KEWAUNEE
2005 RO EXAMINATION

1. Given the following:

   - The unit has tripped from 100% power.
   - 15 minutes later when performing Step 12, Maintain Stable Plant Conditions, of ES-0.1, Reactor Trip Response, RXCP A trips.
   - Both SG levels are 10% narrow range.
   - Prior to the RXCP trip, AFW flow was adjusted to maintain current SG levels while maintaining the RCS at 547ºF.

   How would the AFW flow rate be changed, after the RXCP trip, to maintain the same stable conditions?

   (RD 11.2.4, Decay Power Production for 148 Hours After Shutdown, attached.)

   A. AFW flow to both SGs would be lowered equally, with total flow being approximately 13% less than before the RXCP trip.

   B. AFW flow to both SGs would be lowered equally, with total flow being approximately 26% less than before the RXCP trip.

   C. AFW flow to SG A would be lowered significantly below that to SG B, and total flow would be approximately 13% less than before the RXCP trip.

   D. AFW flow to SG A would be lowered significantly below that to SG B, and total flow would be approximately 26% less than before the RXCP trip.

Answer: C
COGNITIVE LEVEL:

2-DS - Describe the differences between RCS heat production and required AFW flow before a RXCP trip and after the RXCP trip.

K/A:

003K3.03 – Knowledge of the effect that a loss or malfunction of the RCPS will have on the following: Feedwater and emergency feedwater.

OBJECTIVE:

RO4-05-LP002.004 - EXPLAIN how each of the Three types of IHR events will affect the various terms in the equations:

$Q_{RCS} = Q_{SG}$ $\qquad$ $Q_{SG} = UA(T_{avg} - T_{stm})$

$Q_{RCS} = M_{RCS}\,C_P\Delta T$ $\qquad$ $Q_{SG} = M_{SG}\,C_P\Delta T$

REFERENCES:

Kewaunee Thermodynamics Theory, Chapter 7, Thermodynamic Evaluation of Energy Transformations in the Steam Generators, pages 7-22 - 7-24.

Kewaunee Thermodynamics Theory, Chapter 12, Establishment of Initial PWR Plant Parameters, page 12-17.

Reactor Data Manual RD 11.2.4, Rev 8.
Reactor Data Manual RD 11.3.1, Rev 10/28 2004 (USAR Table 4.1.5)

PROVIDED REFERENCE: RD 11.2.4, Decay Power Production for 148 Hours After Shutdown

SOURCE:

New

JUSTIFICATION:

At 15 minutes following a trip the decay heat output from the core is 26.65 MWt (RD 11.2.4). The heat input from each RXCP is 4763 kW (4.76 MWt) under hot coolant conditions (RD 11.3.1). Thus total heat production from RCS prior to the RXCP trip is approximately 36 MWt. Steam and feed flow would be reduced the percentage of total heat that the one RXCP contributes, which is 4.76/36 x 100% or approximately 13%. Since more of the heat (since the flow rate is higher) is transferred to the SG with the running RXCP. The flow rate to the SG in the loop with the stopped RXCP must be reduced more than that to the other SG.

A: The reverse flow in the loop with the tripped RXCP would be only about 15% of normal flow within approximately 15 seconds following the trip of the RXCP. Since the flow is drastically lower for that loop, and heat transfer for that loop will also be lower and thus the need for AFW flow is lower. As show above the total AFW flow rate would be about 13% less since the total RCS heat output is about 13% less at the time the RXCP trips.

B: The reverse flow in the loop with the tripped RXCP would be only about 15% of normal flow within approximately 15 seconds following the trip of the RXCP. Since the flow is drastically lower for that loop, and heat transfer for that loop will also be lower and thus the need for AFW flow is lower. 26% is double the heat input loss for a RXCP.

D: As shown above the AFW flow to SG A would need to be significantly less than that to SG B. 26% is double the heat input loss for a RXCP at the time the RXCP trips.

2. Given the following:

- Charging Pumps A and B are running in MANUAL.
- Przr Level Control Channel Selector is in the NORMAL 2-3 position.
- Instrument Bus I, BRA-113, is de-energized inadvertently.

How are CVCS charging flow and RXCP seal injection flow affected?

A. BOTH charging flow to the RCS and seal injection flow to the RXCPs rise.

B. Charging flow to the RCS rises, but seal injection flow to the RXCPs is lost.

C. Seal Injection flow to the RCS rises, but charging flow to the RCS is lost.

D. BOTH charging flow and seal injection flow to the RXCPs remain constant.

Answer: D
COGNITIVE LEVEL:

2-DR - Relationship between CVCS flows based on loss of instrument power to two valve controllers.

K/A:

004K6.24 – Knowledge of the effect of a loss or malfunction on the following CVCS components: Controllers and positioners.

OBJECTIVE:

RO2-05-LP035.002 - **DESCRIBE** the Chemical And Volume Control System to include the following in the description;
2. Function/purpose, design basis, operating characteristics, and physical location as appropriate for the following major components:
   b. Charging Subsystem
      4. Charging Flow Control Valve: CVC-7
   c. RXCP Seal Water Injection and Return Subsystem
      1. Seal Injection Filter Block Valve: CVC-200

REFERENCES:

E-845, Rev. BE
E-2025, Rev. AD
E-3000, Rev. N

SOURCE:

New

JUSTIFICATION:

CVC-7 (Charging Control Charging Line) and CVC-200 (Seal Injection Filter Block valve) are control valves but remain unaffected by loss of Instrument power. Note that if a charging pump was in AUTO, the input from Przr level LT-426 (failing low) would result in an increase in the AUTO pump speed; and therefore, a corresponding increase in both charging flow and seal injection flow. This would occur only if the selector was in 2-1 position. It is normally selected to 2-3.

A: Charging header valve positions do not change. If in AUTO Charging Pump speed, flow would rise if controlling Przr level channel was from Bus I. (It is not normally selected to this channel.)

B: This would occur if CVC-200 closed, but it does not.

C: This would occur if CVC-7 closed, but it does not. Note that the Charging Header flow instrument, FI-128, does lose power and fails downscale (ZERO).

3. Given the following:

- The plant is at 100% power.
- Reserve Aux Transformer (RAT) locked out at the same time as a SI signal was generated.

Which of the following identifies an expected starting time from the initiating event for the associated ECCS Pump?

A. IMMEDIATELY upon generation of the signal, SI Pump A starts.

B. FIVE seconds after the signal is generated, SI Pump B starts.

C. TEN seconds after the signal is generated, RHR Pump A starts.

D. TEN seconds after the signal is generated, RHR Pump B starts.

Answer: C
COGNITIVE LEVEL:

2-RI - Recognize the interaction between the ECCS components and sequencer operation for various bus conditions.

K/A:

006K2.01 – Knowledge of bus power supplies to the following: ECCS pumps.

OBJECTIVE:

RO2-05-LP033.004 - **DESCRIBE** the operation, automatic actions, interlocks, instrumentation, controls and alarms associated with the Safety Injection System and the following major system components:
2. Safety Injection Pumps.

RO2-05-LP034.004 - **DESCRIBE** the operation, automatic actions, interlocks, instrumentation, controls and alarms associated with the Residual Heat Removal System and the following major system components:
5. RHR pumps.

RO2-03-LP42A.004 - **DESCRIBE** the operation, automatic actions, interlocks, instrumentation, controls and alarms associated with the Emergency Diesel Generator and TSC Diesel Generator Systems and the following major system components:
6. DG start sequence

REFERENCES:

E-1637, Rev. Y
E-2000, Rev. Z
System Description 42, Rev. 2, 3.1, page 5; 3.8.10

SOURCE:

New

JUSTIFICATION:

Bus 5 (Train A ESF) is normally fed from the Tertiary Aux Transformer (TAT). On a loss of the RAT, the power supply to Bus 5 is NOT affected. The RHR Pump A starts at step 2 on the SI sequencer at 10 seconds.

With lockout on the RAT, Bus 6 (Train B ESF) Voltage Restoration takes place. Since the TAT is unavailable to supply it (limited to carrying one ESF Bus), Bus 6 will be aligned to and load equipment on its DG. The DG meets a design criteria of the ability to pick up load within 10-seconds after starting. Thus there is up to a 10-second delay of loading ESF equipment on Bus 6.

A: The ESF sequencer does not start pump loads at time ZERO, but will allow non-shed equipment start on the DG. The SI Pumps have a start step of 1 at 5 seconds.

B: As above, SI Pumps have a start step of 1 at 5 seconds. However for Bus 6 the delay for powering from the DG allows up to 10-seconds. Thus SI Pump B will start some time after 5-seconds (up to 15 seconds).

D: The RHR pumps have a start step of 2 at 10 seconds. However for Bus 6 the delay for powering from the DG allows up to 10-seconds. Thus RHR Pump B will start some time after 10-seconds (up to 20 seconds).

4. While conducting a walkdown of the Main Control Boards just prior to reactor startup, the following ECCS valve positions are noted:

- SI-11A, Safety Injection to Cold Loop A Cold Leg, indicates CLOSED.
- SI-11B, Safety Injection to Cold Loop B Cold Leg, indicates CLOSED.
- SI-15A, Safety Injection To Reactor Vessel, indicates OPEN.
- SI-15B, Safety Injection To Reactor Vessel, indicates OPEN.

Are these valves in the required positions for reactor startup?

A. Yes, valves are in required positions.

B. No, SI-11A and SI-11B should be OPEN AND SI-15A and SI-15B should be CLOSED.

C. No, All valves listed should be OPEN.

D. No, All valves listed should be CLOSED.

Answer: B
COGNITIVE LEVEL:

1-F - Identify the factual positions for ECCS valves while at normal operating pressure & temperature.

K/A:

006A3.06 – Ability to monitor automatic operation of the ECCS, including: Valve lineups.

OBJECTIVE:

RO2-05-LP033.001 - **DESCRIBE** the Safety Injection System to include the following in the description;
1. System flow path.

RO2-05-LP034.004 - **DESCRIBE** the operation, automatic actions, interlocks, instrumentation, controls and alarms associated with the Safety Injection System and the following major system components:;
4. Motor Operated Valves
   e. SI-11A(B), Safety Injection Loop A(B) Cold Leg Isolation.
   f. SI-15A(B), Reactor Vessel Safety Injection Isolation.

REFERENCES:

N-SI-33-CL, Rev. AH, 4.1, page 4 and APPENDIX A

SOURCE:

New

JUSTIFICATION:

The valve positions for SI-11A/B are required prior to the RCS exceeding 1000 psig.

A: The given positions are the opposite of the required positions. SI-9A, Safety Injection to RCS Cold Legs and SI-9B, Safety Injection to Reactor Vessel, which are upstream of SI-11A/B and SI-15A/B are required to be open above 1000 psig also.

C: SI-15A/B are required to be CLOSED per the lineup. SI-9A, Safety Injection to RCS Cold Legs and SI-9B, Safety Injection to Reactor Vessel, which are upstream of SI-11A/B and SI-15A/B are required to be open above 1000 psig also.

D: SI-11A/B are required to be OPEN above 1000 psig.

5. Given the following:

- The plant was operating at 100% power when a reactor trip and SI occurred
- Shortly following the SI actuation the following conditions were observed:

  - R-7, Incore Seal Table Area monitor high alarm actuated
  - Other Containment Radiation monitors showed a rapid increase in radiation levels.
  - Containment humidity spiked to 100%.
  - Containment pressure has a rising trend.
  - 47031-Q CONTAINMENT SUMP A LEVEL HIGH actuated.
  - 47031-R REACTOR CAVITY SUMP LEVEL HIGH/LOW is normal.
  - 47043-B PRESSURIZER RELIEF TANK ABNORMAL actuated.

Assuming NO operator action was taken, which of the following would result in these conditions?

A. RC-45B, Reactor Head Vent Train B, has failed open.

B. An Incore Thimble Tube has ruptured at the bottom of the reactor vessel.

C. PR-2B, PRZR PORV, has stuck open.

D. RXCP B #1 seal has failed.

Answer: C
COGNITIVE LEVEL:

2-DR - Describe the relationship between given containment conditions and the initiating event that would result in these indications.

K/A:

007K3.01 – Knowledge of the effect that a loss or malfunction of the PRTS will have on the following: Containment.

OBJECTIVE:

RO2-01-LP36B.001 - **DESCRIBE** the function/purpose, design basis, and operating characteristics of the PRZR and PRT Systems.

2. Function/purpose, design basis, operating characteristics, and physical location as appropriate for the following major components:
   f. Power Operated Relief Valves (PORVs)
   h. Pressurizer Relief Tank (PRT)

RO4-03-SED04.004 - **Respond** to the following In accordance with Annunciator Procedures 47042A + 47043D and A-RC-36D, Reactor Coolant Leak:
a. PRZR PORV failure OPEN.

REFERENCES:

System Description 36, Rev. 4, 3.6.8, page 33.

SOURCE:

BANK (INPO NRC Exam 12/11/2000)

JUSTIFICATION:

A: RC-36B alone failing open will not result in relief to the containment or the PRT. Opening of RC-49, RX/PRZR Head Vent to Containment, in addition, would result in abnormal containment radiation conditions as listed but not the abnormal PRT condition. Opening of RC-46, RX/PRZR Head Vent to PRZR Relief tank would result in abnormal conditions in PRT, but since the flow restrictor in the line limits flow to to less than the flow capacity of one Charging Pump (60 gpm) it is unlikely to result in pressurization of the PRT to rupture disc failure in such a short time. Also if this were to occur SI actuation is not likely.

B: Incore thimble tube failure is likely to result in the containment radiation conditions described but will not result in the abnormal PRT conditions.

D: Failure of the RXCP seal in conjunction with Si actuation is likely to result in abnormal PRT conditions as the Seal Return Line is isolated (CVC-112 and CVC-111), and the excess letdown relief lifts at 150 psig to the PRT. However the amount of flow from a failed # 1 seal is limited and would not result in pressurization of the PRT to rupture disc failure in such a short time.

6. Given the following:

- The plant is at 100% power.
- RXCP A Component Cooling Return Flow Indicating Switch, fails HIGH.

What effect, if any, does this have on CC for the RXCP A?

A. It has <u>NO</u> effect on CC flow to the RXCP A.

B. CC-600, Component Cooling to RXCPs & Excs Ld Hx, closes, isolating ALL CC flow to Containment.

C. CC-601A, Component Cooling to RXCP A, and CC-612A, RXCP A Component Cooling Return Isol, close, isolating ALL CC flow to RXCP A.

D. CC-610A, RXCP A Thermal Barr Comp Cooling Return, closes, isolating CC flow from RXCP A thermal barrier.

Answer: D

COGNITIVE LEVEL:

2-DR - Recognize the interrelationship between CC system and the RXCPs to include sense high return flow from RXCPs in CC system.

K/A:

008K1.02 – Knowledge of the physical connections and/or cause-effect relationships between the CCWS and the following systems: Loads cooled by CCWS.

OBJECTIVE:

RO2-01-LP031.004 - **DESCRIBE** the operation, automatic actions, interlocks, instrumentation, controls and alarms associated with the Component Cooling Water System and the following major system components:
h. RXCP A/B Thermal Barr Comp Cooling Return (CC-610A & 610B).

REFERENCES:

A-CC-31, Rev. G, 3.2, Steps 13 (CA), 14, 16 (CA).
System Description 31, Rev. 4, 3.6.6., page 15.
E-2045, Rev. AD

SOURCE:

NEW

JUSTIFICATION:

FICA-613 is a local indicating switch on the CC return header in the Aux Bldg.

A: There is also a local return flow indication in Containment, on the RXCP Thermal Barrier Return Line. This flow indicator has no associated functions but is called out in A-CC-31 to be checked for excessive flow when checking status of RXCP Thermal Barrier.

B: CC-600 is the common line providing CC flow to Containment. It could be isolated to stop leaks on the common header (RXCPs & Excess Letdown Hx).

C: CC-601A and CC-612A are isolated (A-CC-31) if CC piping to/from RXCPs is expected to be leaking.

7. Given the following:

- A load rejection has occurred from 100% power.
- Reactor power is now 80%.
- RCS Tave is 572ºF.
- Pressurizer level is 50%.
- Pressurizer pressure is 2300 psig.

Which of the following is an expected Pressurizer control systems indication?

A. Backup heaters are ON.

B. Pressurizer spray valves are modulated open.

C. The Charging Pump in AUTO is at maximum speed.

D. Pressurizer spray valves AND Pressurizer PORVs are open.

Answer: B
COGNITIVE LEVEL:

1-I - Knowlege of the setpoints for system operation for Przr pressure and level control .

K/A:

010A4.01 – Ability to manually operate and/or monitor in the control room: PZR spray valve.

OBJECTIVE:

RO2-05-LP36C.004 - **DESCRIBE the** operation, automatic actions, interlocks, instrumentation, controls and alarms associated with the PRESSURIZER PRESSURE CONTROL System and the following major system components:
1. Pressurizer Pressure Controllers.
2. Pressurizer Spray Valves.
3. Pressurizer Heaters
5. Power Operated Relief Valves

RO2-05-LP36D.004 - **DESCRIBE** the operation, automatic actions, interlocks, instrumentation, controls and alarms associated with pressurizer level control and the following major system components:
3. Level programmer.

REFERENCES:

System Description 36, Rev. 4, 3.7.2, page 36, and 3.7.4.
E-2038, Rev. AC
47043-C, Rev. D, Comments 2.
47043-E, Rev. C, Comments 2.

SOURCE:

MODIFIED
Point Beach Bank

Changed values for pressure level and temperature.
Changed question to include both Przr pressure and level programs for possible conditions.

Eliminated reference to Control heaters since they will modulate (even to no output) if pressure is high. Used Charging Pump speed for consideration based on Przr level and also made Backup heaters a more plausible choice since they also respond to Przr level.

JUSTIFICATION:

Level program is 21% @ 547ºF to 46.7% @ 572ºF.
Back up heaters will energize if Przr level is 10% above setpoint level
Charging Pump speed is inversely related to the error signal from level. If Przr level is below setpoint, Charging Pump speed will be increased to attempt to raise level. If level is above setpoint, Charging Pump speed will be lowered to reduced to allow letdown to restore level.

Pressurizer pressure control has:
Backup heaters ON at 25 psig below Master Control setpoint (2235 psig).
Przr Sprays will modulate open between 2260 psig and 2310 psig.
PORVs open at nominal setpoint of 2335 psig (and reclose at 2315 psig).

A: Przr pressure is above setpoint and Przr level is not 10% above setpoint (50% - 46.7 = 3.3%).

C: Charging Pump speed would be running to or at Minimum since Przr level is above setpoint.

D: Spray valves would be modulated & PORVs would not be open at this pressure.

8. Concerning the Engineered Safety Features (ESF) initiation instrumentation for Pressurizer pressure, there are _____(1)_____ channels that input to Safety Injection for _____(2)_____ independent safety trains of ESF.

	__(1)__	__(2)__
A.	4	4
B.	4	2
C.	3	3
D.	3	2

Answer: D
COGNITIVE LEVEL:

1-F - Knowlege of the specific instrumentation trains as related to ESF inputs.

K/A:

013K5.01 – Knowledge of the operational implications of the following concepts as they apply to the ESFAS: Definitions of safety train and ESF channel.

OBJECTIVE:

RO2-05-LP055.002 - **DESCRIBE** the Engineering Safety Features System to include the following:

1. Function/purpose, design basis, operating characteristics, and physical location, as appropriate, for the following major components:
   a. Instruments required for ESF Actuation
   b. ESF Actuation Signals

REFERENCES:

KNPP Technical Specifications, Amend. 172, Table TS 3.5-3, Item 1.d.
XK-100-546, Rev. 2S
USAR, Rev. 19, 7.2.2, 2nd Paragraph, page 7.2-14 and Low Pressure, page 7.2-31.

SOURCE:

MODIFIED
Braidwood 7/2002 NRC Exam

Question changed ESF instrumentation From SG level to Przr pressure.

Changed 3 of 4 selections to represent the instrument logics at KNPP, and subsequently changed the number of trains to reflect these changes. Correct answer changed due to KNPP specifics.

JUSTIFICATION:

There are 4 channels of Pressurizer pressure. Only 3 of these are used to input to ESFAS.

A: There are 4 channels Pressurizer pressure. The fourth channel inputs to Reactor Protection Trip (RPS). There are 2 channels for ESF actuation (A & B).

B: There are 4 channels Pressurizer pressure. The fourth channel inputs to Reactor Protection Trip (RPS). There are 2 channels for ESF actuation (A & B).

C: There are 4 channels Pressurizer, three of which input to SI. The fourth channel inputs to Reactor Protection Trip (RPS). There are 2 channels for ESF actuation (A & B)

9. Given the following:

- A reactor trip and safety injection has occurred.
- A main steamline break in Containment has been diagnosed.
- Containment pressure is 4.8 psig.
- Containment humidity is 100%.
- Both RXCPs are running.
- During the performance of E-0, Reactor Trip Or Safety Injection, the operator reports RBV-150A and RBV-150B, CNTMT Fan Coil Unit Emergency Discharge Dampers are CLOSED.
- RBV-150A and RBV-150B CANNOT be opened.

What is the area of concern, related to the Containment Air Cooling System design, with the system operating in this condition?

A. Overheating of RXCP B.

B. Loss of cooling for Shroud Cooling Coils if natural circulation cooldown is required.

C. Reduced Containment cooling due to the lack of air mixing in Containment because of damage to the normal ducts.

D. Reverse airflow through the Containment Fan Coil Units causing damage to the Containment Fan Coil Unit motors.

Answer: C
COGNITIVE LEVEL:

1-F - Knowlege of the design operation of the Containment Fan Coil Units

K/A:

022K4.05 – Knowledge of CCS design feature(s) and/or interlock(s) which provide for the following: Containment cooling after LOCA destroys ventilation ducts

OBJECTIVE:

RO2-04-LP018.002 - **DESCRIBE** the Reactor Building Ventilation System, include the following in the description;
2.Function/purpose, design basis, operating characteristics, and physical location as appropriate for the following major components:
a. Containment Fan Coil Units A, B, C, and D

RO2-04-LP018.004 - **DESCRIBE** the operation, automatic actions, interlocks, instrumentation, controls and alarms associated with the Reactor

Building Ventilation System and the following major system components
1. Containment Fan Coil Unit A (B) (C) (D).
2. Containment FCU A&B (C&D) Discharge to Refueling Floor (no control switch).
3. Containment Fan Coil Unit A (B) (C) (D) Emergency Discharge Damper RBV-150A (B) (C) (D).

REFERENCES:

System Description 18, Rev. 1, 3.3. pages 11 and 12
OPERM-602, Rev. BB
USAR, Rev. 19, 6.3.2, Flow Distribution and Flow Characteristics, page 6.3-5.

SOURCE:

KNPP Bank Question RO2-04-LP018.004 003

JUSTIFICATION:

A: The normal alignment is with normal dampers open and provides cooling to the RXCP vaults. If the Emergency Damper actuates or opens during normal operation cooling air flow is lost to the RXCP vaults, and there is concern with RXCP overheating. (In this case the Emergency dampers have failed closed.)

B: Service Water flow is altered to the Shroud Cooling Units due to the realignment for the SI signal. However the air flow path through the Shroud Cooling Coils is not affected.

D: Reverse flow though any idle fan is not a concern since the units are provided with reverse flow dampers to prevent backflow. These dampers remain unaffected by the Emergency Damper failures.

10. Given the following:

- A LOCA has occurred.
- Containment Spray has actuated.
- ICS pumps are running and delivering flow.
- RWST level currently reads 40%.
- Caustic Additive Standpipe level currently reads 100%.

Which of the following describes the effect of these conditions and the actions necessary to mitigate the event?

A. Containment radiation levels are higher due to the increased radioactive noble gas production.
Press both CONTAINMENT SPRAY START pushbuttons.

B. Containment pressure peaks at a higher value due to the reduced heat removal capacity of the ICS spray.
Manually open CI-1001A and B, Caustic Additive To CNTMT Spray.

C. Corrosion of components in containment increases due to lower pH value of the containment sump fluid.
Manually open CI-1001A and B, Caustic Additive To CNTMT Spray.

D. Removal of hydrogen in the containment atmosphere is lower due to the reduced volume of injected sodium hydroxide.
Press both CONTAINMENT SPRAY START pushbuttons.

Answer: C
COGNITIVE LEVEL:

2-DR - Recognition of the relationship between Containment Emergency Sump pH and Spray Additive. Also recognize actions required to establish proper alignment.

K/A:

026A2.05 – Ability to (a) predict the impacts of the following malfunctions or operations on the CSS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Failure of chemical addition tanks to inject.

OBJECTIVE:

RO2-01-LP023.002 - **DESCRIBE** the ICS System. Include the following in the description:
2. Function/purpose, design basis, operating characteristics, and physical

location as appropriate for the following major components:

a. Caustic Additive Standpipe.

REFERENCES:

System Description 23, Rev. 4, 1.1, 2.0, and 3.7, page 10.
USAR, Rev. 19, 6.4.1, page 6.4-1.

SOURCE:

KNPP Bank Question RO2-01-LP023.002 003
Altered to add actions but not MODIFIED

JUSTIFICATION:

A: The caustic does not affect the radioactive noble gases in Containment. It does reduce the Containment atmosphere activity from Iodine and particulates by maintaining Iodine in solution and particulates in the sump. Depressing the manual START should not affect the position of the valves since CNTMT Spray is already running.

B: The heat removal capability of CNTMT Spray is not affected by NaOH. The correct action is to open both valves manually.

D: Hydrogen is not removed from containment atmosphere by NaOH, although Iodine gas is, as well as some particulates. Depressing the manual START should not affect the position of the valves since CNTMT Spray is already running.

11. Operators monitor specific parameters to ensure that the safety analysis assumptions for Shutdown Margin, Ejected Rod Worth, and Power Distribution Peaking Factors are preserved.

Which of the following is a list of these operator monitored parameters?

(Note: A list of abbreviations used in the answer selections is provided below.)

QPTR - Quadrant Power Tilt Ratio
DNBR - Departure From Nucleate Boiling Ratio
AFD - Axial Flux Difference
CHF - Critical Heat Flux

A. QPTR, DNBR, AFD, and Rod Insertion Limits.

B. Rod Alignment Limits, CHF, AFD, QPTR.

C. Rod Insertion Limits, AFD, QPTR, Rod Alignment Limits.

D. RCS Pressure, Rod Insertion Limits, Critical Boron Concentration, CHF.

Answer: C
COGNITIVE LEVEL:

2-DR - Describe the relationship between safety analysis assumptions and the parameters monitored by the operator to ensure compliance.

K/A:

2.1.10 – Conduct of Operations: Knowledge of conditions and limitations in the facility license.

OBJECTIVE:

O-FND-LP 2.3.9, Rev. ORIG, 2.2.5 Discuss reactor operational limits and define quadrant power tilt ratio (QPTR) and axial flux difference.

RO4-05-LP001.006: Describe the consequences asscoiated with each one of four conditions in the ANS Classifications.

RO4-05-LP008.002; List the plant design features which limit consequences of Reactor Power and Distribution Anomalies.

REFERENCES:

WEC Fundamentals of Nuclear reactor Physics, Chapter 6, Summary, Page 6-27.

USAR, Rev. 19, 14.2.6, pages 14.2-27 & 28

TS Basis: TS 3.10.a (2nd pp), TS B3.10-1; TS 3.10.c (1st pp), TS B3.10-5; TS 3.10.d (2nd pp), TS B3.10-5; TS 3.10.e (1st pp), TS B3.10-6;

SOURCE:

INPO Bank Point Beach NRC exam 2/2/2002
Kewaunee NRC exam 9/6/2002

JUSTIFICATION:

A: DNBR is not a parameter that is directly monitored by the operator.

B: CHF is not a parameter that is directly monitored by the operator.

D: CHF is not a parameter that is directly monitored by the operator.

12. What condition is indicated by a bright illuminated status light on the SI READY Panel with the plant operating at full power?

A. The component has lost AC control power.

B. The component has lost DC control power.

C. The component is in an abnormal alignment.

D. The component is in its normal post-accident alignment.

Answer: C
COGNITIVE LEVEL:

1-F - System fact on indication of equipment status.

K/A:

2.1.31 – Conduct of Operations; Ability to locate control room switches, controls and indications and to determine that they are correctly reflecting the desired plant lineup.

OBJECTIVE:

RO2-05-LP055.004: **DESCRIBE** the operation, automatic actions, interlocks, instrumentation, controls and alarms associated with the Engineering Safety Features System and the following major system components:
5. SI Ready Status Panel - 44909

REFERENCES:

System Description 55, Rev. 1, 3.8
E-2032, Rev. W (SI-5A/B: 44909-0101, -0105)
E-2036, Rev. AO (RHR-101, RHR-8A/B: 44909-0503, 44909-0401, -0405)

SOURCE:

Point Beach 1 2/2002 NRC Exam (INPO Exam Bank)

JUSTIFICATION:

A: This condition used to be represented by a Thermal Overload trip of the beakers supplying power to SI-2A/B, Boric Acid Tank Outlet Isols. and was displayed as SI-2A THERMAL OLOAD TRIP (0102) and SI-2B THERMAL OLOAD TRIP (0106) status lights. This was removed by a

Design Change Request (DCR).

