AS SUBMITTED FOR REVIEW (SRO) WRITTEN EXAMINATION

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.

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:

KNPP Bank Question

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 LOCA has occurred.

- RCS subcooling is 20ºF.

- RCS pressure is 1730 psig.

- Containment pressure is 4.2 psig.

- Annunciator 47051-C, CNTMT EMERG DISCH DMPRS ACTIVATED, has alarmed.

- Indications show the following for Fan Coil Unit Emergency Discharge Damper positions:

What action should be taken?

A. Manually close RBV-150A and RBV-150B.

B. Manually open RBV-150C and RBV-150D.

C. Manually close RBV-100A-B, Cntmt FCU A & B Disch to Rflg Floor.

D. Manually open RBV-100C-D, Cntmt FCU C & D Disch to Rflg Floor.

Answer: B

COGNITIVE LEVEL:

2-DR - Describe the relationship between the SI actuation signal and the ventilation system. Identify the correct action to take.

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:

3. Containment Fan Coil Unit A (B) (C) (D) Emergency Discharge Damper RBV-150A (B) (C) (D)

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:

b. ESF Actuation Signals

REFERENCES:

E-3310, Rev. E

BKG E-0, Rev. B, Step 9 ACTIONS

SOURCE:

KNPP Bank Question 0180000001K04 002

JUSTIFICATION:

A: This would be the correct action to take if containment pressure had NOT exceeded 4 psig. Operation with the Emergency Dampers open would have isolated cooling to the RXCP vaults.

C: A similar alarm exist for damper misalignment, 47052-C CNTMT EMRG DISCH DMPR ABNORMAL. The alarm given indicates that the actuation signal for actuation of the Emergency Dampers has occurred.

D: A similar alarm exists for damper misalignment, 47052-C CNTMT EMRG DISCH DMPR ABNORMAL. Directions are provided to OPEN RBV-100A-B and RBV-100C-D if SI is NOT actuated.

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.

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% DK/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% DK/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% DK/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% DK/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% DK/K is based on the New Fuel Storage Racks that are designed to maintain Keff < 0.98 when moderated by aqueous foam. The reactor criticality event is the design event for dilution accident.

B: The 2% DK/K is based on the New Fuel Storage Racks that are designed to maintain Keff < 0.98 when moderated by aqueous foam. The SFP is designed that when flooded with borated water, Keff will remain < 0.95. The concern is with the reactor criticality event.

D: The 5% DK/K is at least ensured by maintaining the boron concentration specified in the COLR. The SFP is designed that when flooded with borated water, Keff 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	H2 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 NOT 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 NOT 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 DT should remain unchanged.

A: Based on same Kexcess, the DT should remain unchanged.

B: Based on same Kexcess, the DT 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.

21. 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 <u>NOT</u> 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 "EWRP" 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.

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). 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 375EF.

- 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 CANNOT be restored, and, the reactor CANNOT 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 350EF during this period.

C. Bus 51 and Bus 61 can be cross-connected up to 7 days, and the reactor CANNOT 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 >350EF 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 350EF, 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]