KEWAUNEE
INITIAL LICENSE EXAM

DECEMBER 11 THRU 20, 2000

The ***final written examination and answer key*** (enclosure to examination report).

U.S. Nuclear Regulatory Commission
Site-Specific
Written Examination

Applicant Information

Name: MASTER EXAMINATION	Region: III
Date: 12/11/00	Facility/Unit: Kewaunee
License Level: RO	Reactor Type: W
Start Time:	Finish Time:

Instructions

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. The passing grade requires a final grade of at least 80.00 percent. Examination papers will be collected four hours after the examination starts.

Applicant Certification

All work done on this examination is my own. I have neither given nor received aid.

_____
Applicant's Signature

Results

Examination Value	_____ Points
Applicant's Score	_____ Points
Applicant's Grade	_____ Percent

Kewaunee NRC RO Examination Answer Key

1 .a
2 .a
3 .d
4 .b
5 .a
6 .d
7 .a
8 .d
9 .d
10 .d
11 .d
12 .a
13 .a
14 .b
15 .a
16 .d
17 .b
18 .a
19 .d
20 .d
21 .d
22 .a
23 .c
24 .c
25 .a

26 .c
27 .b
28 .b
~~29 .d~~ DELETED DAP
30 .a
~~31 .c~~ DELETED DAP
32 .c
33 .c
34 .b
35 .a
36 .b
37 .a
38 .c
39 .b
40 .a
41 .a
42 .a
43 .c
44 .c
45 .c
46 .b
47 .d
48 .c
49 .d
50 .b

51 .c
52 .a
53 .a
54 .c
55 .b
56 .d
57 .b
58 .b
59 .d
60 .b
61 .a
62 .c
63 .d
64 .c
65 .b
66 .a
67 .b
~~68 .b~~ DELETED DAP
69 .d
70 .b
71 .c
72 .a
73 .c
74 .c
75 .c

76 .d
77 .b
78 .d
79 .a
80 .c
81 .b
82 .b
83 .b
84 .c
85 .d
86 .b
87 .c
88 .d
89 .a
90 .a
91 .b
92 .c
93 .c
94 .d
95 .c
96 .d
97 .b
98 .c
99 .a
100 .b

Reactor Operator Examination

1 . Given the following conditions:

- A Reactor Startup is in progress after a reactor trip
- TWO NCOs and the CRS are present in the Control Room
- The RO has completed withdrawal of Shutdown Bank B and has placed the Control Rod Bank Selector switch to MAN
- The CRS directs the BOP to record the power readings on the Day-End Meter Readings worksheet in preparation for Generator startup

Which of the following describes how the requirement for Control Room manning is maintained for the above conditions?

The operator...

a. can perform the task since ONE NCO is allowed to go to the relay room.

b. can perform the task since the CRS will fulfill the missing NCO's responsibilities while he is out of the RO Watchstation

c. can perform the task since only ONE licensed operator is required in the RO Watchstation when rods are NOT being moved.

d. CANNOT perform the task since TWO licensed operators must be present in the RO Watchstation.

2 . Given the following conditions:

- RT-GWP-32B "Waste Gas System Leakage Test" is being performed to identify the source of leakage from the Waste Gas System
- The SS has completed the IPTE Checklist and the NAOs have been dispatched to perform the procedural actions

Which of the following describes the requirements for procedure use to be followed by a NAO performing the evolution?

The operator...

a. will have a copy of the procedure with him/her locally as the steps are performed.

b. may have procedure readily available for reference but NOT necessarily at the work location.

c. will take direction from the NCO in the Control Room who is responsible for reading and initialing steps.

d. may perform the steps from memory without having the procedure available locally since a detailed briefing was held.