B: This condition does not exist but is a plausible choice based on "A" selection.

D: This is part of the SI ACTIVE Status Panel and CI ACTIVE Status Panel functions. These panels have lights that become brightly lit when the component is in its expected post-accident condition (valve open or close; pump or fan running).

13. Given the following:

- Reactor power is 1.5 x 10^{-3}% during a reactor startup.
- ECP Data is being recorded in the Control Room Log.
- Annunciator 47041-P, ROD BOTTOM ROD DROP alarmed.
- A reactor trip signal was NOT generated.
- Control rod G-7 has dropped into the core.
- Individual rod position indication for G-7 reads ZERO steps.
- Intermediate Range SUR reads negative 0.2 dpm.

What action(s) would the operator be required to take?

A. Manually trip the reactor and go to E-0, Reactor Trip or Safety Injection.

B. Maintain present power level by initiating a dilution per N-CVC-35B, Charging and Volume Control.

C. Shutdown the reactor by inserting control rods and opening the reactor trip breakers per N-CRD-49C, Reactor Shutdown.

D. Recover control rod G-7 to the current Bank C Group 2 position using A-CRD-49, Abnormal Rod Control System Operations, ATTACHMENT A, and then continue the reactor startup.

Answer: C
COGNITIVE LEVEL:

3-SPK - Solve the problem of the action to take with the reactor critical, but not in the power range and a dropped rod occurs that results in a negative SUR.

K/A:

2.1.31 – Conduct of Operations; Ability to perform specific system and integrated plant procedures during all modes of plant operation.

OBJECTIVE:

RO4-02-LPD13.002: **PERFORM** a Reactor Startup in accordance with N-0-02 and N-CRD-49B and the following supporting procedures.

RO4-02-LPD13.003: Given a plant startup from Hot Shutdown, **APPLY** the Precautions and Limitations N-CRD-49B and its supporting procedures per the Operations Precautions and Limitations Reference Book.

RO4-03-LPD013.002: In accordance with A-CRD-49, “ABNORMAL ROD

CONTROL SYSTEM OPERATION," **Summarize** the subsequent operator actions that are necessary to respond to the following:
b. Single Dropped Rod at Power.

REFERENCES:

A-CRD-49, Rev. N, Step 1.b. (CA)

SOURCE:

KNPP Bank Question 0490030501A01 002

JUSTIFICATION:

A: Tripping the reactor is only required when two or more rods drop, OR if rods do not stop moving when the the Control Rod Bank Selector is taken to MAN.

B: Dilution or boration is directed to maintain Tavg on program, but only if the reactor remains critical.

D: Rod recovery is allowed at power, but only if the reactor remains critical after the rod drop.

14. Given the following:

- The plant is in REFUELING Mode.
- Fuel movement in Containment is in progress.

What is the responsibility of the Control Operator concerning the Source Range nuclear instrumentation during this operation?

As a MINIMUM, the operator must verify...

A. ONE channel is operating, and it is visually monitored in the Control Room.

B. ONE channel is operating, it is visually monitored in the Control Room and has audible indication in the Containment.

C. TWO channels are operating, each channel is visually monitored in the Control Room, and ONE channel has audible indication in the Control Room.

D. TWO channels are operating, each channel is visually monitored in the Control Room and ONE channel has audible indication in the Containment.

Answer: D
COGNITIVE LEVEL:

1-F - System fact on indication of equipment status.

K/A:

2.2.30 – Equipment Control; Knowledge of RO duties in the control room during fuel handling such as alarms from fuel handling area, communication with fuel storage facility, systems operated from the control room in support of fueling operations, and supporting instrumentation.

OBJECTIVE:

RO2-05-LP048.007: EXPLAIN the LCO, applicability and action requirements for each of the following Technical Specifications associated with the Excore Nuclear Instrumentation System:
4. TS 3.8 REFUELING OPERATIONS

REFERENCES:

SP-87-149, Rev. Q, Data Sheet (page 3)
KNPP Technical Specifications, 3.8.a.3, Amendment No. 165.

SOURCE:

KNPP 12/2000 NRC Exam (INPO Exam Bank)

JUSTIFICATION:

This meets the requirement for minimum neutron flux monitoring during core geometry changes.

A: When core geometry is NOT being changed during REFUELING OPERATIONS, one neutron flux monitor shall be in service and monitored for any flux rate changes during drain-down operations.

B: When core geometry is NOT being changed during REFUELING OPERATIONS, one neutron flux monitor shall be in service and monitored for any flux rate changes during drain-down operations. There is no requirement for the flux to be monitored in Containment. See below that audible indication from one channel is required in containment during core geometry changes.

C: Two channels are only required during core geometry changes. However, if core geometry changes are in progress, then audible indication from one channel is required in the Containment, not the Control Room.

15. Who has the authority to authorize the use of the bypass keys to enable the OVERLOAD BYPASS for the Manipulator Crane when lowering a fuel assembly into the reactor vessel?

A. The Shift Manager only

B. The Refueling SRO in Containment only

C. The Shift Manager or Reactor Engineering

D. The Refueling SRO in Containment and Reactor Engineering

Answer: B
COGNITIVE LEVEL:

1-F - Procedural fact identifying the individual that has the authority.

K/A:
2.2.28 – Equipment Control; Knowledge of new and spent fuel movement procedures.

OBJECTIVE:

RO4-01-LP-A02.003: **DISCUSS** department and department personnel responsibilities associated with the following:
b. NAD 02.02, PLANT ORGANIZATION
1. RO, SRO/Shift Manager

REFERENCES:

NAD-02.07, Kewaunee Refueling Operations, Rev. A, Page 2

SOURCE:

KNPP 12/2000 NRC Exam (INPO Exam Bank)

JUSTIFICATION:

Procedure states that the SRO in Containment controls and authorizes use of the bypass key for refueling interlocks.

A: The Shift Manager approval is NOT required for operation of the bypass.

C: Shift Manager approval is NOT required and Reactor Engineering approval alone is NOT sufficient..

D: The additional Reactor Engineering approval is NOT required.

16. How does the decrease in the effective delayed neutron fraction ($\overline{B}_{eff}$) over core life affect the power response following a reactor trip?

For a trip occurring at near the end of core life compared to a trip occurring early in core life, the prompt drop causes power to fall immediately to ...

A. a larger value, and then power decreases at a slower rate than early in life.

B. the same value, and then power decreases at a faster rate than early in life.

C. the same value, and then power decreases on the same stable period (startup rate) as early in life.

D. a smaller value, and then power decreases on the same stable period (startup rate) as early in life.

Answer: D

COGNITIVE LEVEL:

2-DS - Describe the differences and similarities of prompt drop and power decrease late in core life to that earlier in core life based on the Beta-effective change.

K/A:

007EK1.04 – Knowledge of the operational implications of the following concepts as they apply to the reactor trip: Decrease in reactor power following reactor trip (prompt drop and subsequent decay).

OBJECTIVE:

O-FND-LP 2.1.5, Rev. ORIG,
2.2.24. DESCRIBE how a prompt jump (prompt drop) takes place when reactivity changes.
2.2.29. Sketch the time-dependent behavior of startup rate and reactor power, given the time-dependent behavior of the reactivity.

O-FND-LP 2.2.1, Rev. ORIG
2.2.10. DESCRIBE the effects of fuel burnup on the reproduction factor, ?, thermal utilization, f, and the effective delayed neutron fraction, .

REFERENCES:

WEC Fundamentals of Nuclear Reactor Physics, Chapter 1, Summary, Page 1-29.

WEC Fundamentals of Nuclear Reactor Physics, Chapter 7, Effective Delayed Neutron Fraction, Pages 7-37 & 7-38; Mixture of Fuel NuclidesReactivity Transients, Pages 7-68 & 7-70; Chapter Summary, Pages 7-88 - 7-90.

WEC Kewaunee Core Control Theory, Chapter 1, VI. Summary, page 1-29.

SOURCE:

Bank (INPO Bank)

JUSTIFICATION:

The equation for determining the change in reactivity is given by the inhour equation:

and

the delayed neutron fraction is larger BOL than EOL.

A: The first would occur only if the $\bar{\beta}_{eff}$ had gotten larger over core life. The stable shutdown rate is dependent on the longest lived fission product decay time. This is constant over core life.

B: The first would occur only if $\bar{\beta}_{eff}$ remained constant over core life. The stable shutdown rate is dependent on the longest lived fission product decay time. This is constant over core life.

C: The first would occur only if $\bar{\beta}_{eff}$ remained constant over core life. The stable period remains the same.

17. Given the following:

- Plant heatup is in progress.
- RCS pressure is 100 psig with solid plant conditions.

If an 85 gpm liquid space leak develops in the Pressurizer, what would be the expected leak rate when RCS pressure has decreased to 50 psig?

A. Approximately 21 gpm

B. Approximately 42 gpm

C. Approximately 60 gpm

D. Approximately 71 gpm

Answer: C
COGNITIVE LEVEL:

3-SPK - Solve the problem of the leak rate change knowing the relationship between leak rate and the driving head for the leak (pressure).

K/A:
008AK1.02 – Knowledge of the operational implications of the following concepts as they apply to a Pressurizer Vapor Space Accident: Change in leak rate with change in pressure.

OBJECTIVE:

O-FND-LP1.4.4 , Rev. ORIG, 2.2.6 Explain the relationship between pressure head and velocity head in a fluid system.

REFERENCES:

NRC Generic Fundamentals Examination Question Bank--PWR
August 2005
Fluid Statics and Dynamics.

SOURCE:

Modified Changed initial conditions for where leak originates: RCS vs. cooling system.
Incorrect selection values changed.

NRC Generic Fundamentals Examination Question Bank--PWR

August 2005
Fluid Statics and Dynamics.

JUSTIFICATION:

The solution is given as $P_1/P_2 = (v_1/v_2)^2$. So v_2 is equal to the square root of (50 $(85)^2$ / 100) = 60.1 gpm.

A: 21.25 is the rate if $v_1/v_2 = (P_2/P_1)^2$.

B: 42.5 is the rate a linear relationship existed between pressure and flow rate P_1/P_2. = v_1/v_2.

D: 70.7 is the rate if the value of the pressure difference is multiplied by the square root of 2. (100 -50) x R(2)

18. Given the following:

- A small-break LOCA has occurred.
- Both RXCPs were stopped in E-0, "Reactor Trip Or Safety Injection."
- Charging Pump A was started in E-1, "Loss Of Reactor Or Secondary Coolant."
- Charging Pump B was started 5 minutes ago, charging has been maximized.
- Both SI Pumps are running.
- RCS Subcooling is 31ºF.
- PRZR level at 22%.
- RCS pressure is stable at 800 psig.

The CRS directs one RXCP to be started per ES-1.2, "Post-LOCA Cooldown and Depressurization," step 11.

What precaution should the operator consider prior to starting the RXCP?

A. A steam bubble may exist in the upper head region of the reactor vessel during RXCP startup. The bubble could collapse rapidly, lowering PRZR level and decreasing the subcooling margin.

B. A temperature differential in the RCS loops exists due to seal injection flow with RXCPs stopped. This could result in a rapid pressure increase leading to a Pressurized Thermal Shock condition.

C. When the RXCP is started an insurge into the PRZR will occur. This will increase RCS pressure and RCS subcooling margin.

D. Flow out the break will significantly increase when the RXCP is started. PRZR level and subcooling margin will decrease.

Answer: A
COGNITIVE LEVEL:

3-PEO - Predict an outcome based on knowledge of procedural purpose and usage.

K/A:

009E 2.1.32 – Small Break LOCA: Ability to explain and apply all system limits and precautions.

OBJECTIVE:

RO4-04-LP19.002 - **SUMMARIZE** purposes or bases of the following items as they relate to ES-1.2, “Post LOCA Cooldown and Depressurization”
b. Cautions

e. All Procedural Steps

RO2-01-LP36A.005 - **EXPLAIN** the purpose of the following procedures used to govern the normal, abnormal, and emergency operation of the Reactor Coolant Pumps.
1.a N-RC-36A Reactor Coolant Pump Operation

REFERENCES:

ES-1.2, Rev. Q, Caution prior to Step 10, and Step 11.e
BKG ES-1.2, Rev. B, 4. STEP 10
N-RC-36A, Rev. AG, 2.4

SOURCE:

KNPP Bank Question E010030501K03 003

JUSTIFICATION:

B: This is the Precaution noted in N-RC-36A which requires a bubble in the Przr before starting a RXCP. Note that it is also NOT applicable in accordance with the Background Document since a LOCA has been confirmed and solid RCS conditions do not exist under given conditions.

C: Due to the possibility of head voiding this would not occur as starting the RXCP provides for no additional volume. However, the effect described could occur if such insurge was large enough.

D: The only way to raise break flow is to significantly raise RCS pressure. This would not occur under given conditions just starting a RXCP. The increase in dynamic head of running RXCP may be considered, and the effects of break flow increases are correct.

19. Given the following:

- A Large Break LOCA has occurred.
- Preparations to establish one train of containment sump recirculation are in progress per ES-1.3, Transfer To Containment Sump Recirculation.
- Steps to "Align Charging Pump Suction To The VCT," and "Establish Charging Flow" are about to be completed.

Why are actions taken to (1) align Charging Pump suction to the VCT and (2) establish charging flow?

A. (1) Protect charging pumps from losing suction.
(2) Provide RCP Seal Injection flow.

B. (1) Preclude gas entrainment in the ECCS (RHR and SI) piping.
(2) Provide flow to the RCS while the ECCS (RHR and SI) trains are being swapped to the Containment Sump B.

C. (1) Allow makeup to be established to the RWST.
(2) Provide condition for reestablishing letdown via the Regenerative Hx.

D. (1) Preserve RWST inventory.
(2) Establish conditions for SI termination.

Answer: A
COGNITIVE LEVEL:

1-P - Knowledge of the procedural actions for aligning charging when establishing containment sump recirculation.

K/A:

011EA2.05 – Ability to determine and interpret the following as they apply to a Large Break LOCA: Significance of charging pump operation.

OBJECTIVE:

RO4-04-LP021.002 - **Summarize** the purposes or basis of the following items as they relate to ES-1.3, Transfer to Containment Sump Recirculation.
e. All Procedural Steps

REFERENCES:

ES-1.3, Rev. Z, Step 7.
BKG ES-1.3, Rev. C, 4. STEP 7

SOURCE:

New

JUSTIFICATION:

B: Gas entrainment has been a concern in the industry, normally from systems such as the CVCS where gas is entrained in the RCS fluid. Swapover to the VCT would not affect the condition since the CVCS and ECCS do not cross feed. Normally with both trains operable only one train of ECCS at a time is aligned to the Containment sump. However if RWST level lowers to 4%, any pumps still taking suction off the RWST are stopped.

C: Placing charging suction to the VCT limits the ability to makeup to the RWST since maintaining VCT level with letdown isolated is dependent on the ability of the makeup system to supply the VCT. Therefore as long as makeup is supplied to the RWST it cannot go to maintaining VCT level. Makeup to RWST is a concern in ECA-1.1. Letdown would only be a concern if RCS pressure control was to be established using Aux. Spray from the CVCS. This is not a concern in ES-1.3 for a large-break LOCA.

D: Neither identifies the reasons. While swapping the Charging Pump suction to VCT will provide some relief (maximum of ~100 gpm) from RWST depletion, it is not a consideration. The consideration is to establish the charging capability within the limits of the RMCS so that swapover to an empty RWST does not result in loss of suction to the Charging Pumps. Establishing conditions for SI Termination are not a consideration in ES-1.3, but may be addressed in E-1 or ES-1.2.

20. Given the following:

- The plant is at 50% power.
- Seal injection flow has been lost to the RXCPs.
- CVC-207A, RXCP A #1 Seal Leakoff Isolation valve has been closed due to a high RXCP seal leakoff flow alarm.
- CVC-204A RXCP Seal Supply Line Throttle Valve has been closed.
- After 2 hours, the line blockage was cleared and seal injection is ready to be restored to RXCP A.
- The operator then rapidly reopens CVC-204A to its original position.

What is the result of his actions?

The RXCP ...

A. seal package will be damaged as it undergoes a greater than 1ºF/minute cool down.

B. #1 seal will become cocked due the pressure surge.

C. Thermal Barrier HX will fail due to thermal shock of the tubes.

D. Thermal Barrier HX will become steam bound as the stagnant seal water is flushed into the RCS.

Answer: A
COGNITIVE LEVEL:

2-DR - Recognizing the relationship between the isolation of the seal injection resulting in seal heatup and then cooling when injection reestablished.

K/A:

022AK1.01 – Knowledge of the operational implications of the following concepts as they apply to Loss of Reactor Coolant Pump Makeup: Consequences of thermal shock to RCP seals.

OBJECTIVE:

RO4-03-LPD03.003 - In accordance with A-RC-36C, Abnormal RXCP Operation, SUMMARIZE the subsequent operator actions that are necessary to respond to the following:
1. Loss of Seal Injection

REFERENCES:

A-RC-36C, Rev. T, .Step 9.c. (CA), and Attachment A.
BKG ECA-0.0, Rev. C, 2.1; Benefits and Consequences of Restoring Seal Cooling, page 12

SOURCE:

KNPP 12/2000 NRC Exam (INPO Bank)

JUSTIFICATION:

The RCP Vendor Manual identifies limits for reestablishing seal cooling to a hot seal package to prevent further damage due to thermal shock and to prevent warping of the RXCP shaft due to uneven cooling. These limits are only intended for a loss of seal cooling of short enough duration that the seal package heatup is limited.
Since the cooldown rate of seals is limited to 1 °F per minute regardless of the method used to cool the seals (i.e., thermal barrier cooling would not reduce leakage any faster than an RCS cooldown).

B: During the heatup of the seals due to lack of adequate seal injection or thermal barrier cooling, the thermal gradients affect the faceplate tapers of the number 1 seal ring and runner and the shrink fit of the number 2 seal ring insert, affecting sealing surfaces. Nonuniform thermal gradients and extrusion of O-rings may result in nonuniform sealing surfaces. However, this is the resultant factor of the heatup, not the cooldown by reestablishing seal injection that exceeds recommended cooldown limits.

C: Thermal Barrier tube failure is likely to occur only if both seal injection and component cooling flow was lost for a period of time. The component cooling water flow in the thermal barrier maintains a thermal gradient across the tubes with the cooler side being the inside diameter. When seal injection is reestablished, with flow now also being directed to the RCS, the thermal gradient will be lessened (OD side will experience compression as cools). This is not a situation for thermal failure.

D: As above, steam voiding in the Thermal Barrier is likely to occur only if both seal injection and component cooling flow was lost for a period of time. Since CC flow is maintained it is unlikely a steam void will be formed. As seal flow is reestablished the thermal gradient across the tubes are lessened; therefore, steam voids are unlikely to occur.

21. Given the following:

- The plant had been operating at 100% power for 100 days prior to shutdown.
- The plant was shutdown at 1800 on 10-11-05 due to excessive RCS leakage.
- Current plant parameters are:
  - RCS temperature (WR) is 122ºF.
  - RHR suction temperature is 123ºF.
  - Reactor vessel level is 13%.

It is now 1300 on 10-31-05, and a total loss of RHR has just occurred.

Using the attached reference, how long will it take for the RCS to reach saturation temperature?

A. 21-23 minutes.

B. 25-27 minutes.

C. 28-30 minutes.

D. 36-38 minutes.

Answer: C
COGNITIVE LEVEL:

3-SPR - Solve the problem of when saturation conditions are expected to exist using the graph for time following shutdown.

K/A:

0252AA2.05 – Ability to determine and interpret the following as they apply to the Loss of Residual Heat Removal System: Limitations on LPI flow and temperature rates of change.

OBJECTIVE:

RO4-03-LPD18.005 - **SUMMARIZE** the subsequent operator actions In accordance A-RHR-34C, “Loss of RHR Cooling While Operating In A Reduced Inventory Condition”, that are necessary to respond to the following:
1. RHR Pump Trip

REFERENCES:

A-RHR-34C, Rev. I, FIGURE 1

PROVIDED REFERENCE: A-RHR-34C FIGURE 1

SOURCE:

KNPP Bank Question 0340020401K01 001

JUSTIFICATION:

The time after shutdown comes out to 19 days, 19 hours - 475 hours. Using the first formula the calcualtion becomes: (212-122)/72 x 23.5 = 29.375 minutes.

A. Time is based on using the Initial Temperature 140ºF curve at 432 hours (based on missing a day & forgetting to add the additional 19 hours).

B: Time is based on calculating using Initial Temperature 122ºF at 432 hours using the first equation (based on missing a day & forgetting to add the additional 19 hours).

D: Time is based on using the Initial Temperature 100ºF curve at 475 hours.

22. Given the following:

- The plant is at 100% power when the turbine tripped.
- The reactor failed to trip.
- The attempt to remove power to the Rod Drive MG Sets also failed.

- The Reactor Operator has performed the following actions of FR-S.1:
  Charging flow has been raised to 90 gpm.
  Letdown flow is 40 gpm.
  Boric Acid Transfer Pump A was the only BAT pump to start.
  CVC-440, Emergency Boration Valve, has been opened.
  Boration flow has been verified at 40 gpm.

If LT-112, VCT level transmitter, fails to 50%, what is expected to occur in the CVCS system if NO operator action is taken?

A. The charging pump suction will automatically shift over from the VCT to the RWST when V

B. Auto make-up will occur and should be sufficient to restore VCT level with normal letdown a

C. VCT level will continue to drop until the VCT is empty and the charging pumps become gas

D. The spring-loaded check valves from the Charging Pumps suction to the VCT open as suct

Answer: C
COGNITIVE LEVEL:

2-RI - Recognizing interaction of VCT level with failed instrument and change in charging/letdown flow.

K/A:

029EA1.02 – Ability to operate and/or monitor the following as they apply to a ATWS: Charging pump suction valves from RWST operating switch.

OBJECTIVE:

RO2-05-LP035.004 - **DESCRIBE** the operation, automatic actions, interlocks, instrumentation, controls and alarms associated with the Chemical and Volume Control System.

REFERENCES:

A-CVC-35C, Rev. ORIG, 3.1
FR-S.1, Rev. S, Steps 5 through 7
System Description SD-35, Rev. 2, 3.3.4, page 25 (relief valves)

SOURCE:

Modified RO2-05-LP035.004 028

Changed conditions to preclude SI signal generation be considered.
Provided specific actions that have been accomplished in procedure.
Identified failed instrument.
Changed correct answer to remove "NOT" condition and to provide direct explicit answer.
Choice "D" changed to remove non-plausible condition of CVC-7 apparently operating automatically.

JUSTIFICATION:

LT-112 provides for auto makeup actuation for the VCT. This occurs at 17% lowering level and stops at 28% rising level. Also the channel prevents VCT overfill condition by diverting letdown flow from VCT at 78% rising level. With the level failed, auto makeup would not occur. With charging flow exceeding letdown flow, and when added to boration flow, remains less than than total charging flow, VCT level would drop until the tank was empty.

A: Whatever to the RWST does normally occur on lowering VCT level at 5%. However this requires both level channels LT-112 and LT-141 to be less than 40%.

B: This will not occur since the VCT level channel controlling makeup is failed. This is plausible since when auto makeup occurs from LT-141, then an additional flow of up to 60 gpm from Reactor Makeup Water could be expected. This would exceed listed charging.

D: This will not occur as suction pressure will decrease as VCT level lowers. It is plausible that with the addition of boration flow, VCT level could increase and LT-112 would not cause letdown to divert (but LT-141 would provide this function). As the VCT filled, suction pressure at the suction would increase and might exceed the 75 psid value.

23. Given the following:

- A tube rupture is diagnosed on SG A.

Which of the following identifies an automatic action for the radiation monitors designed to sense this condition, and the reason for this response?

Action	Reason
A. Closes Blowdown Isolation valves	Provides the capability to maintain narrow range level in the SG
B. Aligns condenser air ejector to Turbine Building	Prevents spread of contamination to the environment
C. Closes SG Sample Isolation valves	Allows SG activity to be processed through SGBT Ion Exchangers
D. Isolates humidification steam to the Office/Warehouse Annex	Prevents the spread of contamination outside the Turbine Building

Answer: D
COGNITIVE LEVEL:

1-P - Identifying the rad monitor auto action and the associated purpose for this action..

K/A:

038EK3.04 – Knowledge of the reasons for the following responses as they apply to the SGTR: Automatic actions provided by each PRM.

OBJECTIVE:

RO4-04-LP028.002 - **SUMMARIZE** the purposes or basis for the following items as they relate to E-3 “Steam Generator Tube Rupture”.
e. Continuous Actions Steps

RO4-04-LP028.002 - **DESCRIBE** the RADIATION MONITORING System to include the following in the description;
2. Function/purpose, design basis, operating characteristics, and physical location as appropriate for the following major components:
   a. Process Radiation Monitors.

RO4-04-LP028.004 - **DESCRIBE** the operation, automatic actions, interlocks, instrumentation, controls and alarms associated with the

RADIATION MONITORING System and the following major system components:
2. Process Radiation Monitors.

REFERENCES:

A-RM-45, Rev. AO, 3.6 and 3.10.
BKG E-3, Rev. D, 3.1.1 and STEP 3, pages 29-30

SOURCE:

New

JUSTIFICATION:

A: Blowdown isolation valves are closed, but the reason is NOT to maintain SG level. The pressure difference between the primary and secondary will cause flow into the SG and level will rise (until pressures are equalized). Level is a concern to ensure that the SG does not depressurize (Step 4). This is assured by maintaining feed flow until NR level is attained.

B: The air ejector is routed to the Aux Building Vent for cleanup and monitoring, not to Turbine Bldg. The reason for routing is correct.

C: The S/G sample isolation valves are closed by the signal. The reason is not correct. The SG Blowdown Treatment System is primarily used for cleanup during normal operation. There is also an alignment that provides a path to hold up or monitor tanks for SG blowdown if SG tube leakage (Secondary activity) is detected. The SGBT Ion Exchangers may be used during this lineup. It is not used for SGTR.

24. Given the following:

- The plant is at 100% power with all control systems in automatic.
- Steam Generator B Steam Flow is selected to 474/BLUE channel.
- Main Steam Pressure transmitter, PT-478/BLUE channel fails low (0 psig).

What will happen if NO operator action is taken INITIALLY, and what are the subsequent operator actions?

A. Initially, feed flow to SG B will be higher.
The operator will leave FW-7B, Main Feedwater Flow Control Valve, in AUTO and verify that steam flow and feed flow stabilize with level above program value.

B. Initially, feed flow to SG B will be higher.
The operator must take manual control of FW-7B, Main Feedwater Flow Control Valve, and balance steam flow and feed flow to restore SG B level to program value.

C. Initially, feed flow to SG B will be reduced.
The operator will leave FW-7B, Main Feedwater Flow Control Valve, in AUTO and verify that steam flow and feed flow stabilize with level below program value.

D. Initially, feed flow to SG B will be reduced.
The operator must take manual control of FW-7B, Main Feedwater Flow Control Valve, and balance steam flow and feed flow to restore SG B level to program value.

Answer: D
COGNITIVE LEVEL:

3-PEO - Predict the outcome of the controlling instrument failure resulting in loss of feed flow (closing FW-7B).

K/A:

054AK3.02 – Knowledge of the reasons for the following responses as they apply to the Loss of Main Feedwater (MFW): Matching of feedwater and steam flows.

OBJECTIVE:

RO4-04-LPD02.006 - In accordance with A-FW-05A, “Abnormal Feedwater System Operation,” **SUMMARIZE** the subsequent operator actions that are necessary to respond to S/G Level Alarms.

RO4-04-SED02.006 - In accordance with Annunciator Response Manuals, take corrective actions for various control instrumentation failures IAW A-FW-05A, “Abnormal Feedwater System Operation” and A-MI-87, “Bistable

Tripping for Failed Reactor Protection or Safeguards Instrumentation",
RESPOND to a the following:
b. Indications of a controlling SG Steam Flow channel failure.

REFERENCES:

A-FW-05A, Rev. S, 4.4.1
47061-F, Rev. D, Comments: 2

SOURCE:

New

JUSTIFICATION:

When the pressure channel feeding the controlling channel of steam flow fails low, the affected steam flow channel fails low. This results in an error signal between feed flow (initially unchanged) and steam flow. The error signal will close the associated FW-7A/B valve. Procedure directs taking manual control and throttling FW-7A/B to maintain SG level at programmed level.

A: The feed flow will be significantly reduced as FW-7B responds to the error between steam flow (lower) and feed flow. The operator is directed to take manual control. The level dominant SGWLC may stabilize with level BELOW program value prior to reaching the level trip setpoint, but this is NOT procedurally directed.

B: The feed flow will be significantly reduced FW-7B responds to the error between steam flow (lower) and feed flow. The action described is correct.

C: It is not procedurally directed, or administratively allowed, to leave the control in AUTO. The level dominant SGWLC may stabilize with level BELOW program value prior to reaching the level trip setpoint.

