX-18

SUPPRESSION POOL WATER INVENTORY REMOVAL AND MAKEUP

LOCATION: Unit 3 Control Room

ATTACHMENTS: None (√)

****

CAUTION

[NRC/C] Suppression Pool water will be highly radioactive after a LOCA. Chemical Engineering recommendations are used to determine location to pump contaminated water. [NRC Inspection Report 89-16]

****

NOTE: All panel operations performed at Control Room Panel 3-9-3 unless otherwise stated.

1. IF .....Suppression Pool Water makeup is required,
   THEN ...**CONTINUE** in this procedure at Step 5. ____

2. IF .....Gross fuel failure is suspected,
   THEN ... **OBTAIN** SED/SRO permission to pump down Suppression Pool BEFORE continuing in this procedure. ____

3. IF .....Directed by SRO,
   THEN ...**REMOVE** water from Suppression Pool as follows:

   a. **DISPATCH** personnel to perform the following (Unit 3 RB, El 519 ft, Torus Area):

      1) **VERIFY OPEN** 3-SHV-074-0786A(B), RHR DR PUMP A(B) DISCH SHUTOFF VALVE. ____

      2) **OPEN** the following valves:

         - 3-SHV-074-0564A(B), RHR DR PUMP A(B) SEAL WTR SPLY ____
         - 3-SHV-074-0529A(B), RHR DR PUMP A(B) SHUTOFF VLV. ____

      3) **UNLOCK** and **OPEN** 3-SHV-074-0765A(B), RHR DR PUMP A(B) DISCH. ____

      4) **NOTIFY** Unit Operator that RHR Drain Pump 3A(3B) is lined up to remove water from Suppression Pool. ___

      5) **REMAIN** at torus area UNTIL Unit 3 Operator directs starting of RHR Drain Pump 3A(3B). ____