3 . Which of the following identifies those CVCS-related components that can be operated from the Dedicated Shutdown Panel?

a. CVC-11/CV-31229 Charging Line Isolation.
CV-301/MV-32056 RWST Supply to Charging Pumps.
CVC-203B/CV-31689 Seal Injection Filter Bypass Valve.

b. CVC-440/MV-32217 Emergency Boration to Charging Pumps.
CVC-207A(B)/CV-31237(31238) RXCP A(B) #1 Seal Leakoff Isolation.
LD-27/CV31096 VCT/Holdup Tank Divert Valve.

c. LD-4A(B,C)/CV-31231(31232, 31233) Letdown Orifice A (B,C) Isolation.
LD-60/MV-32099 RHR to CVCS Letdown Line.
CVC-7/CV-31103 Charging Control Chg Line.

d. CVC-15/CV-31230 PRZR Auxiliary Spray Valve.
CVC-215B/CV-31682 Seal Water Filter Bypass Valve.
CC-302/CV-31100 Letdown Cont Outl Temp Controller.

4 . Given the following conditions:

- A reactor startup is in progress following a shutdown for valve repair valve
- The power history is 1000 MWD/MTU
- Control Bank D has just been withdrawn to 196 steps
- The ECP was calculated to be at 108 steps on Control Bank D
- The reactor is NOT critical
- Both Source Range & Intermediate Range channels have NOT reached their eighth-fold values

What action is required? (Accumulated Integral Rod Worths, BOL data attached)

The operator will...

a. continue to withdraw Bank D rods until the eighth-fold value is reached.

b. return rods to the ECP rod position and recalculate the ECP.

c. manually trip the reactor and initiate emergency boration.

d. insert all Control Banks to shut down the reactor.

5 . Given the following conditions:

- The plant is in REFUELING
- Fuel shuffle is complete
- Preparations are completed to initiate draining of the Reactor Cavity

What is the responsibility of the NCO concerning the Source Range instrumentation?

As a MINIMUM, the operator must verify...

a. at least ONE channel is operating and it is monitored during the draining operation.

b. at least ONE channel is operating and it shall be monitored both in the Control Room and in Containment.

c. TWO channels are operating and the audio count can be monitored in Containment.

d. TWO channels are operating and each shall be monitored during the draining operation.

6 . Given the following conditions:

- The plant is at 80% power a load increase being initiated
- Bank D Control Rods are at 186 steps
- Rods are being controlled manually
- The Bank Selector switch is in the CBD (Control Bank D) position
- The reactor power is then taken to 100%, all rods out condition
- Rod control is restored to AUTO

If NO further action is taken with respect to rod control, what is the effect when the plant is shutdown?

a. The calculated RIL for Control Bank D will be 40 steps below the RIL value imposed by Technical Specifications.

b. The ROD BOTTOM ROD DROP annunciator will actuate when Control Bank D is inserted below 60 steps.

c. The CONTROL BANK LOW LOW annunciator will actuate when Control Bank D reaches 184 steps.

d. Control Bank C will begin to insert when Control Bank D is inserted below 170 steps.

7. Which of the following describes the conditions for which an Extended Radiation Work Permit (RWP) would be issued?

The extended RWP is used for jobs that...

a. are repetitive in nature and may exist for long periods of time.
b. are expected to exceed the normal administrative dose limits.
c. are located in areas where conditions are subject to rapid change.
d. are corrective maintenance tasks continuing beyond the end of the calendar month.

8. A point source in containment is reading 500 mRem/hr at a distance of two (2) feet. Two options are available to complete a mandatory work assignment near this radiation source:

Option 1 - ONE operator can perform the assignment in forty (40) minutes working at a distance of three (3) feet from the source

Option 2 - TWO operators, trained in the use of special extension tooling, can perform the assignment in sixty-five (65) minutes at a distance of six (6) feet from the source

Which is the preferred option when considering the total exposure based on the ALARA plan?

a. Option 1, which results in total exposure of 0.222 MAN-REM.
b. Option 1, which results in total exposure of 0.148 MAN-REM.
c. Option 2, which results in total exposure of 0.361 MAN-REM.
d. Option 2, which results in total exposure of 0.120 MAN-REM.

9 . Given the following conditions:

- Discharge is in progress from Waste Condensate Tanks.
- R-18, Waste Discharge Liquid radiation monitor, fails off-scale high.

Which of the following actions is NOT required prior to reinitiating the release?

Technically qualified members of the Facility Staff must...

a. complete TWO independent verifications of the discharge line valving.
b. perform TWO independent verifications of the release rate calculations.
c. analyze TWO independent samples from the tanks for gamma and tritium.
d. establish TWO independent locations for taking grab samples during the release.