25. How do FW-101A/CV-31112 and FW-101B/CV-31029, Feedwater Pump A & B Recirculation Flow Control Valves, fail for each of the given conditions?

	Loss of Air	Station Blackout / Loss of All Electrical Power
A.	OPEN	OPEN
B.	OPEN	CLOSED
C.	CLOSED	OPEN
D.	CLOSED	CLOSED

Answer: A
COGNITIVE LEVEL:

1-F - Fact of how the valve responds to loss of power and loss of motive force (air).

K/A:

055EA2.01 – Ability to determine and interpret the following as they apply to a Station Blackout: Existing valve positioning on a loss of instrument air system.

OBJECTIVE:

RO2-02-LP05A.004 - **DESCRIBE** the operation, automatic actions, interlocks, instrumentation, controls and alarms associated with the Main Feedwater System and the following major components
c. FW Pumps Recirculation Control Valves.

RO2-02-LP05A.001 - **DESCRIBE** the Main Feedwater System to include the following;
2. Function/purpose, design basis, operating characteristics, and physical location as appropriate for the following major components:
   c. FW Pumps Recirculation Control Valves.

REFERENCES:

E-AS-01, Rev . O, 3.1 Automatic actions
E-1625, Rev. S

SOURCE:

KNPP Bank LRC-04-LP102 002

JUSTIFICATION:

B: While no Feedwater air valves have this failure mode, it is a possible failure mode.

C: Condensate Dump Control will fail closed on a loss of air, but fails open on a loss of power to solenoid.

D: FW-7A and FW-7B, S/G A & B Main Feedwater Flow CVs fail this way.

26. Given the following:

- RCS is solid.
- RCS temperature is 185ºF.
- RHR Train A is operating to maintain RCS temperature.
- RHR letdown is aligned through the RHR/CVCS spectacle flange.
- LD-10, Letdown Control Pressure, is in AUTO set to 325 psig.
- CC-302, Letdown Cont Outl Temp Controller, is in AUTO set to 120ºF.

How will VCT temperature and VCT pressure change over the next 30 minutes if CC-302 Controller output signal fails to 100% and <u>NO</u> operator action was taken?

	<u>VCT Temperature</u>	<u>VCT Pressure</u>
A.	Increase	Increase
B.	Increase	Decrease
C.	Decrease	Increase
D.	Decrease	Decrease

Answer: D
COGNITIVE LEVEL:

2-DR - Describing the relationship between liquid temperature and gas solubility, after determining failure mechanism.

K/A:

004K5.30 – Knowledge of the operational implications of the following concepts as they apply to the CVCS: Relationship between temperature and pressure in CVCS components during solid plant operation.

OBJECTIVE:

RO2-05-LP035.002 - **DESCRIBE** the Chemical And Volume Control System to include the following in the description;

2. Function/purpose, design basis, operating characteristics, and physical location as appropriate for the following major components:
   a. Letdown Subsystem
      7. Letdown Heat Exchanger
      15. Volume Control Tank (VCT)

O-FND-LP 1.4.2, rev. ORIG, 2.2.3 Describe the effects of temperature and pressure on the density of a liquid.

REFERENCES:

E-2026, Rev. Q
System Description 31, Rev. 4, 3.6.2.
System Description 35, Rev. 2, 3.2.7.

WEC Practical Chemistry for Nuclear Power Plant Operation, Chapter 1, Solubility, Pages 1-7 & 1-8.

SOURCE:

New

JUSTIFICATION:

100% output from the Controller fully opens CC-302. This will result in decreasing letdown coolant temperature, and VCT temperature. The lower temperature coolant being sprayed into the VCT will be able to absorb more gas in solution, and therefore, VCT pressure will lower.

A: This would occur if CC-302 went closed. The warmer letdown coolant would raise VCT temperture and a rise in the coolant temperature would cause more gas to come out of solution.

B: Rise in VCT temperature would occur only if CC-302 went close. The solubility of the gases in the coolant would decrease as the temperature of the coolant rises.

C: The VCT temperature does decrease since cooling increases for the coolant in the Letdown Hx. The solubility of gas in the coolant increases as the temperature of the coolant decreases, and more gas molecules dissolved in the coolant results in a reduction of the VCT pressure.

27. Given the following:

- A LOCA occurred forty minutes ago.
- ES-1.3, Transfer To Containment Sump Recirculation, has been completed with NO abnormalities encountered, other than those identified below.
- RCS pressure is 150 psig.
- Containment pressure peaked at 28 psig, and is currently 7 psig and slowly increasing.
- Only Containment Fan Coil Unit A is running.

What would happen if RHR Pump A were to trip?

A. SI Pump A would cavitate.

B. RHR Pump B would reach runout conditions.

C. Containment Spray flow would drop to ZERO.

D. SI Cold Leg Injection flow would drop to ZERO.

Answer: C
COGNITIVE LEVEL:

3-PEO - Predict the outcome of the trip of the Standby Train RHR Pump on the post-transfer to sump recirculation.

K/A:
005K3.06 – Knowledge of the effect that a loss or malfunction of the RHRS will have on the following: CSS.

OBJECTIVE:

RO4-04-LP021.001 - **Discuss** the following items as they relate to ES-1.3, Transfer to Containment Sump Recirculation.
2. High Level Action Summary Steps.

RO2-05-LP033.001 - **DESCRIBE** the Safety Injection System to include the following in the description;
3. Interfaces with the following:
   b. Residual Heat Removal System
   e. Containment Sump
   f. Internal Containment Spray System

REFERENCES:

ES-1.3, Rev. Z, Steps 11-13; 20-27

SOURCE:

New

JUSTIFICATION:

ES-1.3 alignment has Train B set up for RHR/SI recirculation. At 4% RWST level the injecting train (A) is stopped and then aligned for standby operation in recirculation mode. Finally the need for Containment Spray is assessed, and if required the standby train (A) is aligned to provide suction source to the same train ICS Pump. Thus the trip of the RHR Pump A results in loss of suction to ICS Pump A and loss of containment spray flow.

A: This would only occur if Train B ECCS pumps could not be aligned for recirculation OR failed to deliver recirculation flow. Then Train A would be aligned and started in recirculation mode.

B: This does not occur since the Trains remain separate during recirculation mode operation. This could occur in shutdown cooling mode if both trains were intertied for cooling.

D: This would not occur since Train B is normally aligned for recirculation flow, and RHR Pump B and SI Pump B are unaffected. Train A could be aligned for recirculation flow if Train B flow was not established.

28. What is the MAXIMUM temperature that would be observed for the PRT assuming saturation conditions are maintained within the PRT?

A. 120ºF.

B. 239ºF.

C. 338ºF.

D. 556ºF.

Answer: C
COGNITIVE LEVEL:

1-F - Knowledge of the rupture disc relief setpoint for the PRT and use of steam tables to determine saturation conditions.

K/A:

007A1.03 – Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the PRTS controls including: Monitoring quench tank temperature.

OBJECTIVE:

RO2-01-LP36B.001 - **DESCRIBE** the function/purpose, design basis, and operating characteristics of the PRZR and PRT Systems.

2. Function/purpose, design basis, operating characteristics, and physical location as appropriate for the following major components:
   h. Pressurizer Relief Tank (PRT)

RO4-02-LPD09.004 - Given a parameter or condition associated with a Precaution or Limitation in the following procedures, **DETERMINE** any associated operator action that may be required during a plant heatup:

4. N-RC-36B, Pressurizer Relief Tank Operation

REFERENCES:

N-RC-36B, Rev. Q, 2.3.1.

OPERXK-100-10, Rev. BQ

Steam Tables

SOURCE:

New

JUSTIFICATION:

The PRT has a rupture disc that operates at 100 psig. This is the maximum pressure for the PRT and the saturation temperature for 115 psia in Steam Tables is 338.03ºF. (At 110 psia, the saturation temperature is 334.79ºF and at 120 psia, the saturation temperature is 341.27ºF)

A: This is the maximum water temperature for normal operation of the PRT.

B: This corresponds to the saturation temperature at 10 psig (25 psia) when MG(R)-549, Przr Re3lief Tank Vent Isol. gets an automatic close signal.

D: This corresponds to the saturation temperature at 1100 psig, which is the probable PRZR vapor temperature when the PRT rupture discs blows.

29. How is the OPΔT (Overpower delta-temperature) reactor trip setpoint calculation affected by changes in given parameters?

The setpoint for the OPΔT trip is REDUCED...

A. when Pressurizer pressure is below the full power value for pressure.

B. when Tave exceeds the full power value by more than 1ºF.

C. more when Tave is decreasing than when Tave is increasing.

D. more when exceeding the upper delta-flux limit than when exceeding the lower delta-flux limit.

Answer: B
COGNITIVE LEVEL:

1-F - Knowlege of the factors affecting the setpoints for OPΔT reactor trip protection.

K/A:

012A1.01 – Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RPS controls including: Trip setpoint adjustment.

OBJECTIVE:

RO2-05-LP471.004 - **EXPLAIN** the LCO operation, applicability and action requirements for each of the following Technical Specifications associated with the Reactor Protection Day 1 System:
2. T.S. 2.3.a.3A + B.

REFERENCES:

KNPP Technical Specifications, Amend. 172, 2.3.a.3.B page TS 2.3-3
TRM 2.1, Rev. 6, COLR Cycle 27,Rev. 1, 2.10.

SOURCE:

MODIFIED
KNPP Bank RO2-05-LP471.004 004

Changed layout of question to directly ask which item affects the trip setpoint.
Asked to identify the selection that reduces the setpoint value.

Changed two selections to eliminate setpoint increase.
Correct answer changed since OPΔT setpoint is NOT affected by ΔI. (Input set to ZERO.)
Changed one selection for pressure to deviation form Tave which is also an input.

JUSTIFICATION:

A: Pressurizer input is NOT a factor in the OPΔT setpoint calculation but is for OTΔT setpoint calculation and would affect the setpoint as described.

C: RCS Tave change does affect the setpoint for OPΔT, but in the opposite manner. The factor reduces the OPΔT setpoint on increasing Tave, but has no affect (= 0) for decreasing temperature.

D: ΔI factor for OPΔT setpoint calculation is set to ZERO for all values of ΔI.

30. What is the response of the Heater Drain System as the Heater Drain Tank pressure approaches and exceeds the pressure of Moisture Separator Reheater 1A, with the Moisture Separator Reheater level normal?

A. HD-1A1, MSR 1A to Htr Drain Tank Stop Check CV, CLOSES, and then both BS-206A & B, Heater Drain Tank Steam Dump to Cdsr, modulate OPEN.

B. HD-7A1, MSR 1A Dump to Condenser, OPENS, and HD-1A1, MSR 1A to Htr Drain Tank Stop Check CV, CLOSES.

C. HD-7A1, MSR 1A Dump to Condenser, CLOSES, and then both BS-206A & B, Heater Drain Tank Steam Dump to Cdsr, modulate OPEN.

D. MS-201A1, Reheater A1 Steam Control Valve, CLOSES and then both BS-206A & B, Heater Drain Tank Steam Dump to Cdsr, modulate OPEN.

Answer: A
COGNITIVE LEVEL:

2-RI - interacting systems response to specified condition (Heater Drain and MSRs (MRSS).

K/A:

012A1.01 – Ability to monitor automatic operation of the MRSS, including: Isolation of the MRSS.

OBJECTIVE:

RO2-02-LP011.004 - **DESCRIBE** the operation, automatic actions, interlocks, instrumentation, controls and alarms associated with the HD and BS Systems and the following major components:
3. MSRs.
5. Heater Drain Tank

REFERENCES:

Alarm Response 47062-T, Rev. B, NOTE.
System Description 11 & 12, Rev. 1, 3.2.2 pages 10-11.
E-2070, Rev. J.

SOURCE:

KNPP Bank Question RO2-02-LP011.004 003

JUSTIFICATION:

B: If the MSR level rises then its level is controlled by opening the dump valve HD-7A1. The sequence of operation for response to high dP is incorrect. HD-1A1 closes first (at 2 psid), and then the auctioneered high value is used to close BS-206A & B (at 4 psid). The level valve responds as MSR level rises.

C: If the MSR level rises then its level is controlled by opening the dump valve HD-7A1. The sequence of operation for response to high dP is incorrect. HD-1A1 closes first (at 2 psid), and then the auctioneered high value is used to close BS-206A & B (at 4 psid). The level valve responds as MSR level rises.

D: MS-201A1 is the steam supply to the reheater coil and is controlled by the operating cam of the Timing Controller. The auctioneered high value is used to close BS-206A & B (at 4 psid).

31. Given the following:

- The plant is at 100% power.
- During troubleshooting of an electrical problem associated with the Auxiliary Feedwater Actuation circuitry, both motor-driven AFW Pumps start.
- No other equipment is affected.

How is feedwater flow to the SGs affected?

Main Feedwater flow will...

A. lower due to SG swell.

B. lower due to rising SG level.

C. remain constant due to the higher head of the Main Feed Pumps.

D. remain constant due lack of input from AFW System into SG Water Level Control System.

Answer: B
COGNITIVE LEVEL:

2-DR - Describe the relationship between AFW system and Main Feedwater system and response to inadvertent operation of AFW.

K/A:

059K1.02 – Knowledge of the physical connections and/or cause-effect relationships between the MFW System and the following systems: AFW System.

OBJECTIVE:

RO2-02-LP05A.004 - **DESCRIBE** the operation, automatic actions, interlocks, instrumentation, controls and alarms associated with the Main Feedwater System and the following major components:
f. Main FW Control Valves.

RO2-02-LP05B.001 - **DESCRIBE** the function/purpose, design basis, and operating characteristics of the Auxiliary Feedwater System.

REFERENCES:

System Description 05A, Rev. 4, 3.13.5.
System Description 05B, Rev. 2, 3.3.1 & 3.7

SOURCE:

Point Beach 9/2003 NRC Exam (INPO Bank)

JUSTIFICATION:

The flow control valves (AFW-2A and AFW-2B) for each motor-driven pump are normally fully open with the system in standby. Thus if the pumps start, full AFW flow will be delivered through the header.

A: The AFW water will be significantly lower in temperature than the feedwater flow to the SGs (90ºF vs. 450ºF), lowering the feed temperature to SGs. This will result in shrink, if any affect, in the SG.

C: The discharge head for the Main FW Pumps is 1053 psig at 292 psig suction. The AFW Pumps design head is approximately 1239 psig. Therefore, AFW has adequate head to provide flow to SGs with FW Pumps running.

D: While AFW does NOT have any inputs to SGWLC, the addition of mass from the AFW will cause level in SGs to rise. The SG level does provide input to SGWLC.

32. Given the following:

- The plant startup is in progress.
- Reactor power is 1.4%.
- FW-07A & B, S/G Main Feed valves, AND FW-10A & B, S/G Bypass Feed valves, indicate closed.
- Feedwater Pump A has just been started.
- SG B level begins to slowly rise.

What action is required?

A. Direct the NAO to locally close FW-9B, Feedwater Main Control Valve 1B Bypass Valve Inlet.

B. Cycle FW-10B, S/G B Bypass, fully open and closed.

C. Close FW-12B, S/G B Feedwater Isolation Valve.

D. Stop Feedwater Pump A.

Answer: C
COGNITIVE LEVEL:

1-P - Procedural directions for unexpected SG rise after FW Pump started.

K/A:
059 2.2.1 – Main Feedwater (MFW) System: Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity.

OBJECTIVE:

RO4-02-LPD14.004 - Given that a power and load increase is planned, **SUMMARIZE** the following procedures including any applicable Figures:
- N-FW-05A.

REFERENCES:

N-FW-05A, Rev. AD, 4.1.4.j and preceding NOTE.

SOURCE:

New

JUSTIFICATION:

This step is a recent (5/31/2005) addition to N-FW-05A, directing action to operators after noted problems during the previous plant startup.

A: This action is not correct in that it does not provide for expeditious response and provides only for isolation of one of the 2 valves that could be leaking.

B: This action will only exacerbate the SG level rise as flow will be maximized through FW-10B during the stroking. The stroking process with FW-12B closed does occur earlier prior to start of the FW Pump as part of checking operability of the valves. It also addresses only one of the 2 flowpaths.

D: Stopping the FW Pump would stop the leakby but is not procedurally directed. It is not prudent since SG A does not exhibit any leakage problem.

33. Given the following:

- The plant is at 100% power.
- The LOCAL/REMOTE Switch for AFW Pump A has been placed in LOCAL at the Dedicated Shutdown Panel.

Which of the following identifies the expected response concerning AFW Pump A for the given condition?

A. AFW Pump A will start on low level in either of the S/Gs.

B. A Load Shed signal will NOT affect operation of the AFW Pump A.

C. AFW Pump A indications are now activated at both the main control board and the DSP.

D. If the AFW Pump A Control Switch in the Control Room is placed to PULLOUT, the pump CANNOT be started from the DSP.

Answer: B
COGNITIVE LEVEL:

1-F - Factual information concerning the breaker operation (power supply) characteristics for the AFW Pump.

K/A:

061K2.02 – Knowledge of bus power supplies to the following: AFW electric driven pumps.

OBJECTIVE:

RO2-02-LP05B.004 - **DESCRIBE** the operation, automatic actions, interlocks, instrumentation, controls and alarms associated with the Auxiliary Feedwater System and the following major system components:
2. Auxiliary Feedwater Pump Motors (2).

REFERENCES:

E-1602, Rev. AZ
System Description 05B, Rev. 2, 3.14

SOURCE:

KNPP Bank Question RO2-02-LP05B.004 008

JUSTIFICATION:

A: With the switch in REMOTE, the pump will start on all automatic start signals. Placing the switch to LOCAL removes the auto starts.

C: Placing the switch in LOCAL removes Control Room (CR) indication for the pump. Placing the switch in REMOTE restores CR indication, and will remove indication at the DSP.

D: Placing the Pump in PULLOUT in the Control Room will prevent start of the Pump for automatic start signals. It does not affect operation of the Pump using the DSP control switch while in LOCAL.

34. Given the following:

- The plant is at 100% power.
- AFW Pump B is running for a surveillance test in progress.
- Annunciator 47061-M, AFW PUMP B LOW OIL PRESS, alarms.

What is the expected operator response for this condition?

A. Trip AFW Pump B, and go to N-FW-05B, Auxiliary Feedwater System.

B. Trip AFW Pump B and go to A-FW-05B, Abnormal Auxiliary Feedwater System Operation.

C. Verify the Auxiliary Lube Oil Pump is running, and go to N-FW-05B, Auxiliary Feedwater System.

D. Verify the Auxiliary Lube Oil Pump is running, and go to A-FW-05B, Abnormal Auxiliary Feedwater System Operation.

Answer: A and D
COGNITIVE LEVEL:

1-P - Procedure symptoms or entry conditions for the AFW.

K/A:

061 2.4.4 – Auxiliary / Emergency Feedwater (AFW) System: Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures.

OBJECTIVE:

RO2-02-LP05B.005 - **EXPLAIN** the purpose of the following procedures used to govern the normal, abnormal, and emergency operation of the Auxiliary Feedwater System:
2. A-FW-05B Abnormal Auxiliary Feedwater System Operation.

RO4-03-LPD11.001 - **DESCRIBE** the purpose Immediate operator actions, and automatic actions associated with the following procedures:
b. A-FW-05B, Abnormal Aux Feedwater System Operation.

REFERENCES:

47061-M, Rev. A, Recommended Actions & Comments
A-FW-05B, Rev. AL, 2.1 & 3.3

SOURCE:

New

JUSTIFICATION:

The annunciator alarms on less than 10 psig oil pressure with the pump breaker closed. The Aux Lube Oil Pump for the AFW Pump should also auto start at this pressure. The AFW Pump will trip on lowering lube oil pressure at 4 psig.

A: There is no need to trip the pump since the alarm comes in at 10 psig and the pump should trip at 4 psig. N-FW-05B does not provide the proper guidance in this situation.

B: There is no need to trip the pump since the alarm comes in at 10 psig and the pump should trip at 4 psig. A-FW-05B is the proper procedure for guidance in this situation.

C: The action to verify the Aux LO Pump running is correct; however, N-FW-05B does not provide the proper guidance in this situation.

35. Given the following:

- Plant is at 100% power.
- Radiation monitors R-15, Condenser Air Ejector Gas, and R-19, SG Blowdown Liquid Sample, are in HIGH alarm.
- Chemistry has been requested to sample SG blowdown.

What operator action is taken to allow the SG Sample Isolation valves to be opened?

A. Place the R-15 and R-19 Keyswitch to OFF and maintain in OFF.

B. Place the R-15 and R-19 Keyswitch momentarily to OFF and then back to ON.

C. Place the R-15 and R-19 Keyswitch to KEYPAD and maintain in KEYPAD.

D. Depress the RESET keypad for both R-15 and R-19 monitors.

Answer: C
COGNITIVE LEVEL:

1-I - Interlock for system response

K/A:

073A4.02 – Ability to manually operate and/or monitor in the control room: Radiation monitoring system control panel.

OBJECTIVE:

RO2-01-LP045.004 - DESCRIBE the operation, automatic actions, interlocks, instrumentation, controls and alarms associated with the RADIATION MONITORING System and the following major system components:
2. Process Radiation Monitors

REFERENCES:

N-RM-45, Rev. AQ, 4.3.13 & 4.3.17
E-3748, Rev. B

SOURCE:

Old Facility Part B Bank 000037104A01

JUSTIFICATION:

A: Placing the Keyswitch to OFF results in actuation of automatic functions for the radiation monitor.

B: As in "A" OFF will result in automatic actuation, and when taken back to ON, if the HIGH Radiation signal is in, actuation signal is generated again.

D: Depressing RESET will momentarily reset the HIGH and ALERT input logic but will re-actuate when RESET is released, if the HIGH Radiation signal is still in. High rad signal from either radiation monitor signal will prevent opening of sample valves.

36. Given the following:

- The plant is at 100% power.
- Discharge of Waste Gas Decay Tank C is in progress.
- A HIGH alarm is received on R-13, Aux Bldg Vent Exhaust Monitor.

What changes in indications on the control room control boards would the Control Operator observe to verify the automatic actions for the radiation monitor functioned correctly?

A. Train A Zone SV Exhaust Fan red light lit.
Train A Safeguards Area Fan Coil Units red lights lit.
Both Trains of Aux Building Supply and Exhaust Fans green lights lit.
Train A Zone SV Dampers green lights lit.
AS-31, Containment Air Sample Return to CNTMT for R-11/12, red light lit, and AS-35, Containment Air Sample Return to Aux Vent for R-11/12, red light off.

B. WG-36, Gas Decay Tanks to Plant Vent, green light lit.
Train A Zone SV Exhaust Fan red light lit.
Both Trains of Aux Building Supply and Exhaust Fans red lights lit.
Train A Steam Exclusion Dampers green lights lit.
AS-31, Containment Air Sample Return to CNTMT for R-11/12, red light lit, and AS-35, Containment Air Sample Return to Aux Vent for R-11/12, red light off.

C. Train A Zone SV Exhaust Fan red light lit.
Train A Safeguards Area Fan Coil Units red lights lit.
Train A Aux Building Supply and Exhaust Fans green lights lit and Train B Aux Building Supply and Exhaust Fans red lights lit.
Train A Zone SV Dampers green lights lit.
SFP Charcoal Filter Units Bypass Dampers green lights lit and SFP Charcoal Filter Units Outlet Dampers red lights lit.

D. WG-36, Gas Decay Tanks to Plant Vent, green light lit.
Train A Zone SV Exhaust Fan red light lit.
Train A Aux Building Supply and Exhaust Fans green lights lit and Train B Aux Building Supply and Exhaust Fans red lights lit.
Train A Steam Exclusion Dampers green lights lit.
SFP Charcoal Filter Units Bypass Dampers green lights lit and SFP Charcoal Filter Units Outlet Dampers red lights lit.

Answer: A
COGNITIVE LEVEL:

2-DR - Describe relationship between the failure of the rad monitor and indications that have changed in the control room due to the failure. Involves multiple systems and recognition of which indications are appropriate for control room.

K/A:

2.3.11 – Radiological Controls; Ability to control radiation releases.

OBJECTIVE:

RO2-01-LP045.004: DESCRIBE the operation, automatic actions, interlocks, instrumentation, controls and alarms associated with the RADIATION MONITORING System and the following major system components:
2. Process Radiation Monitors.

RO4-03-LPD17.001: DESCRIBE the purpose, symptoms, immediate operator actions, and automatic actions associated with the following procedures::
a. A-RM-45, Abnormal Radiation Monitoring System

REFERENCES:

A-RM-45, Rev. AO, 3.4
E-1616, Rev. U (Aux Bldg SV units)
E-1623, Rev. W (Aux Bldg Vent, SFP Charcoal Filter Dampers & SFGDS Fan Coil Units)
E-2048, Rev. N (GDT Release valve)

SOURCE:

New

JUSTIFICATION:

B: WG-36 has local indication only at the Waste Disposal Panel. The Steam Exclusion Dampers are not expected to change position, but is plausible since a detected steam exclusion actuation signal will also result in Aux Bldg SV actuation.

C: The high radiation on either R-13 or R-14 will trip both trains of Aux Building Ventilation Supply and Exhaust fans. The SFP Charcoal Filter Unit dampers have local indication only.

D: WG-36 has local indication only at the Waste Disposal Panel. The high radiation on either R-13 or R-14 will trip both trains of Aux Building Ventilation Supply and Exhaust fans. The SFP Charcoal Filter Unit dampers have local indication only.

37.

10CFR20 limits the radiation exposure (dose) to a qualified radiation worker to _______ per year.

Kewaunee Power Station limits the radiation dose to a qualified radiation worker to _______ per year WITHOUT special authorization.

	10CFR20 limit	KPS limit
A.	3000 mrem	1200 mrem
B.	3000 mrem	1800 mrem
C.	5000 mrem	2000 mrem
D.	5000 mrem	3000 mrem

Answer: C
COGNITIVE LEVEL:

1-F - Identify NRC and Plant administrative exposure limits.

K/A:

2.3.1 – Radiological Controls: Knowledge of 10 CFR: 20 and related facility radiation control requirements.

OBJECTIVE:

NGA02F001H, Rev. 4

D. Upon completion of this section, students should be aware of the federal and plant administrative limits on radiation dose.

State the federal radiation dose limits for TEDE, skin, extremities, lens of the eye, and declared pregnant worker. [RWT16]

State the NMC administrative limits/guidelines for radiation dose. [RWT18]

REFERENCES:

10CFR20.1201 (a)(1)(i)
NAD-01.11, Rev. L, 5.1.1 & 5.2.5

SOURCE:

Bank

JUSTIFICATION:

A: 3000 mrem corresponds to the second line administrative limit (the limit employed after first level of approval, at 60% of the NRC normal limits). Previous 10CFR20 limits also used to specify 3 rem/qtr limit. 1500 mrem correspond to the typical first line limit employed at 40% of the NRC allowed limit.

B: 3000 mrem corresponds to the 2nd stage limit (the limit employed after first level of approval). Previous 10CFR20 limits also used to specify 3 rem/qtr limit. 1800 mrem correspond to the typical second line limit employed at 60% of the NRC allowed limit.

D: 5000 mrem corresponds to the correct NRC limit. 3000 mrem correspond to the typical second line limit employed at 60% of the NRC allowed limit.

38. Given the following:

- An accidental gas release has occurred in the Aux Building.
- The Derived Air Concentration (DAC) in the area where the release occurred is 4 DAC.
- It has been determined that it will take 2 hours to complete repairs working in that area.
- An ALARA evaluation has shown that overall man-rem is lower if respirators are NOT used due to time concerns.

What is the expected internal dose (CEDE) to a worker completing the repairs?

A. 8 mrem.

B. 10 mrem.

C. 20 mrem.

D. 40 mrem.

Answer: C
COGNITIVE LEVEL:

3-SPK - Use the relationship between DAC-HOURS and dose to calculate the expected exposure for a given task.

K/A:

060AK1.02 – Knowledge of the operational implications of the following concepts as they apply to Accidental Gaseous Radwaste Release: Biological effects on humans of the various types of radiation, exposure levels that are acceptable for personnel in a nuclear reactor power plant; the units used for radiation intensity measurements and for radiation exposure levels

OBJECTIVE:

NGA02L001H 52 - State the relationship among DAC, ALIs, CEDE, and TEDE (DAC and mrem/hr relationship) [RWT52]

REFERENCES:

10CFR20.1204 h.(1)

SOURCE:

New

JUSTIFICATION:

1 DAC is the equivalent of 2.5 mrem/hr internal exposure.
Therefore 4 DAC is 10 mrem/hr and the task lasts 2 hours, so total exposure is 20 mrem.

A: This is based on multiplying the number of hours given by the DAC value.

B: This is based on the correct DAC to mrem/hr value without accounting for the 2 hours.

D: This is based on the reasoning that 1 DAC-hour is 5 mrem.

39. Given the following:

- The plant is at 100% power.
- Containment pressure has risen from 0.1 psig to 1.1 following startup.
- The pressure rise has been primarily attributed to an air line leak in containment that has since been repaired.
- Preparations are in progress to reduce the Containment pressure.
- Sampling indicates Containment atmosphere radioactivity levels are acceptable for release.

What are the conditions that must be met to allow use of the 2-inch Containment Vent line for the lowering of pressure?

A. Both trains of the Auxiliary Building Ventilation System must be in operation and both trains of the Post-LOCA Vent System are used for venting.

B. Both trains of the Auxiliary Building Ventilation System must be in operation and only ONE of the trains (either A or B) of the Post-LOCA Vent System is used for venting.

C. ONE train of the Auxiliary Building Ventilation System must be in operation and only Train A of the Post-LOCA Vent System is used for venting.

D. ONE train of the Auxiliary Building Ventilation System must be in operation and only Train B of the Post-LOCA Vent System is used for venting.

Answer: D
COGNITIVE LEVEL:

1-P - Procedure steps and caution.

K/A:

2.3.9 – Radiological Controls; Knowledge of the process for performing a containment purge.

OBJECTIVE:

RO2-05-LP018.005: EXPLAIN the purpose of the following procedures used to govern the normal, abnormal, and emergency operation of the Reactor Building Ventilation System:

2. N-RBV-18B Reactor Building Ventilation System Cold Operation and Making Releases.

RO2-04-LP18C.004: DESCRIBE the operation, automatic actions, interlocks, instrumentation, controls and alarms associated with the Reactor

Building Ventilation Post LOCA H_2 Control System and the following major system components:
1. Post LOCA H_2 Containment Isolation Valves.

REFERENCES:

N-RBV-18B, Rev. AA, 2.4 & section 4.1.3.
E-2068, Rev. Z

SOURCE:

Modified RO2-04-LP18C.004 001

Changed question from asking why Train A is NOT used.

Changed selections from reasons why Train A is not used to conditions that allow use. Includes adding aspect of requirements for Aux Bldg Vent System.

JUSTIFICATION:

A: Only one train, specifically Train B of Post-LOCA Vent can be used. Procedure requires at least one train of Aux Building Ventilation be in service.

B: Only one train, specifically Train B of Post-LOCA Vent can be used.

C: Only one train, specifically Train B of Post-LOCA Vent can be used. Train A cannot be used since the Containment Isolation valves do not get an automatic close signal and Containment Integrity is required.

40. Given the following:

- The plant is at 100% power.
- RCS Wide Range Pressure channel P-420 fails low.

Which of the following describes the affect of the failure of this instrument?

A. P-420 is a non-qualified instrument, and therefore is <u>NOT</u> covered in Technical Specifications.

B. P-420 is a post-accident instrument. <u>NO</u> LCO is entered since P-420 does <u>NOT</u> have an input to any required accident monitoring functional unit.

C. P-420 provides a safeguards actuation input. Six hours is allowed to place the associated Safety Injection actuation bistables in TRIP per A-MI-87, Bistable Tripping for Failed Reactor Protection or Safeguards Instr.

D. P-420 is a post-accident instrument. In accordance with Technical Specification Table TS 3.5-6 for RVLIS indication, a 7-day LCO is entered to restore the channel to OPERABLE.

Answer: D
COGNITIVE LEVEL:

2-DR - Recognition of the relationship between P420 post-accident monitor instrument and the RVLIS instrumentation, which has the Tech Spec entry.

K/A:

2.4.3 – Ability to identify post-accident instrumentation.

OBJECTIVE:

RO2-05-LP-050.007 - **DESCRIBE** the operation, automatic actions, interlocks, instrumentation, controls and alarms associated with the Incore Instrumentation & Inadequate Core Cooling Monitor Systems and the following major system components:
4. Reactor Vessel Level Indication (RVLIS)

RO2-05-LP-050.007 - **EXPLAIN** the LCO operation, applicability and action requirements for each of the following Technical Specifications associated with the Incore Instrumentation & Inadequate Core Cooling Monitor Systems:
1. 3.5, Instrumentation System.

REFERENCES:

A-II-50, Rev. H, CAUTION at Step 4.1.1
KPS Technical Specification, Table TS 3.5-6, No. 9 pages 1and 2, Amend No. 174.

SOURCE:

New

JUSTIFICATION:

P-420 is a qualified post-accident instrument. It is indicated so in the Control Room by a GREEN label.

A: P-420 is a post-accident qualified instrument.

B: P-420 is not specifically identified in Table 3.5-6; however, RVLIS is a required post-accident indication and P-420 is one input to RVLIS. the RVLIS instruments do have LCOs associated with instrument(s) out of service.

C: Wide range pressure instruments do not provide a safeguards input. if they did, then A-MI-87 would provide direction for removal from service and delineate the 6-hour bistable tripping.

41. Given the following plant conditions:

- The plant was operating at 75% power when a reactor trip occurred.
- The crew responded to the trip by entering E-0, Reactor Trip Or Safety Injection, and transitioned to ES-0.1, Reactor Trip Response.
- Normal alarm response condition has been directed by the CRS.
- The following annunciators just now alarm:

  47033-34, TLA-14 POST TRIP LOG READY
  47055-U, HP SEAL OIL BACKUP PUMP RUNNING

In addition to acknowledging the alarms, what is the proper course of action in responding to these alarms?

A. Both alarms are unexpected alarms and are announced to the crew. The reading of the Alarm Response sheets are determined by annunciator hierarchy, with 47055-U being addressed first.

B. Both alarms are unexpected alarms and are announced to the crew. The CRS will determine the priority of response based on the current plant conditions.

C. Both alarms are expected alarms and are announced to the crew. The Alarm Response Procedures for these alarms do NOT need to be addressed.

D. Both alarms are expected alarms and as such are NOT announced to the crew. No additional response is required.

Answer: C
COGNITIVE LEVEL:

2-DR - Describes the relationship between the expected alarm condition following a trip from power and the appropriate alarm response actions.

K/A:

2.4.45 – Emergency Procedures/Plan; Ability to prioritize and interpret the significance of each annunciator or alarm.

OBJECTIVE:

RO4-01-LPA16.008: DESCRIBE the expectations for the following as delineated in FP-OP-COO-01 “Conduct of Operations”:
1. Attachment 1, Alarm Response.

REFERENCES:

47055-U, Rev. D, Recommeded Action NOTE.
47033-34, Rev. Orig, Comments
GNP-03.30.02, Rev. A, 6.11
FP-OP-COO-01, Rev. 1, Attachment 1 ALARM RESPONSE

SOURCE:

Modified: Salem 11/2002 NRC exam (INPO Bank)

Changed question to identify similar alarms at KNPP expected following a trip.

Changed selections to reflect alarm status (expected or unexpected) and the directions for setting priority on response to the alarms.

JUSTIFICATION:

These two alarms are expected following a normal trip.
47033-34 indicates that the Post-Trip Logs are avaialbel for printing and actuates 5 minutes after the PPCS detects a reactor trip or SI.
47055-U actuates during turbine coastdown approximately when turbine speed falls below 1300 psig (turbine bearing oil header pressure falls below 12 psig)

Normal annunciators arrangement is to prioritize the alarms from the corners and edges of the alarm panel inward. This allows for ease in determining priority. Nominally the 47065-U alarm would have the higher priority but actual response is determined by the CRS based on plant conditions. Since these are expected alarms, and associated, the CRS will determine the need to address the alarm sheets.

A: These alarms are expected. However in responding 47065-U is positioned for higher priority. Actual priority is set by CRS.

B: These alarms are expected, so the crew does not need to address the alarm response sheets.

D: With normal alarm response condition, expected alarms are announced as "expected" to the crew. Any action taken for expected alarms is the decision of the CRS.

42. Given the following:

- The plant is at 100% power.
- The operator reports the following parameters for RXCP A:

#1 Seal Leakoff flow for RXCP A reads 6.2 gpm.
#1 Seal Water Outlet Temperature for RXCP A reads 195ºF.
RXCP A Bearing Water Temperature reads 200ºF.
RXCP A vibrations are 7.3 mils for the motor
RXCP A vibrations are 14.6 mils for the pump.
Component Cooling Hx Outlet Loop Temperature (PPCS) reads 115ºF.
The highest RXCP A Motor Bearing Temperature (PPCS) reads 160ºF.
RXCP A Motor Stator Temperature (PPCS) reads 125ºC.

What are the appropriate actions, and the associated reasons for the above conditions?

A. RXCP A is immediately tripped to protect the RXCP.
Reactor trip is verified, and the immediate actions of E-0, "Reactor Trip Or Safety Injection," are performed to ensure proper plant response.
CVC-207A, #1 Seal Leakoff Isol, is placed in CLOSE to prevent further damage to the seals.

B. The reactor is manually tripped to ensure the reactor is shutdown.
The immediate actions of E-0, "Reactor Trip Or Safety Injection," are performed to ensure proper plant response.
RXCP A is tripped to protect the RXCP.
PS-1A, Przr Spray Valve Loop A, is placed in MANUAL and CLOSED to maintain Przr spray flow.

C. The reactor is manually tripped to ensure the reactor is shutdown.
The immediate actions of E-0, "Reactor Trip Or Safety Injection," are performed to ensure proper plant response.
RXCPs A and B are tripped to protect the RXCPs.
PS-1A and PS-1B, Przr Spray Valves, are placed in MANUAL and CLOSED to maintain Przr spray flow.

D. Establish seal injection flow greater than 6.2 gpm to RXCP A to protect the seals.
Maintain stable plant conditions to stabilize seal leakoff flow.
Monitor RXCP parameters to ensure stabilization.
If #1 seal leakoff CANNOT be reduced below 6.0 gpm within 2 hours, then initiate a normal plant shutdown to allow an orderly RXCP A shutdown to protect the seals.

Answer: B
COGNITIVE LEVEL:

3-SPK - Solve the problem of abnormal pump parameters and determine that immediate stopping of the affected RXCP is required. Determine appropriate reasons for taken actions.

K/A:

015AK3.03 – Knowledge of the reasons for the following responses as they apply to the Reactor Coolant Pump Malfunctions: Sequence of events for manually tripping reactor and RCP as a result of an RCP malfunction.

OBJECTIVE:

RO4-03-LPD03.003 - In accordance with A-RC-36C, Abnormal RXCP Operation, **SUMMARIZE** the subsequent operator actions that are necessary to respond to the following:
1. Loss of Seal Injection

RO4-03-LPD01.002 - In accordance with A-CC-31, "Abnormal Component Cooling System Operations", **SUMMARIZE** the subsequent operator actions that are necessary to respond to the following:
j. Component Cooling HX Outlet Temperature High Annunciator.

REFERENCES:

A-RC-36C, Rev. T, Step 1.
A-CC-31, Rev. G, Step 4

SOURCE:

New

JUSTIFICATION:

With the exception of the the motor vibration all other parameters are below the value for action. The RXCP vibration limit for the motor is 5 mils (and the vibration limit for the pump is 20 mils).

A: The sequence of action is not acceptable. Procedural and Management direction is to trip the reactor first, prior to creating a situation where an automatic trip will be generated (opening the RXCP breaker). The action is also incorrect in that closing of CVC-207A is required only if #1 leakoff flow is at or above 8 gpm.

C: Both RXCPs would be required to be tripped if the CC Return temperature exceed the limit of 120ºF, since this affects both RXCPs.

D: This is the appropriate required action if none of the values exceed the required trip value. This action is taken since the #1 seal is degraded. Step 5 looks at action for #1 seal leakoff flow at or above 6.0 gpm.

43. Which of the following identifies the MAXIMUM current limit for a running RXCP, and the assumed conditions when determining this value?

A. Maximum current is 4800 amps, expected when the RXCP is operating at cold loop condition and maximum supply voltage of 4400 volts.

B. Maximum current is 1073 amps, expected when the RXCP is operating at cold loop condition and minimum bus voltage value of 3600 volts.

C. Maximum current is 976 amps, expected when the RXCP is operating at hot loop condition and maximum supply voltage of 4400 volts.

D. Maximum current is 745 amps, expected when the RXCP is operating at hot loop condition and minimum supply voltage of 3600 volts.

Answer: B
COGNITIVE LEVEL:

1-F - Recognize the limit for RXCP running current and the basis for these limit.

K/A:

003A3.02 – Ability to monitor automatic operation of the RCPS, including: Motor current.

OBJECTIVE:

RO2-01-LP36A.004 - **DESCRIBE** the operation, automatic actions, interlocks, instrumentation, controls and alarms associated with the Reactor Coolant Pump:
3. Instrumentation
   c. RXCP Ammeter

RO2-01-LP36A.005 - **EXPLAIN** the purpose of the following procedures used to govern the normal, abnormal, and emergency operation of the Reactor Coolant Pumps.
1. Normal Operating Procedures
   a. N-RC-36A Reactor Coolant Pump Operation

REFERENCES:

N-RC-36A, Rev. AG, Appendix A page 3
System Description 36, Rev. 4, 3.5.1, page 19.

SOURCE:

New

JUSTIFICATION:

A: This is the maximum starting current for a RXCP. The maximum current would be expected for the lowest voltage

C: This is the cold running current value for a RXCP. The maximum current would be expected for the lowest voltage.

D: This is the nominal running current for a RXCP under hot conditions.

44. Which of the following situations requires entry into a Technical Specification LCO Action?

A. Pressurizer Pressure Transmitter P-429 fails low while at 38% power.

B. Pressurizer level decreases to less than 17% with a reactor startup in progress.

C. Pressurizer Backup Heaters energize after a 8% load reduction.

D. Pressurizer Pressure Transmitter P-449 fails high while on RHR Cooling.

Answer: A
COGNITIVE LEVEL:

2-DR - Recognizing the relationship between failed instrument or component and the Technical Specification requirement for the instrument/component to be OPERABLE.

K/A:

022A 2.1.33 – Pressurizer Pressure Control (PZR PCS) Malfunction: Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications.

OBJECTIVE:

RO2-05-LP36C.007 - **EXPLAIN** the LCO operation, applicability and action requirements for each of the following Technical Specifications associated with the PRESSURIZER PRESSURE CONTROL System:
2. Pressurizer Pressure Channel Instrumentation Conditions for Trip

REFERENCES:

A-MI-87, Rev. R, 4.1 and 4.3
KNPP Technical Specification 3.5.b, Amend. No. 105
Table TS 3.5-2, No. 7 & No. 8, Amend. No. 137
Table TS 3.5-3, No. 1.d, Amend. No. 116

SOURCE:

KNPP Bank (ILT) Question RO2-05-LP36C.006 001

JUSTIFICATION:

This question is appropriate for an RO in that it asks to recognize the conditions whether a Technical Specification exists for the condition AND when the Technical Specification is applicable, rather than the specific actions required.

B: At 18.3% level lowering. automatic action occurs to trip the Przr heaters and isolate letdown to prevent loss of Przr level. However, there is no Technical Specification requirement for Przr level. There is a reactor trip associated with High Przr Level at 90%.

C: This is an expected response of the Przr pressure control system to the insurge occurring with the RCS heatup as load is lowered. There is no Technical Specification associated with the Backup Heaters energizing during power operations. These is a Technical Specification requirement that at least one group of Pressurizer heaters shall have an emergency power supply available when the average RCS temperature is > 350ºF.

D: This channel of Przr pressure has input to the low pressure reactor trip only. However, this function may be bypassed when below P-7 (3 of 4 Power Range Nuclear Instrument channels < 10% power AND 2 of 2 Turbine Impulse Pressure Channels < 10% power). This is the case when on RHR

45. Given the following:

- An unisolable steam break inside of containment has occurred, and the MSIVs failed to close.
- ECA-2.1, Uncontrolled Depressurization Of Both Steam Generators, actions are being performed.
- RCS pressure is 1800 psig and rising.
- The hottest RCS Hot Leg temperature is 375ºF.
- Containment pressure is 8 psig and lowering slowly after peaking at 15 psig.
- SG A level is 20% WIDE RANGE and lowering.
- SG B level is 15% WIDE RANGE and lowering.
- Feed flow to each SG has been reduced to 60 gpm by operator action.

What is the proper action for this condition?

Transition to FR-H.1, Response To Loss Of Secondary Heat Sink, and ...

A. verify feed flow established, then return to ECA-2.1 and control feed flow at 60 gpm to maintain SGs in a wet condition.

B. verify feed flow established, then return to ECA-2.1 and raise feed flow to at least 205 gpm total flow, until at least ONE SG narrow range level is at or above 15%.

C. perform the Bleed and Feed steps to establish RCS heat removal.

D. raise feed flow to at least 205 gpm total flow, until at least ONE SG narrow range level is at or above 15%.

Answer: A
COGNITIVE LEVEL:

3-SPK - Solve the problem with the correct action to be taken for condition where both SGs are faulted and have low wide-range levels.

K/A:

W/E12EK3.4 – Knowledge of the reasons for the following responses as they apply to the Uncontrolled Depressurization of all Steam Generators: RO or SRO function within the control room team as appropriate to the assigned position, in such a way that procedures are adhered to and the limitations in the facilities license and amendments are not violated.

OBJECTIVE:

RO4-04-LP014.002 - **Summarize** the purposes or basis of the following

items as they relate to ECA-2.1, Uncontrolled Depressurization of Both Steam Generators.
d. Procedure Transitions
e. All procedure steps

RO4-04-LP035.002 - **SUMMARIZE** the purposes or basis for the following items as they relate FR-H.1 “Response to Loss of Secondary Heat Sink”.
b. Cautions

REFERENCES:

FR-H.1, Rev. X, Caution Step 1 and Step 1.
ECA-2.1, Rev. S, Step 2.b. (CA)

SOURCE:

Byron NRC 10/2001 (INPO Bank)

JUSTIFICATION:

The feed flow is throttled to 60 gpm for each SG in ECA-2.1 to minimize cooldown and to keep SGs "wet" to minimize thermal shock when increasing feed flow. With Narrow Range SG levels in BOTH SGs below 15% (Adverse Ctnmt) a RED PATH exist for transition to FR-H.1. The CAUTION at Step 1 states, "If total feed flow is less than 205 gpm due to operator action, this procedure should not be performed." Also in step 1 action for feed or "feed and bleed" occurs only if RCS hottest wide range temperature is greater than 400ºF.

B: Action is inappropriate since this would exacerbate the cooldown. However this is the normal value for minimum flow for decay heat removal.

C: Action is inappropriate since the actions of FR-H.1 are NOT performed if feed flow is below 205 gpm due to operator action. Feed and bleed is normally required if SG Yarway Wide Range levels are less than 20% (Adverse Ctnmt), or Przr pressure is at or above 2335 psig due to loss of heat sink.

D: Action is inappropriate since the actions of FR-H.1 are NOT performed if feed flow is below 205 gpm due to operator action. However the given feed flow rate is the normal value for minimum flow for decay heat removal.

46. Given the following:

- A loss of off site power had occurred.
- A RAT lockout occurred.
- Bus 5 lockout occurred.
- The switchyard was energized using A-SUB-59, Restoration of Offsite Power, with exceptions as noted below.
- When the TAT was reenergized from offsite sources, Bus 6 was aligned to the TAT.
- 30 minutes later, the RAT lockout was cleared and the RAT was repowered from its offsite source.

The operator is restoring Bus 6 to its normal electrical lineup, and has just closed Breaker 1-601 RAT to Bus 6, when a SG level alarm diverts his attention.

Why should the time spent in this electrical configuration be minimized?

A. The potential fault current could overstress the switchgear and result in equipment failures.

B. The Technical Specifications requirement concerning independent transmission lines is violated.

C. 1066 East and 1066 West OCBs to open after 30 seconds resulting in a loss of power to the TAT.

D. This prevents automatic starting and loading of Diesel Generator B in the event Bus 6 voltage was degraded or lost.

Answer: A
COGNITIVE LEVEL:

1-P - Understanding the procedure CAUTION when paralleling .

K/A:

056A 2.1.32 – Loss of Offsite Power: Ability to explain and apply all system limits and precautions.

OBJECTIVE:

RO2-03-LP039.001 - **DESCRIBE** the function/purpose, design basis, and operating characteristics of the 4160 Volt Electrical Distribution System.

RO2-03-LP039.002 - **DESCRIBE** the 4160 Volt Electrical Distribution System. The description should include:

3. Interfaces with the following plant systems:
   c. Substation Emergency Aux. Transformers

RO2-03-LP039.005 - **EXPLAIN** the purpose of the following procedures used to govern the normal, abnormal, and emergency operation of the 4160 Volt Electrical Distribution System:
1. N-EHV-39, 4160V AC Supply and Distribution System Operation

REFERENCES:

N-EHV-39, Rev. P, 2.2.

SOURCE:

New

JUSTIFICATION:

The precaution identifies that the time limit is based on the consequences of the fault current which could be excessive.

B: While there is a requirement for independent transmissions lines serving the substation, this is not applicable to the individual switchgear but dependent on the substation breaker arrangement from the offsite sources. There are normally 3 pair of physically independent transmission lines, which are set up in performing A-SUB-59

C: The OCBs do not automatically trip on the conditions given. they would open on sensed fault on the associated substation bus section, or on a breaker failure trip if fault sensed for the TAT and Breaker TA-199 fails to open. The 30 second timer is associated is associated with opening the Main Generator Output Breaker (G-1) following a turbine trip.

D: Voltage restoration Logic for Bus 6 will prevent closure of the feeder breaker from the TAT to Bus 6 in the event the TAT is already supplying Bus 5 (normal configuration). It does not affect the Voltage Restoration scheme for DG B.

47. Given the following:

- Annunciator 47102-D, INSTRUMENT BUS INVERTER TROUBLE, is in alarm.
- SER Point 500 - BRA-111 Instrument Bus Inverter Trouble is in alarm.
- After reporting to the Battery Room, the EO reports the Alternate Source Supplying Load light is lit .

Which sequence describes the process for transferring load back to the inverter?

A. VERIFY Inverter Output breaker ON.
VERIFY In Sync light, OFF.
PRESS Inverter to Load pushbutton.
VERIFY the following:
1. Inverter Supplying Load light, ON.
2. Alternate Source Supplying Load light, OFF.
3. INSTRUMENT BUS INVERTER TROUBLE (47102-D), OFF.

B. VERIFY Inverter Output breaker OFF.
VERIFY In Sync light, ON.
PRESS Inverter to Load pushbutton.
VERIFY the following:
1. Inverter Supplying Load light, ON.
2. Alternate Source Supplying Load light, OFF.
3. INSTRUMENT BUS INVERTER TROUBLE (47102-D), OFF.

C. VERIFY Inverter Output breaker ON.
VERIFY In Sync light, ON.
PRESS Inverter to Load pushbutton.
VERIFY the following:
1. Inverter Supplying Load light, OFF.
2. Alternate Source Supplying Load light, ON.
3. INSTRUMENT BUS INVERTER TROUBLE (47102-D), OFF.

D. VERIFY Inverter Output breaker ON.
VERIFY In Sync light, ON.
PRESS Inverter to Load pushbutton.
VERIFY the following:
1. Inverter Supplying Load light, ON.
2. Alternate Source Supplying Load light, OFF.
3. INSTRUMENT BUS INVERTER TROUBLE (47102-D), OFF.

Answer: D
COGNITIVE LEVEL:

1-P - Procedural direction on restoring Instrument Bus power supply to inverter.

K/A:

058AA1.02 – Ability to operate and/or monitor the following as they apply to the Loss of DC Power: Static inverter dc input breaker, frequency meter, ac output breaker, and ground fault detector.

OBJECTIVE:

RO2-03-LP38.002 - **DESCRIBE** the DC and Emergency AC Electrical Distribution System. Include the following in the description;

2. Function/purpose, design basis, operating characteristics, and physical location as appropriate for the following major components:
   g. Inverters

RO4-03-LPD14.002 - In accordance with A-EDC-38, “Abnormal DC Supply and Distribution System”, **Summarize** the subsequent operator actions that are necessary to respond to the following:
e. Respond to an Instrument Bus Inverter Trouble Condition.

REFERENCES:

A-EDC-38, Rev. AC, step 4.5.4.

SOURCE:

New

JUSTIFICATION:

Actions also include expected control room operator observation (annunciator).

A: The In Synch light must be lit to assure uninterrupted transfer.

B: The inverter output breaker must be ON for all inverters except BRD-109.

C: This depicts unsuccessful transfer to inverter. Additional action is required to request electricians to investigate and repair.

48. Given the following:

- Diesel Generator (DG) B is being paralleled to bus 6 for a load test.
- The operator adjust voltages as follows:

  Incoming (DG) voltage is 122 volts.
  Running (Line) voltage is 120 volts.

- The synch scope is rotating slowly in the clockwise direction.

What is the electrical response for DG B when its supply breaker closes?

A. Picks up KW with VARS IN.

B. Reverse power with VARS IN.

C. Picks up KW with VARS OUT.

D. Reverse power with VARS OUT.

Answer: C
COGNITIVE LEVEL:

2-DR - Recognize and apply the relationship between generating sources voltages and speed (frequency) when paralleling these sources.

K/A:

062A4.03 – Ability to manually operate and/or monitor in the control room: Synchroscope, including an understanding of running and incoming voltages.

OBJECTIVE:

RO2-03-LP042.001 - **DESCRIBE** the function/purpose, design basis, and operating characteristics of the Emergency Diesel Generator and TSC Diesel Generator Systems.

RO2-03-LP042.005 - **DESCRIBE** the operation, automatic actions, interlocks, instrumentation, controls and alarms associated with the Emergency Diesel Generator and TSC Diesel Generator Systems and the following major system components:
2. DG voltage control switch.
3. DG speed control switch.

O-FND-LP 1.5.4 Obj 2.2.9 - State the purpose of a Synchroscope.

Obj 2.2.10 - Define the Following Terms:

a. Incoming Bus
b. Running Bus
c. Synchronous frequency

REFERENCES:

WEC Applied Electrical Technology for Power Plant Operations, Chapter 4, Synchronizing Procedure, page 4-34 & 35; Chapter 10, Generator Operation 3rd Paragraph, pages 10-19 through 10-21.

SOURCE:

Point Beach Bank.

JUSTIFICATION:

A: The DG will pick up load since the speed of the DG is slightly above that of the (other generating sources on the) grid. However the VARS will be OUT since the DG voltage is above the grid voltage.

B: Reverse power condition could occur if the Grid frequency was significantly above that of the DG. This is not indicated since the synch scope operation shows the DG is only slightly less below the grid frequency and will pick up load. The VARS will be OUT since the DG voltage is above the grid voltage.

D: Reverse power condition could occur if the Grid frequency was significantly above that of the DG. This is not indicated since the synch scope operation shows the DG is only slightly less below the grid frequency and will pick up load. VARS are predicted correctly since when the DG voltage is above grid voltage, the reactive component will be OUT.

49. Given the following:

- An Equipment Operator (EO) is testing BRA-108, Safeguards Train A Battery Charger for system grounds.
- DC Bus Voltage is 130 volts.

If a ground condition exists on the POSITIVE bus, what are the expected indications, and action to be taken, when the EO places the Ground Test toggle switch to the POS position?

	Ground Test Voltmeter	Ground Indicating GRD Light	Action
A	130 volts	Lit Bright	Selectively open breakers until light extinguishes.
B	57 volts	Lit Dim	Selectively open breakers until light extinguishes.
C	35 volts	Lit Bright	Notify the Control Room Supervisor.
D.	0 volts	Extinguished	Notify the Control Room Supervisor.

Answer: C
COGNITIVE LEVEL:

2-RW - Identify the conditions for grounds on the Safeguards DC Battery Bus and associated actions to be taken for that condition.

K/A:

063A2.01 – Ability to (a) predict the impacts of the following malfunctions or operations on the D.C. Electrical System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Grounds.

OBJECTIVE:

RO2-03-LP038.004 - **DESCRIBE** the operation, automatic actions, interlocks, instrumentation, controls, and alarms associated with the DC and Emergency AC Electrical Distribution System and the following major system components:
1. Battery Chargers

RO4-01-LPA09.003 - **DISCUSS** department and department personnel responsibilities associated with the following:
c. GNP-03.17.07, WATCHSTANDING PRACTICES (now incorporated in GNP-03.30.02)

REFERENCES:

N-EDC-38, Rev.P, 4.2.2.a.
System Description 38, Rev. 2, 3.3.5, page 15
NAD-08.02, Rev H, 4.1.1.

SOURCE:

New

JUSTIFICATION:

A: This is the expected voltage on the DC Bus, and the meter will read this when the TEST switch is in the neutral position. The light will glow bright on ground. This type of troubleshooting is not procedurally directed.

B: This is the nominal reading expected on the meter when taken to POS if NO ground exist on the bus. The light is normally lit dim. This type of troubleshooting may be performed but is notis not procedurally directed

D: This is the expected reading for the Non-Safeguards DC Battery Charger (BRC-108, BRD-108) when testing for grounds if NO ground exists. The light is NOT extinguished. Contacting the CRS is correct action.

50. Given the following:

- The plant is at 100% power.
- Diesel Generator (DG) A is supplying Bus 5 in parallel to the offsite power source.

What occurs if the EMERGENCY VOLTAGE SHUTDOWN button for DG A is pressed?

A. A Bus 5 Lockout is generated.

B. Diesel Generator A engine trips.

C. Diesel Generator A engine will begin to motor.

D. Breaker 1-509, DG A to Bus 5, trips.

Answer: D
COGNITIVE LEVEL:

1-I - DG response to actuation of Voltage Shutdown circuit.

K/A:

064K1.01 – Knowledge of the physical connections and/or cause-effect relationships between the ED/G System and the following systems: AC distribution system.

OBJECTIVE:

RO2-03-LP42A.004 - **DESCRIBE** the operation, automatic actions, interlocks, instrumentation, controls, and alarms associated with the Emergency Diesel Generator and TSC Diesel Generator Systems and the following major system components:
2. DG Voltage Control switch
8. DG Output Breaker

REFERENCES:

E-2022, Rev. N
47094-A, Rev. Orig., Comments 1.
System Description 42, Rev. 2, 3.3.1, page 9, second bullet

SOURCE:

KNPP Bank Question 0420000001K02 003

JUSTIFICATION:

A: Bus 5 lockout is generated by overcurrent on one of the source breakers to the bus, or Ground Overcurrent for one of the 3 offsite power breakers. Could consider condition would give overcurrent for Breaker 1-509

B: The engine will continue to run but excitation voltage is lost and the DG Supply breaker to Bus 5 trips.

C: With loss of excitation the engine will not motor, and the DG Supply breaker to Bus 5 trips preventing backfeed. The engine continues to run. Annunciator 47094-A will alarm.

51. Given the following:

- A plant startup is in progress at 6% power.
- The main turbine is being rolled to 1800 rpm.
- An instrument air leak occurs that results in instrument air header pressure subsequently lowering to 50 psig.
- The green and red lights for MS-1B, Main Steam Isolation Valve, are lit.

What action is required to be performed?

A. Trip the reactor and perform immediate actions of E-0, Reactor Trip or Safety Injection.

B. Start Air Compressors A and B, and direct the EO to investigate locally and verify standby air compressors loading per A-AS-01, Abnormal Station Air and Instrument Air System Operation.

C. Trip the main turbine, and initiate request for maintenance inspection of MS-1B per A-MS-06, Abnormal Main Steam and Steam Dump System Operation.

D. Verify turbine tripped and initiate reactor shutdown per N-0-4, 35% Power to Hot Shutdown Condition.

Answer: A
COGNITIVE LEVEL:

1-P - Procedural direction operator immediate action for loss of Instrument Air.

K/A:

065AA1.05 – Ability to operate and/or monitor the following as they apply to the Loss of Instrument Air: RPS.

OBJECTIVE:

RO4-03-LPD17.001 - **DESCRIBE** the purpose, symptoms, immediate operator actions, and automatic actions associated with the following procedures:
f. E-AS-01, *Loss of Instrument Air*:

REFERENCES:

E-AS-01, Rev. O, 2.2 & 3.2.1.

SOURCE:

New

JUSTIFICATION:

If Instrument Air header pressure falls below 60 psig, the actions of E-AS-01 are applicable (which includes tripping the reactor).

B: These are the actions of A-AS-01 which are appropriate for start of air compressors or component failures due to loss of instrument air. It is not appropriate in this instance when header pressure is < 60 psig.

C: The turbine trips when a MSIV comes 3º off its open position (as given in conditions). However, Maintenance inspection of MSIV is required only if the MSIV closes with the unit above 50% rated steam flow (~50% power).

D: This could be the correct action if the loss of Instrument Air was not so severe (> 60 psig) and a normal shutdown of the unit was desired.

52. What system is considered to be the most likely location for a rupture or break outside containment, and therefore is the system of primary concern during ECA-1.2, LOCA Outside Containment?

A. CC

B. CVCS

C. RHR

D. SI

Answer: C
COGNITIVE LEVEL:

1-B - Basis for ECA-1.1 system concerns for LOCA outside containment.

K/A:

W/E04EK2.1 – Knowledge of the interrelations between the LOCA Outside Containment and the following: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

OBJECTIVE:

RO4-04-LP020.001 - **DISCUSS** the following items as they relate to ECA-1.2, “LOCA Outside Containment”.
b. High Level Action Summary Steps.

REFERENCES:

ECA-1.2, Rev. I, Step 1
BKG ECA-1.2, Rev. Orig, Section 2.0, pages 3-5

SOURCE:

KNPP **2004** NRC Exam (INPO Bank)

JUSTIFICATION:

A: CC becomes an interface system between RCS/RHR and SW when Sump Recirculation is placed in service. However pressure between the systems is approximately equal (200 psig vs. 150 psig (CC)) and is therefore NOT a high likelihood of rupture

B: CVCS does intertie to RCS and could be considered, but is isolated on SI signal and therefore is NOT a likely location for rupture.

D: SI does intertie with RCS and RHR but due to piping rating (2100 psig normal maximum discharge pressure) is less likely a location for rupture as RHR (600 psig rated).

53. Given the following:

- The crew has just completed actions of FR-H.1, Response to Loss of Secondary Heat Sink, that establish and verify bleed and feed.
- Annunciators 47021-A, SI TRAIN A ACTUATED and 47021-B, SI TRAIN B ACTUATED, are lit.
- Annunciator 47042-A, PRESSURIZER PORV OPEN, is lit.
- SG A is faulted with ZERO indicated level on Wide Range Yarway.
- SG B is intact and reads 4% on Wide Range Yarway.
- RCS Core Exit Thermocouples are 570ºF (hottest) and slowly lowering.
- RCS Hot Leg temperatures are currently 565ºF in both loops and slowly lowering.
- The Turbine-driven AFW Pump was just made available.

What actions are taken to establish a secondary heat sink and ensure core heat removal?

A. Take NO action until bleed and feed has reduced RCS Hot Leg temperatures below 550ºF, and then feed SG B at maximum available rate. Terminate bleed and feed when SG B Narrow Range level is above 4%.

B. Establish feed to SG A at maximum available rate until RCS Hot Leg temperatures are less than 550ºF, and then establish feed to SG B at 60 to 100 gpm. Terminate bleed and feed when SG B Wide Range Level is above 5%.

C. Initiate feed flow at 60 to 100 gpm to SG B, and then terminate bleed and feed when SG B Narrow Range level is above 4%.

D. Feed SG B at the maximum available rate until RCS Hot Leg temperatures are less than 550ºF, and then take actions to terminate bleed and feed.

Answer: C
COGNITIVE LEVEL:

3-SPK - Determining the correct course of action based on given values from indications including - hot dry SGs - and successful instigation of bleed and feed.

K/A:

W/E05 2.4.31 – Loss of Secondary Heat Sink: Knowledge of annunciators alarms and indications, and use of the response instructions..

OBJECTIVE:

RO4-04-LP035.001 - **DISCUSS** the following items as they relate to FR-H.1

“Response to Loss of Secondary Heat Sink”.
c. Major Action Steps

RO4-04-LP035.003 - Given a set of plant conditions recommend the appropriate procedural action to be taken while implementing FR-H.1 “Response to Loss of Secondary Heat Sink”.

REFERENCES:

FR-H.1, Rev. X, Steps 19, 21, 25, 27, 28, 30 & 31
BKG FR-H.1, Rev. E, Section 2.4, pages 16-17; 3.1.2 through 3.1.4.

SOURCE:

KNPP Bank

JUSTIFICATION:

A: While there is concern with thermal stress to hot, dry SG, procedural direction is to initiate flow in controlled flow rate (60 to 100 gpm) to provide heat removal while minimizing stresses. Feed and bleed is terminated at SG level > 4% NR.

B: Feed flow should not be established to the faulted SG if intact one exists (procedural direction). When established feed flow to hot dry SG should be at a controlled rate, not maximum. Feed and bleed is not terminated until SG level is > 4% NR. Also at >5% WR level, feed flow to the SGs may be raised to a higher rate.

D: Feed flow should not be established at maximum rate until SG levels > 5% WR. Also, feed and bleed is not terminated until SG level is > 4% NR.

54. Given the following:

- Reactor power is at 90% with a power rise in progress using control rods.
- The operator determines that Control Bank D rod C-7 is not moving and its IRPI indication is 191 steps.
- All other Bank D rods are at 204 steps.
- Reactor Engineering has now determined that C-7 is repaired.
- The crew is performing A-CRD-49, Abnormal Rod Control System Operations.

How will control rod C-7 be realigned to control bank D, and how will the control bank insertion limit change following completion of the realignment?

A. Control Bank D will be realigned to control rod C-7 and control bank D insertion limit will be higher.

B. Control Bank D will be realigned to control rod C-7 and control bank D insertion limit will remain the same.

C. Control rod C-7 will be realigned to Control Bank D and control bank D insertion limit will be lower.

D. Control rod C-7 will be realigned to Control Bank D and control bank D insertion limit will remain the same.

Answer: D
COGNITIVE LEVEL:

2-DR - Describe the relationship between the rod realignment methodology and the affect on rod bank operating limits.

K/A:

005AK3.02 – Knowledge of the reasons for the following responses as they apply to the Inoperable/Stuck Control Rod: Rod insertion limits.

OBJECTIVE:

RO2-05-LP049.001 - **DESCRIBE** the function/purpose, design basis, and operating characteristics of the Rod Control and Rod Position Indication System.

RO4-03-LPD12.004 - **SUMMARIZE** the subsequent operator actions In accordance A-CRD-49, “Abnormal Rod Control System Operation” that are necessary to respond to the following:
a. Stuck Rod.

REFERENCES:

A-CRD-49, Rev. N, Attachment A
TRM 2.1, Rev. 6, COLR Cycle 27, Rev. 1, 2.5.1 and COLR Figure 4.
System Description 49, Rev. 2, page 28

SOURCE:

KNPP **2004** NRC Retake Exam (INPO Bank)

JUSTIFICATION:

A: An alternate method for realigning is to match rod bank height for unaffected rods to the affected rod position. This is NOT allowed procedurally but was procedurally allowed until recently. The RIL would remain unaffected provided power remained constant and RCS temperature remained on program. However, this may be considered since the margin to insertion limits would change (lower).

B: An alternate method for realigning is to match rod bank height for unaffected rods to the affected rod position. This is NOT allowed procedurally but was procedurally allowed until recently. As indicated, the RIL would remain unaffected provided power remained constant and RCS temperature remained on program.

C: This is the procedurally directed method for realignment of rods. The RIL would remain unaffected provided power remained constant and RCS temperature remained on program.

55. Given the following:

- Plant is at 75% power.
- RCS boron concentration is 480 ppm.
- Control Bank D is at 200 steps.
- Control Rod Bank Selector is in AUTO.
- Turbine load is being raised by 40 MWe.
- 47042-P, ROD CONTROL URGENT FAILURE, alarms.
- Rod motion stops.
- The AO reports the URGENT FAILURE light on the 1BD Power Cabinet is lit.

What action is taken for this condition?

A. Continue the load pickup using dilution.

B. Stop the load pickup and maintain Tave using boration or dilution.

C. Place rods in MAN and reduce reactor power below 50% within 4 hours.

D. Manually trip the reactor and go to E-0, Reactor Trip or Safety Injection.

Answer: B
COGNITIVE LEVEL:

1-P - Procedural direction for operation with Control Rod malfunction based on alarm conditions.

K/A:

001 2.4.50 – Control Rod Drive System: Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.

OBJECTIVE:

RO2-05-LP049.004: DESCRIBE the operation, automatic actions, interlocks, instrumentation, controls, and alarms associated with the Rod Control and Rod Position Indication System and the following major components:
- Power Cabinets

RO4-03-LPD12.004: SUMMARIZE the subsequent operator actions In accordance A-CRD-49, “Abnormal Rod Control System Operation” that are necessary to respond to the following:
d. Rod Control Urgent Failure

REFERENCES:

47042-P, Rev. A., Comments 2.

A-CRD-49, Rev. N, 2.1, steps 3 & 4.

SOURCE:

New

JUSTIFICATION:

A: This action is capable of being performed but is not allowed in accordance with reactivity management guidelines and the procedure.

C: With this failure, rod motion IN or OUT in MAN or AUTO is prohibited by the system. If the rods indicated misaligned condition on alarm TLA-1, the appropriate action would be to realign the rods or determine the core power peaking factors within four hours. If this cannot be done within the given time frame, then reactor power must be reduced to < 50%.

D: This is the immediate action required per the AOP if rods are moving and fail to stop when the Bank Selector is taken to MAN.

56. Given the following:

- The plant is in REFUELING SHUTDOWN.
- Both RHR Trains are in service to remove decay heat and cool the RCS.
- SW-10A, Aux. Bldg. SW Header A Isolation, has been closed due to a break in the header.
- SW-11A, Aux Bldg SW Header A & B Isolation, is CLOSED.
- The Trend Recorder is showing CC Hx outlet temp at 110ºF and increasing.

What operator actions should be taken to address the increasing CC Hx outlet temperature?

A. Start another SW Pump.

B. Start the standby CC Pump.

C. Open SW-1300B, Component Cooling Heat Exchanger B Outlet.

D. Fail Open SW-1306A, SW From CC Hx A Temp CV, by bleeding off its air regulator.

Answer: C
COGNITIVE LEVEL:

3-SPK - Using information provided including the impairment to SW system determine the appropriate action to establish additional CC cooling.

K/A:
076A4.04 – Ability to manually operate and/or monitor in the control room: Emergency heat loads.

OBJECTIVE:

RO4-03-LPD11.005 - In accordance with A-SW-02, Abnormal Service Water System Operation, **SUMMARIZE** the subsequent operator actions necessary to respond to the following:
4. Low SW Header Pressure from System Leakage.

REFERENCES:

A-SW-02, Rev. W, Step 9.a (CA)

SOURCE:

KNPP Bank question 0020010501A14 001

JUSTIFICATION:

A: Starting another SW Pump may provide additional flow in the SW system but does not significantly increase cooling to the in service Hxs. This is appropriate action for raising SW system pressure and operate with the header isolation valves (SW-3A & B) open

B: Starting the standby CC Pump will increase flow in the CC header but will not increase cooling of CC Hxs (to SW). A-CC-31 does address low CC flow and the expected rise in fluid process temperatures including RHR Hxs. However, the consideration is for inadequate CC flow not SW heat removal.

D: Operation of SW-1306A is an action required for E-O-06 in the event of Fire in Alternate Fire Zone. It assures flow through the Hx even if the control circuit does not provide an open signal to the valve (from SI sequencer). This action will not provide heat removal to SW since the flowpath for SW Train A is isolated.

57. Given the following:

- Service Water is operating with SW Pump 1A1, 1A2 and SW Pump 1B1 running.
- The Service Water headers are currently separated (SW-3A and SW-3B are closed).
- Service Water discharge header pressures read 108 psig on PI-41503, SW Train A Hdr, and 105.5 psig on PI-41506, SW Train B Hdr.

Using the attached Operator Aid 94-1, what is the status of the Service Water Pumps with regard to pump minimum flow requirements?

A. All SW Pumps fail to meet their required minimum flow.

B. SW Pumps 1A1 and 1A2 fail to meet their required minimum flow.

C. SW Pump 1B1 fails to meet its required minimum flow.

D. All SW Pump flows meet their required minimum flow.

Answer: B
COGNITIVE LEVEL:

3-SPR - Using the graph determine the flow for the pump alignments. Using the required minimum flow knowledge from the P&Ls of the Operating Procedure determine if flow values meet minimum flow requirements.

K/A:

076 2.1.32 – Service Water System (SWS): Ability to explain and apply all system limits and precautions.

OBJECTIVE:

RO2-02-LP002.002 - **DESCRIBE** the Service Water System, Include the following in the description;

2. Function/purpose, design basis, operating characteristics, and physical location as appropriate for the following major components:
   a. Service water pumps

RO2-02-LP002.005 - **EXPLAIN** the purpose of the following procedures used to govern the normal, abnormal, and emergency operation of the Service Water System:

1. N-SW-02, Service Water System.

REFERENCES:

Operator Aid 94-1, 3/22/94, Service Water Pump Flow Curves
N-SW-02, Rev. AB, 2.4

PROVIDED REFERENCE: Operator Aid 94-1, Service Water Pump Flow Curves

SOURCE:

KNPP Bank question 0020010501A14 001

JUSTIFICATION:

P&L states required minimum flow of 1800 gpm per pump.
Evaluation of the curve shows that for the 2 pumps running on Header A, the flow is ~2300 gpm @ 108 psig. The flow for Header B (one Pump) is ~2000-2100 gpm @ 105.5 psig.
Header A required minimum flow is 3600 gpm.

A: This would exist if the flow was read incorrectly for SW Pump 1B1 and correctly for Train A SW Pumps.

C: This would exist if the flow was read incorrectly for SW Pump 1B1 and if either the curve for 2 pumps is incorrectly read or the minimum flow is incorrectly applied (for one pump only).

D: This is selected if either the curve for 2 pumps is incorrectly read or the minimum flow is incorrectly applied (for one pump only). This selection could be correct if the 3 pumps were running combined on the SW header (SW-3A and SW-3B open). Then required flow is 5100 gpm and the graph indicates for 3 pumps a flow between 4500 and 7100 for the given pressures.

58. Given the following plant conditions:

- A loss of offsite power has occurred.
- Diesel Generator A has failed to start
- Bus 5 is deenergized.
- Diesel Generator B has started and its output breaker is closed and load sequencing has just initiated.
- NO Service Water pumps are currently running.
- Service Water pump 1A is selected on the SW pump preferred selector switch.
- Annunciator 47053-P SW PUMP BRG SEAL WTR FLOW LOW alarm is in with all associated SER points actuated.
- Service Water header pressure is 75 psig and dropping.

What condition will start Service Water pump B1 while the diesel generator load sequencing is still in progress?

A. The NCO takes the control switch momentarily to START and then releases it.

B. The NCO takes the control switch to START and holds it there for at least 5 seconds.

C. The pump will auto start immediately if the NCO positions the SW Pump Preferred Selector switch to 1B.

D. The pump will auto start as soon as Service Water header pressure falls below 72 psig.

Answer: B
COGNITIVE LEVEL:

1-I - Interlock that allows for operation of the SW Pump breaker when seal low flow condition exists.

K/A:

055K3.01 – Knowledge of bus power supplies to the following: Emergency/essential SWS pumps.

OBJECTIVE:

RO2-02-LP002.003 - **STATE** the power supply/motive power for the following Service Water System components:
1. SW pumps.

RO2-02-LP002.004 - **DESCRIBE** the operation, automatic actions, interlocks, instrumentation, controls and alarms associated with the Service Water System and the following major system components:
1. SW pumps.

REFERENCES:

E-1630, Rev. Q.
System Description 02, Rev. 3, Sections 3.2.3, page 11.

SOURCE:

KNPP ILT Bank Question RO2-02-LP002.004 002

JUSTIFICATION:

A: This would be the "normal" start for a SW Pump. The pump will NOT start by taking the control switch to START if adequate seal water flow does not exist.

C: Operating the Selector switch changes nothing, only the sequencing of pump starts on decreasing header pressure (provided other interlocks satisfied & NO Auto Inhibit).

D: The pump is blocked from auto starting on low header pressure signals (72 psig) until the load sequencing is complete (Auto Inhibit). The low pressure signal does not exist yet.

59. What is/are the Station and Instrument Air System compressor(s) that will trip and lockout on receipt of an SI signal?

A. "A" only.

B. "B" only.

C. "A" and "C".

D. "B" and "C".

Answer: A
COGNITIVE LEVEL:

1-I - System response to SI signal.

K/A:
078K4.01 – Knowledge of IAS design feature(s) and/or interlock(s) which provide for the following: Manual/automatic transfers of control.

OBJECTIVE:

RO2-02-LP001.004 - **DESCRIBE** the operation, automatic actions, interlocks, instrumentation, controls and alarms associated with the Station and Instrument Air System and the following major system components:
3. Small-capacity compressors (A, B and C).

REFERENCES:

E-1603, Rev. AA
System Description 01, Rev. 3, Section 3.3. page 9.

SOURCE:

KNPP Bank question 0020010501A14 001

JUSTIFICATION:

B: Compressor B is stopped by an Auto Inhibit signal and is allowed to start at step 9 of the loading sequence.

C: Compressor A is correct. Compressor C is stopped by an Auto Inhibit signal and is allowed to start at step 9 of the loading sequence.

D: Compressors B & C are stopped by an Auto Inhibit signal and allowed to start at step 9 of the loading sequence.

60. Given the following:

- Plant is at 50% power.
- Valve RC-412, Pressurizer Liquid Sampling Isol (Inside Cntmt), and RC-413, Pressurizer Liquid Sampling Isol (Outside Cntmt), were opened for sampling.
- RC-413 closed when its control switch was placed in the CLOSE position.
- RC-412 failed to close when its control switch was taken to the CLOSE position.

What is the effect on continued plant operations?

A. The plant must be taken to HOT SHUTDOWN within ONE hour, and RC-412 repaired prior to taking the reactor critical.

B. The plant must be taken to HOT SHUTDOWN within ONE hour, and then to COLD SHUTDOWN within the next 24 hours. RC-412 must be repaired prior to raising RCS temperature above 200ºF.

C. Operation may continue indefinitely provided RC-413 is closed and the air supply is isolated to its operator within 24 hours.

D. Operation may continue indefinitely provided RC-413 is maintained in the CLOSE position, and the next downstream manual Valve RC-414,Pressurizer Liquid Sample Heat Exchanger Inlet, is tagged CLOSED within 72 hours.

Answer: C
COGNITIVE LEVEL:

1-P - Technical Specification LCO and ACTION statements.

K/A:

103K3.02 – Knowledge of the effect that a loss or malfunction of the Containment System will have on the following: Loss of containment integrity under normal operations.

OBJECTIVE:

RO2-04-LP056.007: **EXPLAIN** the LCO operation, applicability and action requirements for each of the following Technical Specifications associated with the Containment System:
3. TS 3.6, Containment System

REFERENCES:

KNPP Technical Specifications, 3.6.b.3.A, Amendment 155

SOURCE:

New

JUSTIFICATION:

TS 3.6.b.3.A. requires for a penetration flow paths with two containment isolation valves per penetration with one containment isolation valve inoperable, that the inoperable valve be restored within 24 hours OR isolate the affected penetrations flow path by use of at least one closed and de-activated automatic valve.

A: One hour action is required only if BOTH valves in the penetration are inoperable, and the ACTION is not limited to going to HOT SHUTDOWN.

B: One hour action is required only if BOTH valves in the penetration are inoperable. Going to COLD SHUTDOWN is appropriate. (The time is not consistent with TS ACTION however.)

D: The affected penetrations flow path is isolated by use of at least one closed and <u>de-activated</u> automatic valve. Closure of the isolation valve (RC-414) inside the Sampling Room does not meet the intent of Containment Isolation. Also the allowed time for action is 24 hours.

61. If RVLIS is <u>NOT</u> available, which of the following post-accident monitoring instruments would provide the FIRST indication of the existence of a bubble in the reactor vessel head?

A. Pressurizer water level.

B. RCS wide-range pressure.

C. Core exit thermocouples.

D. RCS wide-range hot leg temperature.

Answer: A
COGNITIVE LEVEL:

1-P - Procedural direction for evaluating void growth.

K/A:

002K6.03 – Knowledge of the effect of a loss or malfunction on the following RCS components: Reactor vessel level indication.

OBJECTIVE:

RO2-05-LP-050.002: **DESCRIBE** the Incore Instrumentation & Inadequate Core Cooling Monitor Systems to include the following in the description;

1. Function/purpose, design basis, operating characteristics, and physical location as appropriate for the following major components:
   e. Reactor Vessel Level Indication (RVLIS)
2. Interfaces with the following:
   b. Reactor Coolant System

RO4-04-LP006.001: **Discuss** the following items as they relate to ES-0.2, Natural Circulation Cooldown.
b. High Level Action Summary Steps.

REFERENCES:

ES-0.2, Rev. P, Step 15.
BKG ES-0.2, Rev. Orig., 3.1.2

SOURCE:

Point Beach NRC Exam 8/1999(INPO Exam Bank)

JUSTIFICATION:

The most direct indication of head void growth other than RVLIS is rising pressurizer level.

B: Wide range pressure is not an indication. However, in combination with Core Exit TC temperatures and/or Hot Leg temperatures, a subcooling value can be calculated that might be indicative of conditions favorable to void formation.

C: Core exit temperatures would not be a good indication of void formation. The instruments are well below the expected temperature zone and natural circ flowpath in the event hotter water was displaced from the head.

D: HL temperatures may be used but are not a good indicator. These would possibly rise as the hotter water in the vessel head was displaced and entered the natural circ flowpath. However, too many other factors can affect HL temperatures, such as changes in natural circ flow or secondary heat removal.

62. Given the following:

- Plant is at 90% power.
- Power Range Instrument N-42 fails and is removed from service per A-MI-87, Bistable Tripping for Failed Reactor Protection or Safeguards Inst.
- The surveillance for SP-47-011A, Reactor Coolant Temperature and Pressurizer Pressure Instrument Channel I (Red) Calibration, comes due.
- The decision is to bypass the failed NIS channel inputs to allow the calibration (Technical Specification 3.5.d).

What is the coincidence for the OT∆T reactor trip while the surveillance is in progress?

A. 2 out of 2

B. 2 out of 3

C. 1 out of 4

D. 1 out of 3

Answer: B
COGNITIVE LEVEL:

2-DR - Recognize the relationship between placing channel in Bypass and channel status while performing surveillance.

K/A:

015K4.06 – Knowledge of NIS design feature(s) and/or interlock(s) which provide for the following: Reactor trip bypasses.

OBJECTIVE:

RO2-05-LP471.007 - **EXPLAIN** the LCO operation, applicability and action requirements for each of the following Technical Specifications associated with the Reactor Protection Day 1 System:
3. T.S. 3.5.

RO2-05-LP472.002 - **DESCRIBE** the Reactor Protection System Day 2 instruments to include the following in the description:.
2. Function/purpose, design basis, operating characteristics, and physical location as appropriate for the following major components:
   f. Reactor Protection Logic Channels.
3. Interfaces with the following:
   a. Process Instrumentation: 2. RCS Temperature

REFERENCES:

A-MI-87, Rev. R, 4.5 & pages 23-25 (N-42 RFS)
KNPP Technical Specification 3.5.d, Amend No. 105
KNPP Technical Specification Table TS 3.5-2, No. 5, Amend No. 137

SOURCE:

New

JUSTIFICATION:

There are 4 channels of OTΔT normally, and the trip logic is 2 of 4. With one channel bypassed (to prevent a trip be generated while the other channel is calibrated) it does not provide input into the logic. Therefore with 3 remaining channels, 2 channels are required to trip the plant.

A: This would be true if the logic was 2 of 3 such as for Pressurizer High Pressure trip or Przr High level trip.

C: Bypassing the failed channel removes its input therefore the 4 channels are reduced to 3. This would be the logic in the case before the failed channel was bypassed.

D: Bypassing the failed channel removes its input therefore the 4 channels are reduced to 3. This is the case for the normal logic with no channel failures.

63. Given the following:

- Power level is 99.9% (1771.5 MWt).
- The current UFMD and RTO OPERATING LIMITs are 1772 MWt.
- SP-87-125, Shift Instrument Channel Checks - Operating, calorimetric was completed 6 hours ago.
- The signal is lost to PPCS from Feedwater Flow channel FT-476.
- The RTO 1-minute average, PPCS point R5110G, drops to 1200 MWt.
- The Computer Group reports the signal will be restored within 10 minutes.

What is the affect on the UFMD and RTO Operating Limits, and what action is taken when the flow channel input is restored and R5110G reads normal?

A. Both the UFMD and RTO Operating limits will read 1749 MWt.
When the signal is restored, the operator will click the APPLY UFMD LIMIT button on the PPCS UFMD Correction Factors screen to return both limits to 1772 MWt.

B. The UFMD Operating Limit will read 1749 MWt and RTO Operating Limit will read 1772 MWt.
Power will be reduced to less than 1749 MWt, and when the signal is restored, the operator will will click the APPLY UFMD LIMIT button on the PPCS UFMD Correction Factors screen to return the UFMD limit to 1772 MWt.

C. Both the UFMD and RTO Operating limits will read 1769 MWt.
When the signal is restored, the operator will click the APPLY UFMD LIMIT button on the PPCS UFMD Correction Factors screen to return both limits to 1772 MWt.

D. The UFMD Operating Limit will read 1769 MWt and RTO Operating Limit will read 1772 MWt.
Power will be reduced to less than 1769 MWt, and when the signal is restored, the operator will click the APPLY UFMD LIMIT button on the PPCS UFMD Correction Factors screen to return the UFMD limit to 1772 MWt.

Answer: A
COGNITIVE LEVEL:

1-I - System Response to input failure and operator action required by procedure.

K/A:

016A2.03 – Ability to (a) predict the impacts of the following malfunctions or operations on the NNIS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Interruption of transmitted signal.

OBJECTIVE:

RO2-03-LP046.002: DESCRIBE the Plant Process Computer System to

include the following in the description; function/purpose, design basis, operating characteristics, and physical location as appropriate for the following major components:
3. Control Room Input/output Devices

REFERENCES:

A-CP-46A, Rev. F, 3.1, steps 1, 6 & 7.
N-O-03, Rev. AW, 2.17

SOURCE:

Modified LRC-03-LPJT2.002 2

Changed values to current (updated).
Removed the given values after failure from premise.
Removed reference to TLA alarm and clearing alarm.
All choices changed to include values for limits after failure.

JUSTIFICATION:

B: If failure did not lower R5110G below 1240 MWt then RTO is not automatically reset. Also if RTO remains higher than the UFMD Limit value, TLA-28, POWER GREATER THAN UFMD LIMIT, will alarm.

C: The UFMD / RTO Limit of 1769 MWt applies only if only the Temperature Correction factors for the UFMD has an invalid signal. With power below 70% (1240 MWt) both the inputs for flow correction and temperature corrector factors will read bad.

D: If failure did not lower R5110G below 1240 MWt then RTO is not automatically reset. Also if RTO remains higher than the UFMD Limit value, TLA-28, POWER GREATER THAN UFMD LIMIT, will alarm. The UFMD / RTO Limit of 1769 MWt applies only if only the Temperature Correction factors for the UFMD has an invalid signal. With power below 70% (1240 MWt) both the inputs for flow correction and temperature corrector factors will read bad.

64. What is the sequence of actions required to display the CET T21 (Train A) temperature value on the ICCMS Train A display?

Press the CET ID/CET TEMP pushbutton to illuminate the...

(See attached picture of CET Monitor Train A display.)

A. CET TEMP lamp, then press the HOT/AVG pushbutton to display T21 CET temperature.

B. CET TEMP lamp, then depress the REF1/REF2 pushbutton to display T21 CET temperature.

C. CET ID lamp, and then press the HOT/AVG pushbutton until "21" is displayed. Press the REF1/REF2 pushbutton to illuminate the REF 2 lamp and display T21 CET temperature.

D. CET ID lamp, and then press the HOT/AVG pushbutton until "21" is displayed. Press the CET ID/CET TEMP to illuminate CET TEMP lamp and display T21 CET temperature.

Answer: D
COGNITIVE LEVEL:

1-P - Procedural direction to achieve desired display.

K/A:

017A1.01 – Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ITM System controls including: Core exit temperature.

OBJECTIVE:

RO2-05-LP050.002: **DESCRIBE** the operation, automatic actions, interlocks, instrumentation, controls and alarms associated with the Incore Instrumentation & Inadequate Core Cooling Monitor Systems and the following major system components:
2. Core Exit Thermocouple (CET) Monitor.

REFERENCES:

N-II-50, Rev. G, 4.2.3.c.

SOURCE:

New

JUSTIFICATION:

A: This is the procedural direction for displaying either the HOTTEST or AVG CET temperature.

B: This is the procedural direction for displaying the Reference Junction Box temperature.

C: This will display cause the displayed CET ID numbers to change in reverse direction (count down).

65. What is the reason for replacing hydrogen with carbon dioxide before admitting air into the Main Generator when the generator is to be entered for maintenance/inspection?

A. Carbon dioxide minimizes the danger of explosion.

B. Carbon dioxide prevents electrical arcing in the generator.

C. Carbon dioxide cools the generator more quickly than hydrogen.

D. Carbon dioxide prevents the internal metal surfaces from warping.

Answer: A
COGNITIVE LEVEL:

1-F - Factual information on using intermediary gas when degassing main generator.

K/A:

017A1.01 – Knowledge of the operational implications of the following concepts as they apply to the MT/G System: Possible presence of explosive mixture in generator if hydrogen purity deteriorates.

OBJECTIVE:

RO2-03-LP043.002: **DESCRIBE** the Electrical Generation System. The description should include:
3. Interfaces with the following plant systems:
d) Miscellaneous Gas Systems

RO2-03-LP043.002: **DESCRIBE** the Electrical Generation System. The description should include:
3. Interfaces with the following plant systems:

REFERENCES:

System Description 84, Rev. 1, 3.2, page 8.

SOURCE:

KNPP Bank Question RO2-03-LP043.004 001

JUSTIFICATION:

B: Carbon dioxide does not conduct electricity, and so it can be used safely on electrical fires. However the purpose for using it is not reducing arcing.

C: CO_2 does not provide cooling more rapidly than H_2. However since CO_2 gas is stored under pressure, when it is released without providing warming, it will cool upon expansion.

D: CO_2 is not used to prevent warping of interior surfaces. The use of H_2 gas dryers is to prevent warping and corrosion products by removing moisture from the H_2 gas

66. Given the following:

- Reactor power is 100%.
- A single set of air ejectors are in service.

What parameter trend will be observed if MS-502, Air Ejector Steam Inlet, fails closed with NO operator action is taken?

A. Lowering condenser hotwell level.

B. Lowering steam header pressure.

C. Rising steam seal header pressure.

D. Rising condenser saturation temperature.

Answer: D
COGNITIVE LEVEL:

2-RI - Recognize the interaction between air ejector supply steam loss and rising condenser vaccum, and other plant system parameters.

K/A:

055K3.01 – Knowledge of the effect that a loss or malfunction of the CARS will have on the following: Main condenser.

OBJECTIVE:

RO2-02-LP003.002 - **DESCRIBE** the Condensate & Air Removal Systems, Include the following in the description;
2. Function/purpose, design basis, operating characteristics, and physical location as appropriate for the following major components:
   j) Air ejectors.

RO4-03-LPD11.004 - In accordance with E-AR-09, Loss of Condenser Vacuum, **SUMMARIZE** the subsequent operator actions necessary to respond to the following::
c. Insufficient air ejector steam supply.

REFERENCES:

E-AR-09, Rev.O, 2.2
System Description 09, Rev. 2, 1.6.
Steam Tables

SOURCE:

New

JUSTIFICATION:

As the condenser pressure rises the saturation temperature for the steam also rises.

A: Condenser level may slowly increase since there is less removal of steam from shell through Steam Jet Ejector.

B: The steam supply to the Air Jet Ejector is closed and steam header pressure will remain approximately the same or rise slightly due to the reduced steam usage..

C: Seal header pressure will not change due to controller operation for MS-302 (~ 18 psig)

67. Given the following:

- Waste Gas Decay Tank A is in service.
- The relief valve for that tank, WG-13A, lifts and fails to reseat.

What is the effect on the plant?

A. The release will be automatically isolated when the Waste Gas Analyzer senses the pressure drop in the Waste Gas Decay Tank.

B. The release will be automatically isolated when the Waste Gas Decay Tank pressure reducing control valve WG-201 closes.

C. The release will <u>NOT</u> be automatically isolated, but will be monitored by the Aux. Building Vent radiation monitors R-13 and R-14.

D. The release will <u>NOT</u> be automatically isolated, but will be detected by the Charging Pump Room Area Monitor R-4.

Answer: C
COGNITIVE LEVEL:

1-S - Understanding of Waste Gas system flowpaths including WGT reliefs.

K/A:

071K1.04 – Knowledge of the physical connections and/or cause-effect relationships between the Waste Gas Disposal System and the following systems: Station ventilation.

OBJECTIVE:

RO2-01-LP045.004 - **DESCRIBE** the operation, automatic actions, interlocks, instrumentation, controls and alarms associated with the RADIATION MONITORING System and the following major system components:
2. Process Radiation Monitors.

REFERENCES:

OPERXK-100-132, Rev. AF.
OPERM-604, Rev. BE
OPERM-601, Rev. CP

SOURCE:

KNPP Bank Question

JUSTIFICATION:

The relief line is ducted into the Aux Building and the SV Zone System ducting. The flow is then released through the pathway monitored by R-13 and R-14.

A: The release is not stopped by the Gas Analyzer as the piping is separate. An alarm may result however.

B: The release is not stopped by the pressure control valve as the piping is separate. It is designed to maintain a downstream pressure of 5 psi in route to the gas analyzer.

D: It is not detected by the Charging Pump room monitor under normal airflow conditions. However, the ducting for the ventilation in the Aux Building is common for both areas, and if fans were to trip it is possible that air could be directed into the charging pump room via the duct.

68. Given the following plant conditions:

- Safety Injection has been manually actuated
- The crew has completed the immediate actions of E-0, "Reactor Trip Or Safety Injection."
- At that time a fire alarm is received.

For the above conditions, what are the indications available in the control room that identifies a fire has actuated the alarm switches associated with the Fire Protection Status light below?

AUX FW PMP 1B
BUS 61-62 SWGR AREA
FIRE

A. Fire Pump A and B GREEN and WHITE lights lit. Fire Protection Header Pressure indicating below 130 psig.
Annunciators 47051-L, FIRE DETECTION SYSTEM ACTIVATED, and 47053-L, FIRE DETECTION SYSTEM TROUBLE, are in alarm condition.

B. Fire Pump A and B RED and WHITE lights lit. Fire Protection Header Pressure indicating approximately 125 psig.
Annunciators 47051-L, FIRE DETECTION SYSTEM ACTIVATED, and 47052-L, FIRE PROTECTION HDR PRESSURE LOW, are in alarm condition.

C. Fire Pump A RED light lit. Fire Protection Header Pressure indicating approximately 155 psig.
Annunciators 47051-L, FIRE DETECTION SYSTEM ACTIVATED, and 47054-L, FIRE PUMPS ABNORMAL, are in alarm condition.

D. Fire Pump B RED light lit. Fire Protection Header Pressure indicating approximately 130 psig.
Annunciators 47053-L, FIRE DETECTION SYSTEM TROUBLE, and 47054-L, FIRE PUMPS ABNORMAL, are in alarm condition.

Answer: A

COGNITIVE LEVEL:

1-F - System actuation identifying the appropriate agent used and indication of actuation.

K/A:

086A4.03 – Ability to manually operate and/or monitor in the control room: Fire alarm switch.

OBJECTIVE:

RO2-02-LP008.002: DESCRIBE the Fire Protection System, Include the following in the description.

2. Function/purpose, design basis, operating characteristics, and physical location as appropriate for the following major components:
   b. Fire Pumps (A/B).
   k. Deluge Valves.

RO2-02-LP008.004: DESCRIBE the operation, automatic actions, interlocks, instrumentation, controls and alarms associated with the Fire Protection System and the following major system components:
1. Fire pumps (A/B)
3. Deluge valves

REFERENCES:

N-FP-08, Rev. AR, 2.2
E-FP-08, Rev. AP, 3.1 NOTE 1.
E-0, Rev. Y, CAUTION prior to Step 17.
Alarm Response, 47051-L, Rev. B, SER0046
Alarm Response, 47053-L, Rev. A, SER0079
E-1619, Rev. X

SOURCE:

Modified RO2-02-LP008.004

Changed the question to a different Fire Zone with a different actuation system and different suppression medium. (Ionization detectors, flow switch and pressure switch instead of temperature [rate of heat rise]; water instead of CO2)
Provided the Fire Panel alarm status light.
Limited selections to annunciator alarm and control room equipment indications.

JUSTIFICATION:

Normally for this actuation, at least Fire Pump A will start, due to the deluge valve opening and the subsequent loss of fire header pressure. Unlike other deluge valve operation, opening of the Deluge valve FP-264, Aux Feed Pumps/Safeguards Alley Sprinkler System, does not automatically start Fire Pump A. However since Safety Injection has actuated and has not been reset, the WHITE lights for the pumps will be lit and the pumps are inhibited from starting until the control switch for the pump(s) is taken to the OFF/RESET position. The indications and value provided are the expected parameters to be seen, with the Fire Jockey Pump supplying 30 gpm to the system (normal header pressure is ~ 130 psig).

B: This condition may be expected if the demand on the system was so great that header pressure fell below 102 psig (Fire Pump A starts on lowering header pressure at 110 psig and Fire Pump B starts at 102 psig), which may occur when the deluge valves open. However, the SI signal prevents automatic starting of the Fire Pumps. Also, the alarm 47052-L would alarm only at less than 62 psig, thereby indicating problems with the fire header and or the Fire Pumps, and the conditions given do NOT allow for it to be in alarm.

C: This condition may be expected since the Deluge valve FP-264 opens to supply water to the fire sprinkler system; however, unlike opening other deluge valves, this one does NOT start Fire Pump A, and the Si signal will prevent auto start of the Fire Pump.

D: This condition is NOT expected since Fire Pump A start on a higher pressure for dropping Fire header Pressure than Fire Pump B (110 psig for Pump A and 102 Psig for Pump B). However both Fire Pumps are affected by the SI signal. It is plausible since the reverse conditions for starting pressures for the Fire Pumps could be considered.

69. Given the following:

- The plant is at 100% power.
- PRZR level transmitter LT-428 (Channel III) has failed to ZERO.
- The crew has completed the actions of A-MI-87, Bistable Tripping for Failed Reactor Protection or Safeguards Instruments, section 1.0, Prerequisite Alignments, for removing LT-428 from service.

Why is the next section, placing the associated bistables in UP (test) position, performed?

A. Allow the PRZR heaters to be reset and letdown to be restored with the channel failed low.

B. Ensure the RPS High Level Trip will function in the event another channel failure occurs.

C. Bypass the failed channel so that an another failure does <u>NOT</u> result in an inadvertent reactor trip.

D. Return the channel to OPERABLE status.

Answer: B
COGNITIVE LEVEL:

2-RI - Recognizing interaction between a failed PRZR level channel and the RPS when the associated bistables are placed in test.

K/A:

028AA1.01 – Ability to operate and/or monitor the following as they apply to the Pressurizer Level Control Malfunctions: PZR level reactor protection bistables..

OBJECTIVE:

RO2-05-LP36D.004 - **DESCRIBE** the operation, automatic actions, interlocks, instrumentation, controls and alarms associated with pressurizer level control and the following major system components:
4. High level reactor trip.

RO2-05-LP36D.007 - **EXPLAIN** the LCO operation, applicability and action requirements for each of the following Technical Specifications associated with pressurizer level control:
3. Table 3.5-2 No. 9.

RO2-05-LP472.005 - **EXPLAIN** the purpose of the following procedures used to govern the normal, abnormal, and emergency operation of the Reactor Protection System:

1. A-MI-87, Bistable Tripping for Failed Reactor Protection or Safeguards Inst.

REFERENCES:

AMI-87, Rev. R, 4.1.4; LT-428 removal From Service, page 42.
KPS TS Table 3.5-2 Item No. 9, Amendment No. 94

SOURCE:

New

JUSTIFICATION:

The next section calls for I&C to place 428A HIGH LEVEL TRIP bistable in UP (test) position. This trips that bistable, and one other PRZR level channel reaching the high level trip setpoint will result in a reactor trip.

A: Allowing the restoration of PRZR heaters and letdown is accomplished by selecting the Przr Level Control Channel Selector to 2-1 position. This removes the control function from the normal (2-3) control channel III.

C: This action does not bypass the trip function. Technical Specifications does allow this channel to be bypassed (blocked) to prevent a reactor trip if one of the other two channels is to be tested (surveillance).

D: This has been a common misconception in the recent pass. The procedure used to imply that the channel could be considered OPERABLE once the bistable was tripped for the failed channel (since the RPS function was now in its required state). This was changed to explicitly indicate that channel is NOT returned to OPERABLE by this action.

70. Given the following:

- The plant is operating at 90% power.
- A primary to secondary leak has developed in SG A.
- 47043-E, PRESSURIZER LEVEL DEVIATION annunciator has just alarmed.
- Pressurizer level is below program and lowering slowly.
- RCS subcooling is 28ºF.

Which of the following actions is required FIRST for these conditions upon entry into E-O-14, Steam Generator Tube Leak?

A. Manually trip the reactor and actuate SI.

B. Maximize charging to restore pressurizer level.

C. Commence a plant backdown to 50% with loading rate of 5% per minute.

D. Maximize VCT blended makeup flow to prevent approach to RWST switchover.

Answer: B
COGNITIVE LEVEL:

3-SPK - Determine the correct action based on the Przr level change indication change.

K/A:

037AA1.11 – Ability to operate and/or monitor the following as they apply to the Steam Generator Tube Leak: PZR level indicator.

OBJECTIVE:

RO4-03-LPD07.006 - In accordance with E-0-14, “Steam Generator Tube Leakage”, **Summarize** the subsequent operator actions that are necessary to respond to a primary to secondary leak in any Steam Generator.

REFERENCES:

47043-E, Rev. C, setpoint; Comment 2
E-O-14, Rev. K, 3.2; Step 1 (CA)

SOURCE:

New

JUSTIFICATION:

At 90% power (569.5ºF), Przr level program setpoint is 44.1%. The annunciator will alarm at +/- 10% from program. This means that the level has just dropped below 34.1%. The first Contingency Action of step 1, when PRZR level is determined NOT to be stable, is to increase charging to maintain level.

A: Conditions do NOT warrant a reactor trip or SI. Indications for this action include RCS subcooling < 20ºF (critical) or < 30ºF (not critical) OR inability to maintain Przr level above 5%.

C: The reduction in power is the first action addressed after determination that a LEVEL 3 (leak rate) condtion exists. It is addressed after attempts to maintain Przr level.

D: There is no procedural direction to maximize makeup flow. There is direction that has the operator verify the Reactor makeup System is operable.

71. Given the following:

- The plant is at 75% power with all systems in automatic alignment.
- PPCS Point P0300A, TURBINE CDSR PRESSURE, has just alarmed on low value.
- Condenser vacuum continues to slowly degrade.

What event would occur first when responding to the above conditions per E-AR-09, Loss of Condenser Vacuum?

A. Turbine runback occurs.

B. Turbine automatically trips.

C. The operator manually trips the reactor.

D. Operator checks Status Light 44905:0301 CDSR AVAIL DUMP PERM, extinguished, and reports Condenser Steam Dumps inoperable.

Answer: C
COGNITIVE LEVEL:

3-PEO - Based on the conditions predict the next expected action/event for a loss of vacuum. Requires knowledge of annunciator setpoints, and specific action setpoints based on condenser pressure.

K/A:
051A 2.4.50 – Loss of Condenser Vacuum: Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.

OBJECTIVE:

RO4-03-LPD11.001 - **DESCRIBE** the purpose Immediate operator actions, and automatic actions associated with the following procedures:
c. E-AR-09, Loss of Condenser Vacuum.

RO4-03-LPD11.004 - In accordance with E-AR-09, Loss of Condenser Vacuum, **SUMMARIZE** the subsequent operator actions necessary to respond to the following:
d. Air in-leakage.

REFERENCES:

47051-W, Rev. C, setpoint
E-AR-09 Rev. O, 4.2.2.a

System Description 09, Rev. 2, 1.6,

SOURCE:

New

JUSTIFICATION:

The PPCS alarm will come in at 2.5" HgA. The annunciator alarm actuates at 5" HgA. The procedure shows the turbine "DO NOT OPERATE" zone has a maximum allowable operating value of 5.5" HgA.

A: The turbine has runback signals from OTΔT and OPΔT, and a VPL runback for a trip of a FW Pump above 67% power. Condenser pressure does NOT cause a runback.

B: Turbine trip will occur at 10" HgA.

D: Condenser Steam Dumps are interlocked with permissive for Condenser pressure at 6.5" HgA and can be overridden to allow operation up to 20" HgA condenser pressure.

72. What type of detector is the New Fuel Pit Area Monitor, R-10, and what warning alarms/indications are associated with it?

A. Ion Chamber.
Sounds the containment evacuation horn and an alarm is provided in the control room.

B. Ion Chamber.
Audible alarm is located at the monitor and an alarm is provided in the control room.

C. GM tube.
Audible alarm is located at the monitor and energizes a flashing light at the entrance to the Auxiliary Building.

D. GM tube.
Local indication of value only and an alarm is provided in the control room.

Answer: B
COGNITIVE LEVEL:

1-F - Fact on the type detector used for radiation monitor and the type of alarm indications provided.

K/A:

061AK2.01 – Knowledge of the interrelations between the Area Radiation Monitoring (ARM) System Alarms and the following: Detectors at each ARM system location.

OBJECTIVE:

RO2-01-LP045.002 - **DESCRIBE** the RADIATION MONITORING System to include the following in the description;
2. Function/purpose, design basis, operating characteristics, and physical location as appropriate for the following major components:
   b. Area Radiation Monitors.

RO2-01-LP045.004 - **DESCRIBE** the operation, automatic actions, interlocks, instrumentation, controls and alarms associated with the RADIATION MONITORING System and the following major system components:
1. Area Radiation Monitors.

REFERENCES:

47011-B, Rev. E, Comments 1
System Description 45, Rev. 1, 3.5, pages 19-20.

E-2021, Rev. Z

SOURCE:

New

JUSTIFICATION:

A: Sounding of the containment evacuation alarm is a MANUAL action but is directed to check for actuation in A-RM-45 if any monitor R2 through R-10 is in ALERT or HIGH alarm.

C: Most of the remaining Area Radiation monitors are GM (except R-5, R-10 & R-30). R-30 for example has alarm indication prevents entry into the area where the Incore NIs are stored.

D: Most of the remaining Area Radiation monitors are GM. many of the area rad monitors only have local indication of reading, examples are R-25 through R-29 and R-39.

73. Given the following:

- The plant is at 10% power.
- A fire breaks out in the Cable Spreading Area.
- Control Room indications are reading sporadically.
- Some equipment associated with Train B has either started when <u>NOT</u> required OR has stopped when operation is desired.
- The Shift Manager determines the main control room must be evacuated and enters E-O-06, Fire In Alternate Fire Zone.
- Following the pressing of the Reactor Trip pushbuttons, the following parameters are observed:

  - Reactor Trip Breaker A RED light is lit.
  - Reactor Trip Breaker B GREEN light is lit.
  - Rod Position Indicators and Rod Bottom Lights are <u>NOT</u> lit.
  - N41 reads 10% and stable.
  - N42 reads 0%.
  - N43 reads 3% and slowly lowering.
  - N44 reads 7% and stable.
  - N35D Start-Up Rate reads ZERO dpm.
  - N36D Start-Up Rate reads ZERO dpm.

What is the NEXT action performed that addresses the status of the reactor?

A. Initiate an emergency boration in accordance with E-CVC-35, Emergency Boration.

B. Establish Cold Shutdown boron concentration from the Dedicated Shutdown Panel.

C. Locally open the reactor trip breakers after evacuating the Control Room.

D. Open Bus 33 and Bus 43 supply breakers.

Answer: C
COGNITIVE LEVEL:

1-P - Knowlege of procedural directions for tripping the reactor.

K/A:

068AK2.02 – Knowledge of the interrelations between the Control Room Evacuation and the following: Reactor trip system.

OBJECTIVE:

RO4-03-LPD21.004 - **DISCUSS** the required actions prior to evacuating the Control Room.

RO4-03-LPD21.005 - **DISCUSS** the actions performed by each Control Room member upon reaching their respective areas.

REFERENCES:

E-O-06, Rev. Z, Step 2 and 13.b (Control Operator B action).

SOURCE:

New

JUSTIFICATION:

The indications for a reactor trip in E-O-06 are,
* Reactor Trip and Bypass Breakers OPEN
* Neutron flux decreasing
With the given indications neither of these conditions are met. None of the other steps performed before leaving the Control Room address the reactor status. The first step after evacuating the Control Room for Control Operator B is if the turbine did not trip, manually trip the turbine, and if the reactor did not trip, locally trip both Reactor Trip Breakers.

A: Emergency boration is an appropriate action if the reactor has not tripped OR if all rods are not inserted. However, this is not directed except in ES-0.1.

B: Establishing Cold Shutdown boron concentration is the next step performed by Control Operator A at step 33, following aligning of the Dedicated Shutdown Panel. This would be performed after the Control Operator B opens the breakers.

D: Opening the breakers supplying Bus 33 and 43 are appropriate in E-O-07, when Control Room is not evacuated.

74. Given the following:

- A Small Break LOCA has occurred.
- Due to a failure of Voltage Restoration for Buses 1 and 2, these buses are deenergized.
- The actions of ES-1.2, Post LOCA Cooldown And Depressurization, are in progress.
- Charging Pumps A and B are running with suction aligned to the RWST.
- Both RHR Pumps are stopped in AUTO.
- Both SI Pumps are running.
- The crew is ready to depressurize the RCS to refill the Pressurizer.

How will this depressurization be achieved?

A. Utilize BOTH Pressurizer Spray Control valves, PS-1A AND PS-1B, to spray down the Pressurizer steam space.

B. Open BOTH Pressurizer PORVs, PR-2A AND PR-2B to vent the Pressurizer.

C. Utilize Pressurizer Auxiliary Spray Valve, CVC-15, to spray down the Pressurizer steam space.

D. Open ONE Pressurizer PORV, PR-2A or PR-2B to vent the Pressurizer.

Answer: D
COGNITIVE LEVEL:

3-SPK - Use knowlege of procedural directions and current systems status to determine the appropriate method for depressurizing the Przr.

K/A:

W/E03EA1.1 – Ability to operate and/or monitor the following as they apply to the LOCA Cooldown and Depressurization: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

OBJECTIVE:

RO4-04-LP019.003 - Given a set of plant conditions **RECOMMEND** the appropriate procedural action to be taken while implementing ES-1.2, “Post LOCA Cooldown and Depressurization”.

REFERENCES:

ES-1.2, Rev. Q, Step 9

SOURCE:

New

JUSTIFICATION:

A: This is the "normal" method used to depressurize the RCS. However, with Buses 1 and 2 deenergized, the RXCPs are NOT running and are therefore unable to provide the driving head for normal sprays.

B: Opening TWO PORVS is not an appropriate action. This action has a less stable depressurization rate and raises the probability of a PORV failing to close.

C: CVC-15 is a method for depressurization and for a SGTR is utilized as the third method. However, it is not used in this case since the requirements place a limit of spray dT, and letdown is required to be in service if Aux Spray is to be used.

75. Given the following:

- A LOCA that resulted in significant core damage occurred 1.5 hours ago.
- Containment radiation levels rose to a peak of 450,000 R/hr at 20 minutes into the event, and have just decreased to 50,000 R/hr.
- Peak containment pressure was 9.2 psig and has been lowering since the peak to the current value of 3.5 psig.

Which of the following describes the effect of containment parameters on implementation of the IPEOPs?

ADVERSE CONTAINMENT conditions ...

A. exist due to the current containment pressure.

B. exist due to the current containment radiation dose rate.

C. previously existed because of containment radiation levels and pressure, and must still be used until the integrated dose has been evaluated.

D. previously existed because of containment radiation levels and pressure, but are NO longer required because of the limited integrated dose and pressure reduction.

Answer: C
COGNITIVE LEVEL:

1-P - Procedure step for use of ADVERSE CONTAINMENT values.

K/A:

W/E16EA2.2 – Ability to determine and interpret the following as they apply to the High Containment Radiation: Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments

OBJECTIVE:

RO4-04-LP001.007 - **EXPLAIN** the requirements of IPEOP implementation in accordance with UG-0.

REFERENCES:

UG-0, Rev. F, 6.10.3 and 6.10.5.
E-0 QRF, Rev. I, NOTE

SOURCE:

Cook 1 12/2002 NRC Exam (INPO Bank)

JUSTIFICATION:

A: Containment pressure > 4 psig is the setpoint for ADVERSE CONTAINMENT. If pressure falls to below setpoint then the ADVERSE CONTAINMENT does not apply (for Cntmt pressure only)

B: Containment radiation > 10^{+05} R/HR is the setpoint for ADVERSE CONTAINMENT. If radiation falls below this value, then evaluation for integrated dose (< 10^{+6} R) must be performed before ADVERSE CONTAINMENT values do not apply

D: The actual values for ADVERSE CONTAINMENT are true; however, the evaluation of integrated dose must occur before the use of ADVERSE CONTAINMENT values can be dropped.

1. Given the following:

   - The plant was stable with reactor power at 100%.
   - A reactor trip and safety injection occurred due to a Pressurizer PORV failing open and remaining full open.
   - All safeguards equipment has responded per design.
   - The crew implemented E-0 and transitioned to E-1, Loss of Reactor or Secondary Coolant.
   - The failed PORV has just been isolated.
   - The crew is currently performing Step 12 of E-1, "Check If SI Should Be Terminated".

Which combination of SI Termination Criteria is expected to be satisfied at this point?

A. Pressurizer level AND secondary heat sink

B. RCS subcooling AND secondary heat sink

C. Pressurizer level AND RCS subcooling

D. Pressurizer pressure AND RCS subcooling

Answer: A
COGNITIVE LEVEL:

3-PEO - Predict the SI termination parameters that would be satisfied based on the event (open PORV).

K/A:

008AA2.23– Ability to determine and interpret the following as they apply to the Pressurizer Vapor Space Accident: Criteria for throttling high-pressure injection after a small LOCA.

OBJECTIVE:

RO4-04-LP018.003: Given a set of plant conditions **RECOMMEND** the appropriate procedural action to be taken while implementing E-1, "Loss of Reactor or Secondary Coolant".

RO4-04-LP044.004: Using INPO Document INPO 88-008, Three Mile Island Accident, summarize the event, relate how these events could occur at KNPP, and explain the actions that could be taken to prevent the event from occurring.

REFERENCES:

E-1, Rev. Q, Step 12
INPO 88-008, Material For A Case Study On The Three Mile Island Unit 2 Accident, pages 7-10

SOURCE:

Prairie Island 5/2000 NRC exam (INPO Bank)

JUSTIFICATION:

Przr level would have risen due to the location of the leak in the Przr steam space. Secondary heat sink remains unaffected by the event and would have established satisfactory level.

B: Subcooling would be expected to be low since RCS pressure would be very low due to the PORV opening. Secondary heat sink would be satisfied.

C: Przr level would be satisfactory due to ECCS (SI) flow and because of the location of the leak. Subcooling would be expected to be low since RCS pressure would be very low due to the PORV opening.

D: Przr pressure would be expected to be off-scale low at 1700 psig. RCS pressure would be low due to the PORV opening. Subcooling would be expected to be low since RCS pressure would be very low due to the PORV opening.

2. Given the following:

- The plant was operating at 100% power when a loss of offsite power occurred.
- The crew responded to the trip by entering E-0, Reactor Trip Or Safety Injection.
- At Step 4, CHECK SI STATUS, it was noted that neither 47021-A, SI TRAIN A ACTUATED, NOR 47021-B, SI TRAIN B ACTUATED are in alarm.
- The following conditions are noted:
  47022-D, CONTAINMENT HIGH PRESSURE SI is lit.
  47031-Q, CONTAINMENT SUMP A LEVEL HIGH is lit.
  47042-F, PRESSURIZER LVL LETDOWN ISOL & HEATERS OFF is lit.
  47043-J, CHARGING PUMP IN AUTO HIGH/LOW SPEED is lit.
  47051-C, CNTMT EMER DISCH DMPRS ACTIVATED is lit.
  Safeguards Status light 440908-0901 (RED) CONTAINMENT 4 PSIG is lit.
  Safeguards Status light 440908-0902 (WHITE) CONTAINMENT 4 PSIG is NOT lit.
  Safeguards Status light 440908-0904 (YELLOW) CONTAINMENT 4 PSIG is lit.

What is the proper action for this condition?

A. Transition to ES-0.1, Reactor Trip Response, since NO SI is actuated or required.

B. Transition to ES-0.2, Natural Circulation Cooldown, since power was lost to both RXCPs.

C. Transition to ES-0.0, Rediagnosis, and verify the status of Safety Injection.

D. Manually actuate Safety Injection and continue in E-0.

Answer: D
COGNITIVE LEVEL:

3-SPK - Solve a problem using the knowledge of the logic for SI actuation and the alarm indications that provide for plant status.

K/A:

009E 2.4.46 – Small Beak LOCA; Ability to verify that the alarms are consistent with the plant conditions.

OBJECTIVE:

RO4-04-LP002.009: Given a set of plant conditions Recommend the appropriate procedural action to be taken while implementing E-0, Reactor Trip or Safety Injection.

REFERENCES:

E-1635, Rev.Q.
47022-D, Rev. A
47051-C, Rev. C, NOTE
E-0, Rev. X, Step 4 (Immediate Operator Action)
BKG E-0, Rev. B, Section 1.3 and 3.1

SOURCE:

New

JUSTIFICATION:

The Containment pressure input to SI is based on three of the six Containment pressure channels. Actuation is expected to occur when two of the three channels are actuated (lit). A condition beyond the loss of offsite power is indicated by the alarm status, leakage into containment is suspected with Sump levels rising and Przr level below isolation value. Also the proper actuation of Containment SI signal generation is indicated. However, the SI signal has not been generated. The action would be to verify containment pressure value above setpoint and manually actuate SI. The action then continues in E-0.

A: Transition to ES-0.1 would be expected if the indications did not confirm that SI is required. With 2/3 pressure channels above their setpoint, manual actuation of SI is required.

B: Transition to ES-0.2 may occur if cooldown is required without restoring power to RXCPs (Buses 1 and 2); however, this occurs from ES-0.1 not directly from E-0.

C: Transition to ES-0.0 is not performed at this step but following diagnosis where SI is actuated or required. IPEOP ES-0.0, REDIAGNOSIS, is entered based on operator judgment, most likely when there is doubt in his mind that he is in the correct procedure. The applicability of the procedure is limited to those cases when 1) safety injection is in service or is required and 2) E-0 has been executed and a transition has been made to another Optimal Recovery Procedure.

AS ADMINISTERED SRO EXAMINATION

3. Given the following:

- The plant has recently shutdown.
- RCS boron concentration is 2550 ppm.
- RCS temperature was stable at 110ºF.
- Control rod unlatching is underway.
- RHR Train A was in service for cooling, but the pump has tripped, and neither RHR pump will start.
- RCS temp has risen to 190ºF and is slowly rising.

Identify the initial plant MODE, the current plant MODE and any notifications required using the attached GNP-11.08.04, Table 1, Reportability Determination Matrix and EPIP-AD-02, Table 2-1, Emergency Action Level Charts.

A. Initially the plant was in COLD SHUTDOWN MODE and is still in COLD SHUTDOWN.
NO notification is required.

B. Initially the plant was in REFUELING MODE and is now in COLD SHUTDOWN.
NO notification is required.

C. Initially the plant was in REFUELING MODE and is now in COLD SHUTDOWN.
1-hour notification to the NRC is required since an an ALERT Emergency Classification is declared due to both RHR pumps failed.

D. Initially the plant was in COLD SHUTDOWN MODE and is now in INTERMEDIATE SHUTDOWN.
8-hour notification to the NRC is required since both RHR pumps have failed.

Answer: C
COGNITIVE LEVEL:

3-SPK/SPR - Determine the change in plant MODE based on identified parameters and determine the required off-site notifications required by these conditions.

K/A:

025A 2.4.46 – Loss of Residual Heat Removal System (RHRS); Ability to determine Mode of Operation.

OBJECTIVE:

ROI-01-LPTS2.001: DESCRIBE the definition of terms defined in Section 1 of the Kewaunee Technical Specifications.
j. Modes

RO4-01-LPA19.001: While performing the duties of the Shift Manager, EVALUATE a set of hypothetical plant conditions with respect to submitting reports in accordance with 10 CFR 50.72.

REFERENCES:

KNPP Technical Specification, 1.0.j, Amend No. 172 page TS 1.0-4
GNP-11.08.04, Rev. D, Table 1
EPIP AD-02, Rev. AL, Table 2-1, Chart F.

PROVIDED REFERENCE: GNP-11.08.04, Table 1 Reportability Determination Matrix
PROVIDED REFERENCE: EPIP-AD-02, Table 2-1, Emergency Action Level Charts

SOURCE:

New

JUSTIFICATION:

A: With RCS Tave below 140ºF the plant is in REFUELING MODE. COLD SHUTDOWN exists between 140ºF and 200ºF. With loss of both trains of RHR at less than 200ºF, the Emergency Plan implementation results in a declaration of an ALERT and associated 1-hour notification to the NRC.

B: The change in MODE identification is correct. However, with loss of both trains of RHR at less than 200ºF, the Emergency Plan implementation results in a declaration of an ALERT and associated 1-hour notification to the NRC.

D: The change in MODE is incorrect since INTERMEDIATE SHUTDOWN is not reached until RCS Tave is above 200ºF. A 1-hour notification is required also due to ALERT classification. (8-hour notification criteria may be appropriate also.)

4. Given the following:

- A rupture in SG A has been diagnosed.
- The crew is performing the actions of E-3, Steam Generator Tube Rupture.
- RCS pressure and SG A pressure have been stabilized at 850 psig.

- Subcooling is 65ºF.
- Pressurizer level is being maintained at 54% with charging and letdown in service.
- SG A narrow range level is 68% and stable.
- SG B narrow range level is 10% and stable.
- CST levels are at 35% and makeup to the CSTs CANNOT be established.
- Management has directed that the cooldown and depressurization be conducted as quickly as possible.

What procedure will be used in this condition?

A. ECA-3.2, SGTR With Loss of Reactor Coolant – Saturated Recovery Desired.

B. ES-3.1, Post-SGTR Cooldown Using Backfill.

C. ES-3.2, Post-SGTR Cooldown Using Blowdown.

D. ES-3.3, Post-SGTR Cooldown Using Steam Dump.

Answer: D
COGNITIVE LEVEL:

2DR - Determine the relationship between the current plant conditions and desired operation, and the procedure that addresses these issues.

K/A:

038E 2.4.1 – SGTR; Knowledge of EOP entry conditions and immediate action steps.

OBJECTIVE:

RO4-04-LP029.007: DISCUSS the following items as they relate to ES-3.3, “Post-SGTR Cooldown Using Steam Dump”:
a. Entry Conditions

RO4-04-LP029.009: Given a set of plant conditions RECOMMEND the appropriate procedural action to be taken while implementing ES-3.3, “Post-SGTR Cooldown Using Steam Dump”.

REFERENCES:

E-3, Rev. Y, Step 43
ES-3.3, Rev. O, 2.1.a (Entry Condition)

BKG ES-3.3, Rev. B, Section 3, 1st paragraph.

SOURCE:

KNPP Bank RO4-04-LP029.002

JUSTIFICATION:

A: ECA-3.2 is warranted if pressurizer level cannot be maintained or if RCS subcooling cannot be maintained. These conditions do not exist.

B: ES-3.1 is the preferred transition if there is time allowable for cooldown with limited waste inventory. The primary benefit of this method is the limited secondary side contamination and waste.

C: ES-3.2 is an alternative method for cooldown time allowable while minimizing secondary side contamination. It is limited by the amount of blowdown flow that can be developed form the affected SG.

5. Given the following:

- The plant was operating normally at 100% just prior to a refueling outage (End of Cycle).
- Both Main Feedwater Pumps are stopped.
- The following conditions are noted 60 seconds after the Feed Pumps stopped (NO operator action has been taken):

  - SG A Wide Range level is 35% and lowering.
  - SG B Wide Range level is 45% and stable.
  - MS-1A, MSIV A, is open.
  - MS-1B, MSIV B, is closed.
  - Buses 1 and 2 are deenergized.
  - The Power Range NIs read between 2% and 4%.
  - Intermediate Range SUR reads -0.5 dpm.
  - All AFW pumps are running.
  - RCS Tave is 500ºF and lowering.

Based on the plant indications, the ________(1)________ has NOT tripped and accident analysis limits ________(2)________ be violated.

(1)	(2)
A. turbine	will NOT
B. turbine	will
C. reactor	will NOT
D. reactor	will

Answer: A
COGNITIVE LEVEL:

3SPK - Determine the plant condition based on the the given conditions and determine if the Safety Analysis accident limits are exceeded.

K/A:

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

OBJECTIVE:

RO4-04-LP008.001: **Discuss** the following items as they relate to FR-S.1, Response to Nuclear Power Generation/ATWS
a. Entry Conditions
b. High Level Action Summary Steps

RO4-04-LP029.009: **Summarize** the purposes or basis of the following items as they relate to FR-S.1, Response to Nuclear Power Generation/ATWS.
e. All procedure steps.

REFERENCES:

BKG FR-S.1, Rev. C, 2.2 pages 8

SOURCE:

Prairie Island 2003 NRC Exam (INPO)
Modified
Added time in life to premise to change correct answer
In selections, changed "may" to "will" to remove ambiguity.
Adjusted parameters to reflect KPS plant instrumentation and response.

JUSTIFICATION:

B: The projected RCS pressure remains below that assumed in the accident analysis for loss of feed event with ATWS.

C: The reactor is tripped since RCS Tave has trended down and NI power is consistent with the cooldown.

D: The reactor is tripped since RCS Tave has trended down and NI power is consistent with the cooldown.

6. Given the following:

- The plant is at 100% power.
- Surveillance Test SP-42-312B, Diesel Generator B Availability Test, is in progress with DG B running.
- Component Cooling Pump B trips on overcurrent.
- BRB-104, ckt 10 supplying DG B tripped open and CANNOT be closed.

What is the effect of this condition?

A. DG B is inoperable and must have its fuel supply locally isolated.
A plant shutdown must commence within one hour using the Standard Shutdown Sequence (Technical Specification 3.0.c).

B. DG B is Degraded but Operable and can be controlled locally.
A 24-hour LCO is applicable for restoration of the DC Distribution System and a 72-hour LCO is applicable for restoration of Component Cooling Pump B.

C. DG B is inoperable and must have its fuel supply locally isolated.
A 72-hour LCO is applicable for restoration of Component Cooling Pump B and a 7-day LCO is applicable for restoration of DG B.

D. DG B is Degraded but Operable and can be controlled locally.
Only the 72-hour LCO is applicable for restoration of Component Cooling Pump B.

Answer: C
COGNITIVE LEVEL:

3-PEO - Predict the affect of a loss of DC power to DG and apply to Tech Spec LCO.

K/A:
058AA2.03 – Ability to determine and interpret the following as they apply to the Loss of DC Power: DC loads lost; impact on to operate and monitor plant systems.

OBJECTIVE:

RO4-03-LPD14.006: In accordance with E-EDC-38B, “Loss of B Train Safeguards DC Power”, Summarize the subsequent operator actions that are necessary to respond to an abnormal condition on “B” Train Safeguards

DC Power.

RO2-03-LP038.002: DESCRIBE the DC and Emergency AC Electrical Distribution System. Include the following in the description;
3. Interfaces with the following plant systems:
   d. Diesel generator (DG) control

RO2-03-LP42A.002: DESCRIBE the Emergency Diesel Generator and TSC Diesel Generator Systems, Include the following in the description;
3. Interfaces with the following plant systems:
   e. DC and Emergency AC.

RO2-03-LP42A.007: EXPLAIN the LCO operation, applicability and action requirements for each of the following Technical Specifications associated with the Emergency Diesel Generator System:
b. 3.7.b.2.

REFERENCES:

E-EDC-38B, Rev. F, 4.4.2
KNPP Technical Specification 3.3.d.2, Amend No.116, page TS 3.3-6
KNPP Technical Specification 3.7.b.2 & 3.7.c, Amend No. 122, page TS 3.7-2
E-844, Rev. BX

SOURCE:

New

JUSTIFICATION:

OBD - (Operable But Degraded)

A: The DG is inoperable, because in order to control the DG, it must be shutdown by isolating the fuel supply valves from the day tank. Since the affected CC Pump is in the same Train as the DG, the requirements of TS 3.7.b.2 and 3.7.c are met therefore the LCO statement for one CC Pump inoperable applies. This action is considered since if the affected CC Pump had been Train A, applying TS 3.7.c, for CC Pump B, its normal power source is operable (via RAT), but the redundant train CC (CC Pump A) is NOT operable; and therefore, CC Pump B cannot be considered operable. The crew would apply TS 3.0.c.

B: The DG is inoperable, because in order to control the DG, it must be shutdown by isolating the fuel supply valves from the day tank. If the DG were OBD, then the CC Pump LCO would apply; however, the DC system is NOT inoperable just because the power is lost to a single component. If the supply breaker to BRB-104 from BRB-102 (Battery Charger and Battery supply lost) was tripped then the Train B Emergency DC would be inoperable.

D: The DG is inoperable, because in order to control the DG, it must be shutdown by isolating the fuel supply valves from the day tank. If the DG were OBD, then this would be correct. The CC Pump LCO would be the only action applicable.

7. Given the following:

- The plant is at 100% power.
- A Control Bank B control rod stationary gripper fails.
- The control rod is inserted into the core 52 steps from fully withdrawn.
- TLA-1 ROD SUPERVISION ALARM, is in alarm.

How is the affected rod INITIALLY identified, and what actions are required to be performed for this condition?

The location of the rod may be determined by observing. . .

A. an abnormal DECREASE in ALL power range detectors.
If the rod CANNOT be recovered, power is reduced to less than 85% within TWO hours and less than 50% power within the next FOUR hours since the rod CANNOT be restore within alignment limits.

B. an abnormal INCREASE in ONLY ONE power range detector.
If the rod CANNOT be recovered, within ONE hour action must be taken to be in HOT STANDBY within SIX hours since Rod Insertion Limits CANNOT be restore within limits.

C. a localized INCREASE in the CET temperature nearest the affected fuel assembly.
If the rod CANNOT be recovered, power is reduced to less than 50% power within the FOUR hours since the rod CANNOT be restore within alignment limits.

D. a localized DECREASE in the CET temperature nearest the affected fuel assembly.
If the rod CANNOT be recovered, within ONE hour action must be taken to be in HOT STANDBY within SIX hours since Rod Insertion Limits CANNOT be restore within limits.

Answer: D
COGNITIVE LEVEL:

2-RI - Identify the plant parameters that would identify the misposition rod and based on given conditions determine the proper Technical Specification action.

K/A:

058AA2.03 – Ability to determine and interpret the following as they apply to the Inoperable/Stuck Control Rod: Interpretation of computer in-core TC map for dropped rod location

OBJECTIVE:

RO4-03-LPD14.006: **SUMMARIZE** the subsequent operator actions In accordance A-CRD-49, “Abnormal Rod Control System Operation” that are necessary to respond to the following:
a. Stuck Rod

RO2-03-LP038.002: **EXPLAIN** the LCO operation, applicability and action requirements for Technical Specifications associated with the Rod Control and Rod Position Indication System
- TS 3.10.d.
- TS 3.10.e
- TS 3.10.g

REFERENCES:

A-CRD-49, Rev. N, Step 7
A-CRD-49E, Rev. D, 4.3
Reactor Data Manual, RD-12.1, Rev. April 5, 2003
KNPP Technical Specification 3.10.d.2, Amend No.165, page TS 3.10-4
KNPP Technical Specification 3.10.e, Amend No.181, page TS 3.10-5
TRM 2.1, Rev. 6, COLR Cycle 27, Rev. 1, Figure 4

SOURCE:

New

JUSTIFICATION:

The Rod Insertion limits are NOT met if Bank B is positioned at 174 steps withdrawn. With Bank B rod affected, boration will not allow establishing rod above RIL since the ZERO power RIL is at 194 steps on Bank B.

A: While the decrease in PR NIS may be used (typically in one or two quadrants), a drop in all indications is NOT expected since the rod is not a centerline rod. The action is not correct for two reasons: 1) Action must be initiated within one hour, and 2) if RIL was not affected the initial power limit of 85% is only required after 4 hours.

B: The increase in one quadrant is the inverse of what is expected for a dropped/misaligned rod. The NI in the same quadrant as the affected rod reads lower and the remaining read higher. The action is correct for exceeding RIL

C: Localized increase in CET will not identify the location. The CET will be lowest in the area of the affected rod and the remainder of the CETs will globally read slightly higher. The action is incorrect in that action is required within 1 hour. Reduction to less than 50% would be required if misalignment still existed and RIL was restored.

8. Given the following:

- There has been a fire necessitating the evacuation of the control room.
- E-0-06, Fire in Alternate Fire Zone, has been entered from E-FP-08, Emergency Operating Procedure - Fire.
- A cooldown and depressurization has been started.
- The cooldown rate has been sustained at 24ºF/hr over the past 7 hours.
- Pressurizer pressure is 1550 psig and lowering.
- RCS Loop A Hot Leg temperature is 551ºF.
- RCS Loop A Cold Leg temperature is 400ºF.
- Pressurizer Cold Cal level is 35% and stable.
- SG A WR Level is 72% and stable.

If Control Operator A reports a rapid rise in Pressurizer level to 45%, what direction should be provided using the attached E-O-06 references?

A. Reduce Charging Pump C speed.

B. Open SD-3A, SG A PORV.

C. Close CVC-15, Aux Spray valve.

D. Place Pressurizer Heater backup Group 1A to OFF.

Answer: C
COGNITIVE LEVEL:

3-PEO - Predict the corrective action necessary to resolve a given condition

K/A:

068AA2.07 – Ability to determine and interpret the following as they apply to the Control Room Evacuation: PZR level.

OBJECTIVE:

RO4-03-LPD21.05 - **DISCUSS** the actions performed by each Control Room member upon reaching their respective areas.

REFERENCES:

E-O-06, Rev. Z, Steps 48 & 49, Table E-O-06-1 & Figure E-O-06-1

PROVIDED REFERENCES: Table E-O-06-1, Reactor Coolant System Subcooling
Figure E-O-06-1, Cooldown Operating Region

SOURCE:

New

JUSTIFICATION:

Using the table we see that 50ºF subcooling temperature is 552ºF. The rise in Przr level is likely due to voiding in the reactor vessel head.
Using the chart we see that we are very close to the upper allowed value for temperature-pressure. Thus if we continue to depressurize it is likely we will cross the upper limit. Action should be to stop the depressurization by closing CVC-15.

A: Adjusting the Charging Pump speed will only affect the level in the Przr. Currently the band is 20% to 50% Cold Cal Level. This is satisfied.

B: Opening SD-3A will increase the cooldown rate and exceed the 25ºF limit. With the Hot Leg temperatures being high, this option should be considered.

D: Placing the BU Heater Group to off will only allow Przr pressure to decrease more rapidly. This will result in exceeding the upper limit.

9. Given the following:

- Plant is at 80% power.
- Chemistry reports the RCS coolant gross radioactivity is above 91/E mCi/cc.

What is the required course of action?

A. Immediately shutdown following the standard shutdown sequence.

B. Continue operations for 48 hours, then proceed to HOT SHUTDOWN using normal operating procedures.

C. Be in at least INTERMEDIATE SHUTDOWN with an average coolant temperature less than 500ºF within six hours.

D. Continue operation for 800 hours with sampling and analyzing every 4 hours, then shutdown using normal procedures.

Answer: C
COGNITIVE LEVEL:

1-P - Identify Tech Spec limits on radioactivity are exceeded and identify the TS ACTION.

K/A:

058AA2.03 – High Reactor Coolant Activity: Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications.

OBJECTIVE:

RO2-01-LP362.007 - **EXPLAIN** the LCO operation, applicability and action requirements for each of the following Technical Specifications associated with the Reactor Coolant System:
6. Maximum Coolant Activity (T.S. 3.1.c)

REFERENCES:

KNPP Technical Specification 3.1.c.2.B, Amend No. 167, page TS 3.1-7

SOURCE:

JUSTIFICATION:

A: The standard shutdown sequence applies for conditions in which NO Tech Spec ACTION exist for the (TS covered system/component/structure) condition. The ACTION provides for RCS conditions if the value for I-131 exceeds 60 microCi/gm.

B: The ACTION for allowing 48 hours continuous operation applies if the actual I-131 activity exceed the LCO value of 1.0 microCi/gram AND is less than/equal to 60 microCi/gm. After condition persist for the 48 hour period, then shutdown and cooldown below 500ºF is required.

D: The sampling schedule and total time allowed does not apply for coolant activity.

10. Given the following:

- A LOCA has occurred.
- The crew is currently performing E-1, Loss of Reactor or Secondary Coolant.
- The STA reports a RED path for F-0.4, Integrity.

Which of the following identifies the parameter the STA used to make his determination and the reason it was used?

A. Tcold temperature since it most closely reflects the temperature in the beltline region of the reactor vessel.

B. Tcold temperature since it most closely reflects the temperature in the core.

C. Incore thermocouple temperature since it most closely reflects the temperature in the beltline region of the reactor vessel.

D. Incore thermocouple temperature since it reflects the temperature in the core.

Answer: A
COGNITIVE LEVEL:

1-F/B - Identify the reason the Tcold parameter is used for evaluation of PTS condition.

K/A:

W/E08 2.1.14 – Pressurized Thermal Shock: Knowledge of system status criteria which require the notification of plant personnel.

OBJECTIVE:

RO4-04-LP016.004 - Given an Imminent Pressurized Thermal Shock Condition, **EXPLAIN** the basis for actions taken, per FR-P.1 background document.

RO4-04-LP016.004 - **EXPLAIN** the purpose of procedure “FR-P.1, Response to Imminent Pressurized Thermal Shock.”

REFERENCES:

F-0.4, Rev. E

BKG FR-P.1, Rev. C, 2. Pressurized Thermal Shock Events, page 4.

SOURCE:

New

JUSTIFICATION:

B: Tcold should be well below the average temperature of the core since decay heat is still being added by the fuel. This could be considered since it is the largest volume in the vessel. but does not directly interface with the vessel beltline (due to the core barrel).

C: CET temperatures do not reflect the temperature of the vessel beltline but rather that at the top of the fuel.

D: CET temperatures do better reflect the temperature of the core area, but has little to do with the mechanism of PTS at the vessel beltline.

11. Given the following:

- A loss of off-site power has occurred.
- The reactor and turbine are tripped
- Reactor Trip Breaker A did NOT open, and could NOT be opened locally.
- Natural circulation conditions have been established.
- Annunciator 47054-A, CRDM COOLING FANS OFF, is lit.

Which of the following RCS Pressure and Cold Leg Wide Range Temperature relationships is acceptable for a Natural Circulation cooldown in accordance with ES-0.2, Natural Circulation Cooldown under the above conditions?

(Figure ES-0.2-1 and Figure ES-0.2-2 provided)

A. 1600 psig and 425ºF.

B. 650 psig and 400ºF.

C. 450 psig and 180ºF.

D. 360 psig and 375ºF.

Answer: D
COGNITIVE LEVEL:

3-SPR - Determine which condition is acceptable using the correct Figure for given conditions.

K/A:

022 2.4.46 – Containment Cooling System (CCS): Ability to verify that the alarms are consistent with the plant conditions.

OBJECTIVE:

RO2-04-LP018.004 - **DESCRIBE** the operation, automatic actions, interlocks, instrumentation, controls and alarms associated with the Reactor Building Ventilation System and the following major system components:
5. CRDM Cooling Fan A (B) and Damper RBV-111A (B)

RO4-04-LP006.003 - Given a set of plant conditions **Recommend** the appropriate procedural action to be taken while implementing ES-0.2, Natural Circulation Cooldown.

AS ADMINISTERED SRO EXAMINATION

REFERENCES:

47054-A, Rev. Orig, Comment 1.
ES-0.2, Rev. P, step 13, Figures ES-0.2-1 & ES-0.2-2.
BKG E-0.2, Rev. Orig, Step 13 BASIS

PROVIDED REFERENCE:

Figure ES-0.2-1, Cooldown Operating Region - With Full CRDM Cooling
Figure ES-0.2-2, Cooldown Operating Region - Without Full CRDM Cooling

SOURCE:

KNPP Bank Question 12/11/2000 NRC Exam

JUSTIFICATION:

By the alarm indication both CRDM fans are in OFF. With no CRDM fans running, the depressurization and cooldown are limited to the Acceptable Operating Region For Cooldown of Figure ES-0.2-2. Only the value of 360 psig and 375ºF falls within that region. Also note that if Figure ES-0.2-1 is used for evaluating this condition, the locus falls outside the Acceptable Operating Region.

A: This is acceptable if only if BOTH CRDM fans are running using Figure ES-0.2-1.

B: This is NOT acceptable using either Figure.

C: This is NOT acceptable using either Figure.

12. Given the following:

- A LOCA has occurred.
- ES-1.3, Transfer To Containment Sump Recirculation, has just been entered.

How do the actions taken with respect to the Containment Spray System (ICS) affect the RWST radiation levels?

During the injection phase, RWST radiation levels will...

A. remain the same since the ICS Pumps and ECCS pumps are taking suction from the RWST.
Following completion of ES-1.3, radiation levels remain the same even with recirculation flow from RHR Pump since ICS-201 and ICS-202, ICS Recirculation to RWST, were closed by the Containment Isolation signal.

B. remain the same since the ICS Pumps and ECCS pumps are taking suction from the RWST.
Following completion of ES-1.3, the RWST radiation levels will rise due to the addition of recirculation flow from the RHR Pump, and flow through ICS-201 and ICS-202, ICS Recirculation to RWST.

C. rise since ICS recirculation to the RWST is maintained and caustic solution is added to the ICS Pump suction.
Following completion of ES-1.3, the RWST radiation level will stabilize since ICS-201 and ICS-202, ICS Recirculation to RWST, are manually closed during alignment.

D. rise since ICS recirculation to the RWST is maintained and caustic solution is added to the ICS Pump suction.
Following completion of ES-1.3, the RWST radiation levels will continue to rise due to the addition of recirculation flow from the RHR Pump to the ICS system, and flow through ICS-201 and ICS-202, ICS Recirculation to RWST.

Answer: A
COGNITIVE LEVEL:

2-RI - Describe the relationship between operation of the ECCS and ICS, and the alignment provided by the ESF Actuation signals. Relate these factors to the expected condition for RWST.

K/A:

026A2.09 – Ability to (a) predict the impacts of the following malfunctions or operations on the CSS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those: Radiation hazard potential of BWST.

OBJECTIVE:

RO2-01-LP023.004 - **DESCRIBE** the operation, automatic actions, interlocks, instrumentation, controls, and alarms associated with the ICS System and the following major system components:
1. Refueling Water Storage Tank.

RO4-04-LP021.002 - **Summarize** the purposes or basis of the following items as they relate to ES-1.3, Transfer to Containment Sump Recirculation.
e. All procedure steps.

REFERENCES:

BKG ES-1.3 Rev. C, 2, pages 3-4; 3.1.2; Step 26.
OPERM-217, Rev. AP
E-1604, Rev. W

SOURCE:

New

JUSTIFICATION:

B: This would be the correct if the ICS recirc to RWST was not isolated. It should be noted that these valves are not credited as containment isolation valves but do get this ("T") signal.

C: The addition of the caustic to ICS fluid does not impact the radiation levels of RWST (but does affect chemistry). One of the purposes of the caustic addition is to strip Iodine from containment atmosphere and maintain it in solution. This action is applicable to the SI Pump recirc valves SI-208 and SI-209 (Step 9).

D: The addition of the caustic to ICS fluid does not impact the radiation levels of RWST (but does affect chemistry). One of the purposes of the caustic addition is to strip Iodine from containment atmosphere and maintain it in solution. This would be the correct if the ICS recirc to RWST was not isolated.

AS ADMINISTERED SRO EXAMINATION

13. What is the reason for the Feedwater Isolation signal generated from High-High SG level?

A. Preclude excessive SG tilts due to cooler feedwater supplied to ONE SG.

B. Prevent overfill of the SG that may result in damage to secondary components.

C. Ensure containment pressure remains within maximum internal pressure limit with the affected SG faulted inside containment.

D. Protect the Feedwater Pumps from operating in runout condition with FW-7A/B, S/G A/B Main Valve, fully open.

Answer: B
COGNITIVE LEVEL:

1-B - Identify the basis for the FW Isolation signal

K/A:

059 2.2.22 – Main Feedwater (MFW) System: Knowledge of limiting conditions for operations and safety limits.

OBJECTIVE:

RO2-02-LP05A.006 - **IDENTIFY** the Main Feedwater System components with Technical Specification requirements.

RO2-05-LP055.008 - **EXPLAIN** the basis for each of the following Technical Specifications associated with the Engineering Safety Features System:
1. TS 3.5, Instrumentation System

REFERENCES:

KNPP Technical Specification Basis Main FW Isolation, page TS B3.5-2.

SOURCE:

New

JUSTIFICATION:

A: During normal operation the excessive feeding of one SG would result in the condition of a SG Tilt, as indicated on TLA-10. The action for this TLA is to verify correct operation of FW-7A/B. As indicated the alarm shows improper load balance between SGs.

C: This item is addressed under the Setting Limits for containment ESFAS response. The SG isolation is not set based upon the SG water volume from the accident analysis. However, this would have to be considered since the volume of water in the SG would impact the Containment pressure with a faulted SG. The accident analysis assumes SG level at program with ZERO power inventory and 100%-rated FW flow. The isolation of FW results from the SI actuation signal.

D: Concern for pump runout is not related to main Feedwater, but is a concern for AFW Pumps with the control valve AFW-2A/B controller failure.

14. Given the following:

    - The plant is at 100% power.
    - The EO reports AFW Pump B discharge piping is warm to the touch.

Which of the following describes the impact of main feedwater system backleakage over the next FOUR hours while addressing this condition?

A. Reactor power can be maintained at 1772 MWt.
During the time action is taken to reseat AFW-4B check valve, only AFW Train B is declared inoperable.

B. Reactor power can be maintained at 1772 MWt.
During the time action is taken to reseat AFW-4B check valve, AFW Train B and the Turbine-Driven AFW are inoperable.

C. Within TWO hours, reactor power must be reduced to or below 1673 MWt.
During the time action is taken to reseat AFW-4B check valve, only AFW Train B is declared inoperable.

D. Within TWO hours, reactor power must be reduced to or below 1673 MWt.
During the time action is taken to reseat AFW-4B check valve, AFW Train B and the Turbine-Driven AFW are inoperable.

Answer: D
COGNITIVE LEVEL:

2-DR - Identify the relationship between the actions to reseat the check valve (isolation of the AFW headers) and pump operability, and allowable thermal power level for inoperable AFW train.

K/A:

061A2.06 – Ability to (a) predict the impacts of the following malfunctions or operations on the AFW System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Back leakage of MFW.

OBJECTIVE:

RO2-02-LP05B.0067: **EXPLAIN** the LCO operation, applicability and action requirements for each of the following Technical Specifications associated with the Auxiliary Feedwater System

1. Technical Specifications 3.4.b Auxiliary Feedwater System

RO2-02-LP05B.007: **EXPLAIN** the basis for each of the following Technical Specifications associated with the Auxiliary Feedwater System:
1. Technical Specifications 3.4.b Auxiliary Feedwater System

RO4-03-LPD11.003: In accordance with A-FW-05B, Abnormal Aux Feedwater System Operation, **SUMMARIZE** the subsequent operator actions necessary to respond to a AFW Discharge Piping > Ambient Temperature.

REFERENCES:

A-FW-05B, Rev. AL, Step 11-13
KNPP Technical Specification, 3.4.b.3, Amend No. 183, page TS 3.4-2
KNPP Technical Specification Bases, TS 3.4.b, page TS B3.4-1

SOURCE:

New

JUSTIFICATION:

A: TS 3.4.b.3 requires power reduction to less than or equal to 1673 MWt with two trains of AFW inoperable. With RTP at 1772 MWt the equivalent power level is 94.4%. Because A-AFW-5B has AFW-10B closed during the action to reseat the check valve, the TD AFW is inoperable since AFW-10B is part of the Turbine-driven train.

B: TS 3.4.b.3 requires power reduction to less than or equal to 1673 MWt with two trains of AFW inoperable. With RTP at 1772 MWt the equivalent power level is 94.4%. The Turbine-driven AFW train is made inoperable when AFW-10B is closed.

C: The power reduction is required. TS Bases identifies Turbine-driven AFW train is made inoperable when AFW-10B is closed.

15. Given the following:

- The plant has experienced a fire in the Control Room.
- The actions of E-O-06, Fire In Alternate Fire Zone, are being performed.

What are the indications of an air dryer filter failure, and what are the procedural actions performed by the Control Room Supervisor to mitigate the consequences of this failure.

A. Instrument Air Drier/Filter 1A/1B differential pressure will rise above 5 psid. The CRS will verify SA-121, Air Drier/Filter Bypass CV, opens.

B. Instrument Air Drier/Filter 1C differential pressure will rise above 10 psid. The CRS will open SA-100A , Air Drier 1A Supply, and IA-300, 1 1/2" Alt IA, and then request the Control Operator A to start Air Compressor B.

C. Instrument air header pressure will drop below 95 psig. The CRS will depress the Air Drier 1C RESET pushbutton to confirm problem, and then align flow through Instrument Air Drier 1B.

D. Instrument air header pressure will drop below 100 psig. The CRS will open SA-70 and SA-71, 1 1/2" Dedicated Instrument Air Header Isolations, that bypass the Instrument Air Driers.

Answer: D

COGNITIVE LEVEL:

2-DS - Recognizing the differences in IA alignemnt based on conditions with a fire in Alternate Fire Zone compared to normal system operations.

K/A:

078A2.01 – Ability to (a) predict the impacts of the following malfunctions or operations on the IAS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Air dryer and filter malfunctions.

OBJECTIVE:

RO4-03-LPD17.006: In accordance with A-AS-01, *Abnormal Station and Instrument Air System Operation*, **SUMMARIZE** the subsequent operator actions necessary to respond to a decreasing Instrument Air header pressure.

RO4-03-LPD21.05: **DISCUSS** the actions performed by each Control Room member upon reaching their respective areas.

REFERENCES:

E-O-06, Rev. Z, step 19.

OPERM-213-1, Rev. CD.

A-AS-01, Rev. R, 2.2; Step 4.5.

SOURCE:

New

JUSTIFICATION:

Air header pressure will drop below 100 psig, which is the loading setpoint for 1C Air Compressor which would be operating.
The CRS, as part of his actions in E-O-06, will open the Dedicated Air Header isolation valves to supply air to the required components. This path uses a separate filter (no drier).

A: This condition is true for normal Instrument Air header alignment. SA-121 will start to open if the Air Drier/Filter dp rises to 5 psid and will be full open at 10 psid bypassing Instruemnt Air Driers 1A and 1B.

B: Instument Air Drier 1C dp may rise to this value for a clogged filter. However, the actions are performed are from E-O-07, Fire in Dedicated Zone (Train A equipment), for which the Train B equipment is considered unaffected. The crew would not start Compressor B.

C: Air Compressor C should also have started automatically at 105 psig. The action described is the corrective action for Air Drier 1C Switching Failure light as delineated in A-AS-01, Abnormal Staion and Instrument Air System Operation, and is not applicable to the current situation.

16. What is the bases for maintaining the Spent Fuel Pool at the same boron as the water in the refueling cavity during REFUELING OPERATIONS?

   A. Maintain a minimum shutdown margin of at least 2% $\Delta K/K$ and provide a margin of at 30 minutes for the reactor to go critical for a dilution event.

   B. Maintain a minimum shutdown margin of at least 2% $\Delta K/K$ and provide a margin of at least 30 minutes for criticality in the SFP for a dilution event.

C. Maintain a minimum shutdown margin of at least 5% ΔK/K and provide a margin of at least 30 minutes for the reactor to go critical for a dilution event.

D. Maintain a minimum shutdown margin of at least 5% ΔK/K and provide a margin of at least 30 minutes for criticality in the SFP for a dilution event.

Answer: C
COGNITIVE LEVEL:

1-B - Identify the basis for maintaining refueling boron concentration in the SFP (and reactor cavity).

K/A:

033 2.2.25 – Spent Fuel Pool Cooling System (SFPCS): Knowledge of bases in technical specifications for limiting conditions for operations and safety limits.

OBJECTIVE:

RO2-01-LP021.008: **EXPLAIN** the basis for the following Technical Specifications associated with the Spent Fuel Pool Cooling & Cleanup System:
1. 3.8.a.2
3. 5.4.c

RO2-01-LP053.008: EXPLAIN the basis for Technical Specification 3.8 associated with refueling.

ROI-01-LP TS4.001: DESCRIBE in summary the items or components covered in the Design Features section of TS.

REFERENCES:

KNPP Technical Specification 3.8.a.5, Amend No. 165 & 5.4.a, Amend No. 162.
KNPP Technical Specification Basis (TS 3.8), Amend No. 172, 2nd paragraph page TS B3.8-1

SOURCE:

New

JUSTIFICATION:

A: The 2% $\Delta K/K$ is based on the New Fuel Storage Racks that are designed to maintain K_{eff} < 0.98 when moderated by aqueous foam. The reactor criticality event is the design event for dilution accident.

B: The 2% $\Delta K/K$ is based on the New Fuel Storage Racks that are designed to maintain K_{eff} < 0.98 when moderated by aqueous foam. The SFP is designed that when flooded with borated water, K_{eff} will remain < 0.95. The concern is with the reactor criticality event.

D: The 5% $\Delta K/K$ is at least ensured by maintaining the boron concentration specified in the COLR. The SFP is designed that when flooded with borated water, K_{eff} will remain < 0.95. The concern is with the reactor criticality event.

17. Waste Gas Decay Tank A is to be opened to atmosphere for maintenance. The release is planned and Discharge Permit generated.

What is the the total limit for the dose rate at and beyond the SITE BOUNDARY due to the noble gases included in the gaseous effluents, and what is the limit on hydrogen content at the time the the Waste Gas Decay Tank is opened for maintenance?

	Dose Rate Limit	H_2 Concentration Limit
A.	500 mrem/year	2% hydrogen
B.	1500 mrem/year	2% hydrogen
C.	500 mrem/year	4% hydrogen
D.	1500 mrem/year	4% hydrogen

Answer: A
COGNITIVE LEVEL:

1-F - Identify the limit for hydrogen and the limit for site dose rate when making a release from a WGDT prior to maintenance.

K/A:

071 2.2.22 – Waste Gas Disposal System (WGDS): Knowledge of limiting conditions for operations and safety limits.

OBJECTIVE:

RO4-01-LP-A01.004 - Given a requirement and/or note from the following NADs, EXPLAIN the purpose and implementation of the requirement and/or note.
b. NAD 01.12, RADIOLOGICAL GASEOUS WASTE DISCHARGE

AOI-81-LP32B.007 - **EXPLAIN** the LCO, applicability, and action requirements for each of the following Technical Specifications or Offsite Dose Calculation Manual requirements associated with the Gaseous Waste Processing System**:**

REFERENCES:

N-GWP-32B, Rev. AD, 4.2.8.d.14.E.1.
KNPP Offsite Dose Calculation Manual (ODCM), Rev. 8, Spec 3.4.1.a

SOURCE:

New

JUSTIFICATION:

B: The value for the hydrogen concentration is correct. The value for dose limit is that for the CEDE (organ dose) for Iodine, tritium and other R/A particulates.

C: 4% is the maximum value for oxygen concentration in WGDT, with immediate corrective action required. The value for the dose limit is correct.

D: 4% is the maximum value for oxygen concentration in WGDT, with immediate corrective action required. The value for dose limit is that for the CEDE (organ dose) for Iodine, tritium and other R/A particulates.

18. Given the following:

- The plant is in HOT SHUTDOWN at normal operating temperature and pressure.
- Circulating Water (CW) Pump A is in service, CW Pump B control switch is in pullout.
- Severe icing of the travelling screens is occurring and Forebay level is lowering.
- MSIV's are open
- A vacuum is being drawn on the condenser.
- Forebay level lowers to 41%.

What is the plant response and required crew actions for these conditions?

A. Forebay level will rise and condenser absolute pressure will rise.
A-CW-04, Abnormal Circulating Water System Operation, should be used to respond to these conditions.

B. Forebay level will rise and condenser absolute pressure will rise.
E-CW-04, Loss of Circulating Water, should be used to respond to these conditions.

C. Forebay level will lower and condenser absolute pressure will lower.
A-CW-04, Abnormal Circulating Water System Operation, should be used to respond to these conditions.

D. Forebay level will lower and condenser absolute pressure will lower.
E-CW-04, Loss of Circulating Water, should be used to respond to these conditions.

Answer: B
COGNITIVE LEVEL:

2-DR - Describe the relationship between circulating water and the forebay level and condenser vacuum. Apply the appropriate procedure.

K/A:

075A2.02 – Ability to (a) predict the impacts of the following malfunctions or operations on the Circulating Water System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of circulating water pumps.

OBJECTIVE:

RO2-02-LP004.005: **EXPLAIN** the purpose of the following procedures used to govern the normal, abnormal, and emergency operation of the Circulating Water System:
2. A-CW-04, Abnormal Circulating Water System Operation.
3. E-CW-04, Loss of Circulating Water

REFERENCES:

E-CW-04, Rev. X, step 3.
System Description 04, Rev. 4, 1.3.

SOURCE:

New

JUSTIFICATION:

A: The plant response is correct. However, A-CW-04 addresses those events or conditions affecting an individual pump, and a low temperature inlet condition. E-CW-04 addresses the loss of circ water pump(s).

C: The plant response is incorrect. Condenser absolute pressure will rise. A-CW-04 addresses those events or conditions affecting an individual pump, and a low temperature inlet condition. E-CW-04 addresses the loss of circ water pump(s)

D: The plant response is incorrect. Condenser pressure will rise and the forebay level should increase since no more water is being taken for the Circ Water flowpath. The procedural direction is correct.

19. Given the following:

- SP-36-082, Reactor Coolant System Leak Rate Check, has just been completed.
- RCS pressure and temperature were stable.
- Pressurizer level was on program.
- VCT level was stable but then decreased from 29% to 18% during the last TWO hours of the surveillance.

Using Operator Aid 02-20, what is the RCS leak rate and what are the actions?

A. Less than 1 gpm RCS leak. The source of the leak must be identified within the next 4 hours.

B. Less than 1 gpm RCS leak. The source of the leak must be identified within 48 hours or the plant must be placed in HOT SHUTDOWN.

C. More than 1 gpm RCS leak. The plant must be placed in HOT SHUTDOWN within 4 hours.

D. More than 1 gpm RCS leak. The source of the leak must be identified within 12 hours or the plant must be placed in HOT SHUTDOWN.

Answer: D
COGNITIVE LEVEL:

3-SPR - Use the plant information to determine leak rate and then apply the appropriate action.

K/A:

2.1.25 – Conduct of Operations: Ability to obtain and interpret station reference materials such as graphs, monographs, and tables which contain performance data.

OBJECTIVE:

RO2-01-LP-362.007 - **EXPLAIN** the LCO operation, applicability and action requirements for each of the following Technical Specifications associated with the Reactor Coolant System:
7. Leakage of Reactor Coolant (T.S. 3.1.d)

REFERENCES:

Operator Aid 02-20, Rev. 9-20-02
KNPP Technical Specification 3.1.d.1, Amend No. 165, page TS 3.1-8

PROVIDED REFERENCE: Operator Aid 02-20, VCT Level Conversion

SOURCE:

New

JUSTIFICATION:

Using VCT level the gallon conversion becomes 550.49 - 409.58 = 140.91 gallons over the 2-hour period. 140.91 / 2 = 1.174 gpm.

A: The leakage rate is wrong. The action for a leak greater than 1 gpm is the leak shall be subject to investigation and evaluation initiated within 4 hours of the indication.

B: The leakage rate is wrong. The plant must be in COLD SHUTDOWN within 48 hours if the leak greater than 1 gpm has NOT been identified.

C: The leak rate is correct. The ACTION to be in HOT SHUTDOWN is 12 hours if the leak has not been identified.

20. What is the MINIMUM on-duty shift complement when the plant is in COLD SHUTDOWN?

A. ONE Shift Manager (SRO), ONE Licensed Reactor Operator, TWO Nuclear Auxiliary Operator, ONE Radiation Technologist.

B. ONE Shift Manager (SRO), ONE Licensed Reactor Operator, TWO Nuclear Auxiliary Operator, ONE Radiation Technologist and the STA within 10 minutes of control room.

C. ONE Shift Manager (SRO), TWO Licensed Reactor Operators, TWO Nuclear Auxiliary Operators, and ONE Radiation Technologist.

D. ONE Shift Manager (SRO), TWO Licensed Reactor Operators, TWO Nuclear Auxiliary Operator, ONE Radiation Technologist and the STA within 10 minutes of control room.

Answer: C
COGNITIVE LEVEL:

1-F - Identify the shift staffing requirements for COLD SHUTDOWN.

K/A:

2.1.5 – Conduct of Operations: Ability to locate and use procedures and directives related to shift staffing and activities.

OBJECTIVE:

ROI-01-LP TS4.002: LIST by title and give the number required of each title for a full shift complement of personnel for all MODES.

REFERENCES:

Technical Specification 6.2.b.1, Amend No. 162

SOURCE:

KNPP Bank Question ROI-01-LPTS4.002 002

JUSTIFICATION:

A: Fails to meet the requirement for TWO ROs. Note that only (at least) ONE licensed operator is required in the control room when fuel is in the reactor.

B: Fails to meet the requirement for TWO ROs. Note that only (at least) ONE licensed operator is required in the control room when fuel is in the reactor. ALSO the STA requirement is applicable only above COLD SHUTDOWN condition.

D: The STA requirement is applicable only above COLD SHUTDOWN condition. Note that this is the above COLD SHUTDOWN compliment without the additional SRO.

21. Given the following:

- The core loading pattern will be changed during the next refueling outage.
- The new fuel assemblies are to be placed more toward the center of the core and the "twice-burned" assemblies more toward the periphery.
- Kexcess is the same at the beginning of both fuel cycles.

What affect would this loading pattern have on the plant operations?

A. The expected full power loop Delta T value should be significantly lower for this fuel cycle when compared to the value of full power loop Delta T for the previous fuel cycle.

B. The expected full power loop Delta T value should be significantly higher for this fuel cycle when compared to the value of full power loop Delta T for the previous fuel cycle.

C. IF PR NI channel gains are <u>NOT</u> changed during the outage, the PR NIs would read significantly below actual power when the first calorimetric is performed after the refueling outage.

D. IF PR NI channel gains are <u>NOT</u> changed during the outage, the PR NIs would read significantly above actual power when the first calorimetric is performed after the refueling outage.

Answer: C
COGNITIVE LEVEL:

2-DR - Recognize the relationship between the radial flux profile based placement of fuel and the effect on NI readout.

K/A:

2.2.32 – Equipment Control: Knowledge of the effects of alterations on core configuration.

OBJECTIVE:

O-FND-LP 2.2.6, 8 . State the effects of the following on the radial power distribution
a) Fuel loading pattern.

RO2-05-LP048.009 - **DESCRIBE** operating events associated with the Excore Nuclear Instrumentation System to include the following as appropriate:
1. General plant conditions that existed prior to and during the event.
2. Cause(s) of the event.
4. How the event or a similar event affected plant operations at KNPP

REFERENCES:

Kewaunee Core Control Theory, Chapter 6, page page 6-18

SOER 90-03, Nuclear Instrument Miscalibration Events

SOURCE:

Prairie Island 9/2003 NRC Exam (INPO Bank)

JUSTIFICATION:

Based on same Kexcess, the ΔT should remain unchanged.

A: Based on same Kexcess, the ΔT should remain unchanged.

B: Based on same Kexcess, the ΔT should remain unchanged.

D: The radial flux pattern should shift with the peak toward the center and less flux at the core periphery, therefore the NIs would sense less flux than before and read lower not higher.

22. Given the following:

- The plant is at 100% power.
- Annunciator 47013-C RXCP B OIL LEVEL HIGH/LOW alarms.
- The cause was determined to be a faulty low level switch for the lower bearing oil reservoir.
- The upper and lower bearing reservoirs have been determined to be normal and compensatory measures have been established.
- The following decisions have been made:
  - Repair the level switch during the upcoming outage scheduled to begin in 30 days.
  - Lift the input leads from the faulty switch into the alarm circuit.
  - During the lead lifting, disable the alarm.
  - Re-enable the alarm to restore alarm capability for the remaining oil reservoir level switches.

What are the requirements for 10CFR50.59 Evaluation and PORC review for these actions?

A. Requires 10CFR50.59 Evaluation prior to lifting level switch leads and disabling the alarm.
PORC review is required before the implementation of the activity.

B. Requires 10CFR50.59 Evaluation prior to lifting level switch leads and disabling the alarm.
PORC review is required within five days of implementation of the activity.

C. Requires 10CFR50.59 Evaluation within 12 hours of lifting level switch leads and disabling the alarm.
PORC review is required within two weeks of implementation of the activity.

D. Requires 10CFR50.59 Evaluation within 12 hours of lifting level switch leads and disabling the alarm.
PORC review is NOT required.

Answer: A
COGNITIVE LEVEL:

2-DR - Use the knowledge to identify SSC required by USAR, and apply appropriate reviews to meet commitments.

K/A:

2.2.7 – Equipment Control: Knowledge of the process for conducting tests or experiments not described in the safety analysis report.

OBJECTIVE:

RO4-01-LP-A14.002 - **DEFINE** the terms listed in the definitions section of the following GNPs:
a. GNP 04.04.01, 50.59 Applicability Review and Pre-Screening.

RO4-01-LP-A14.003 - **DISCUSS** department general applications associated with the following:
d. GNP 04.04.02, 50.59 Screening and Evaluation.

REFERENCES:

GNP-04.04.01, Rev. E, page 1-2
GNP-04.04.02, Rev. D, 6.2.9, 6.2.10, and 6.2.12
NAD-03.19, Rev. I, 5.2.3.2 and 5.2.7.4

SOURCE:

New

JUSTIFICATION:

This results in a change to a system described in the USAR during the time the alarm is disabled and requires a 50.59 Screening and Evaluation. This must be reviewed and approved by PORC prior to implementation.

B: It is not allowable to perform installation before PORC approval. However, PORC does have a 5-day limit for commenting on PORC minutes.

C: The 50.59 Evaluation must be completed and approved prior to installation. It is not allowable to perform installation before PORC approval. However, there is a requirement for having submitted items distributed to PORC Members at least two weeks in advance of the scheduled meeting.

D: The 50.59 Evaluation must be completed and approved prior to installation. PORCS approval is also required prior to installation.

23. Given the following:

- A LOCA has occurred
- A radioactive spill has occurred in the Auxiliary Building.
- An SITE EMERGENCY has been declared.
- All ERO positions have been filled.
- Entry into the Auxiliary Building is required to repair the leaking flange on the RHR system
- Estimated dose for the entry and work is 7 REM TEDE.

What are the MINIMUM requirements that must met for this entry into the Auxiliary Building?

A. EPIPF-AD-11-04, Emergency Exposure Authorization, and EPIPF-AD-11-01, Emergency Radiation Work Permit, must be completed and approved prior to entry.

B. EPIPF-AD-11-04, Emergency Exposure Authorization, must be completed and approved prior to entry, and the individuals must sign onto the existing Maintenance Radiation Work Permit prior to entry.

C. If a Priority Entry is made, EPIPF-AD-11-04, Emergency Exposure Authorization, must be completed and approved prior to entry, and EPIPF-AD-11-01, Emergency Radiation Work Permit, must be completed immediately following the completion of the entry.

D. If a Priority Entry is made, EPIPF-AD-11-04, Emergency Exposure Authorization, and the individuals must log entry on the existing Maintenance Radiation Work Permit immediately following the completion of the entry.

Answer: A
COGNITIVE LEVEL:

1-F - Knowledge of the Emergency Radiation Entry Controls.

K/A:

2.3.7 – Knowledge of the process for preparing a radiation work permit.

OBJECTIVE:

REFERENCES:

EPIP-AD-11, Rev. U, 5.1.2 & 5.1.4

EPIP-RET-02D, Rev. N, 3.3 & 3.4

SOURCE:

New

JUSTIFICATION:

RCA entries where the 10CFR20 dose limit is likely to be exceeded requires completed "ERWP" and "Emergency Exposure Authorization" forms. Additionally all exposures which could exceed 10CFR20 dose limits shall be approved by the ED.

B: The completed "Emergency Exposure Authorization" is correct. use of an existing RWP is allowed only if the task is not expected to exceed any 10CFR20 dose limit.

C: A PRIORITY ENTRY can be used to expedite entry of emergency response personnel into the RCA when conditions should not exceed 10CFR20 dose limits. In this case the RWP may be completed following the entry. Additionally a qualified Radiation Technologist must accompany the team (to provide the level of protection normally afforded by the RWP). In this case the EWRP must be complete prior to entry.

D: A PRIORITY ENTRY can be used to expedite entry of emergency response personnel into the RCA when conditions should not exceed 10CFR20 dose limits. In this case the RWP may be completed following the entry. Additionally a qualified Radiation Technologist must accompany the team (to provide the level of protection normally afforded by the RWP). A normal RWP is not used for exposures expected to exceed 10CFR20 limits.

24. Given the following:

- A plant heatup is in progress following an outage.
- RCS temperature is 375°F.
- ICS Pump B is OUT-OF-SERVICE for maintenance.
- A fire in Station Service Transformer 51 has resulted in loss of power from 4160V Bus 5 to Bus 51.

Which of the following addresses the capability of establishing power to the ICS Pump A and continued plant operation?

A. Power <u>CANNOT</u> be restored, and, the reactor <u>CANNOT</u> be taken critical.

B. Bus 51 and Bus 61 can be cross-connected for up to 7 days only if RCS temperature is reduced to and maintained less than 350°F during this period.

C. Bus 51 and Bus 61 can be cross-connected up to 7 days, and the reactor <u>CANNOT</u> be taken critical during this period.

D. Bus 51 and Bus 61 can be cross-connected for up to 24 hours, normal power operations may be commenced during this period.

Answer: C
COGNITIVE LEVEL:

2-RI - Recognize the relationship between ICS Tech Spec requirements, and Electrical Bus Tech Spec and administrative requirements.

K/A:

2.4.11 – Emergency Procedures/Plan: Knowledge of abnormal condition procedures.

OBJECTIVE:

RO2-01-LP023.007 - **EXPLAIN** the LCO operation, applicability and action requirements for each of the following Technical Specifications associated with the ICS System:
1. Technical Specification 3.3.

RO2-03-LP040.008 - **EXPLAIN** the LCO operation, applicability and action requirements for each of the following Technical Specifications associated with the 480 V SUPPLY AND DISTRIBUTION System:

1. Technical Specification LCO 3.7.

REFERENCES:

N-ELV-40, Rev.Q, 2.6
KNPP Technical Specification 3.3.c.1.A.1, Amend. No. 184, page TS 3.3-4
KNPP Technical Specification 3.7.a.5, Amend. No. 122

SOURCE:

2000 Kewaunee NRC Exam **-** (INPO Bank)

JUSTIFICATION:

Following an outage, TWO trains of ICS are required to take the reactor critical. Buses 51 & 61 may be cross-connected for up to seven days in INTERMEDIATE SHUTDOWN with RCS >350°F or in HOT SHUTDOWN. Also, TS 3.7.c requires at least its normal OR Emergency power source be operable (which they aren't) and that the redundant train be operable (which it isn't).

A: The ICS Pump can have power restored to it by cross connecting Bus 51 & 61. Heatup is still allowed.

B: At less than 350°F, the buses can be cross-connected if the same Train RHR and SG are operable, but it is NOT a requirement that temperature be lowered.

D: Technical Specifications do NOT allow the reactor to be made critical. Also 24 hour limitation is provided for deenergized 480V bus (but only if the redundant bus and its ESF loads are OPERABLE). The startup limitation is under both TS for ICS and Electrical Buses since neither Train is operable (and conditions are NOT following recovery from an inadvertent trip).

25.

NOTE: This Question and supporting information may contain Security Sensitive Information and should be kept from public disclosure. [10CFR 2.390]