10 . Given the following conditions:

- Steam Generator A level rose to 70% following failure of FW-7A/CV-31027, S/G A Main FW Valve
- The reactor tripped following a turbine trip
- Offsite power was lost to the plant when the generator output breakers opened
- All equipment operated normally

Which of the following describes the response of the NCO following the trip?

The operator should...

a. perform ECA-0.0 "Loss Of All AC Power", immediate actions.
b. immediately reduce Auxiliary Feedwater flow to 200 gpm total.
c. close both MS-1A and MS-1B, S/G A and B Main Steam Isolation Valves.
d. verify plant indications and response for the E-0, "Reactor Trip Or Safety Injection", immediate actions.

11 . Which Safety Function would be addressed FIRST if they were all discovered at the same time?

a. Subcriticality Orange Path.
b. Containment Red Path.
c. Integrity Orange Path.
d. Heat Sink Red Path.

12 . Given the following conditions:

- The plant is at 100% power.
- The Carbon Dioxide Hose Station located adjacent to 1A and 1B Battery rooms has been declared INOPERABLE.

What Operator action shall be taken within ONE hour per FPP 08-01 "Fire Plan Operability, Surveillance & Contingency Requirements"?

a. Establish backup fire suppression equipment.

b. Verify the Zone fire detectors operating.

c. Initiate actions for Plant backdown.

d. Post an hourly fire watch.

13 . Given the following conditions:

- A LOCA has occurred
- Containment hydrogen is 0.5% and has a rising trend
- Actions are underway to establish Train B Hydrogen Dilution using Instrument Air
- A "new" NAO has been assigned to assist by aligning the valves to establish Instrument Air flow to containment

Where would you direct the NAO to go?

a. Main Steam Penetration Area A.

b. Auxiliary Building Loading Dock.

c. Boric Acid Tank Room.

d. Decon Storage Room.

14. Given the following conditions:

- The plant is at 35% power
- Circ Water Pump B is out of service for maintenance
- Service Transformer 1-33 is out of service for maintenance
- Bus 61 lockout was received 5 minutes ago and is being investigated
- Battery Charger A is out of service for testing

A loss of power to which of the following will result in an IMMEDIATE reactor trip?

a. Bus 3.

b. Bus 4.

c. Bus 5.

d. Bus 6.

15. Given the following conditions:

- Reactor power is at 75% and increasing slowly
- PRZR pressure is slowly increasing
- PRZR level is increasing
- Tavg is increasing
- Containment parameters are normal

What event is occurring?

a. Control Bank D continuous withdrawal.

b. A loop Tavg circuit is drifting high.

c. A PORV is leaking to the PRT.

d. Turbine runback in progress.

16. Given the following conditions:

- The plant is at 60% power steady state
- PRZR Spray Valves PS-1A & PS-1B are currently closed in AUTO
- PRZR Pressure Control Channel Switch is selected to the 2-3 position
- PRZR pressure PT-429 reads 2230 psig and rising
- PRZR pressure PT-430 reads 2235 psig and rising
- PRZR pressure PT-431 has just failed to 2185 psig
- PRZR pressure PT-449 reads 2235 psig and rising

Assuming NO operator action is taken, what is the Pressurizer Pressure Control System response to these conditions?

a. PRZR spray Valves PS-1A/B will fully open and continue to depressurize the PRZR until saturation conditions exist in the RCS.

b. PRZR pressure will oscillate between 2210 psig and 2250 psig by the cycling of the proportional and backup heaters.

c. PRZR pressure will stabilize below 2310 psig by the operation of the PRZR spray valves PS-1A/1B.

d. PRZR pressure will oscillate between 2315 and 2335 psig by the cycling of PORV PR-2B.

17. The plant was manually tripped from rated load because of a slow uncontrollable decrease in RCS pressure. A Safety Injection was automatically initiated a short while ago.

Current indications are as follows:

- RCS pressure is 1700 psig slowly decreasing
- Highest CET temperature reads 575°F
- All PRZR heaters are energized
- PRZR level is 30% and decreasing on LT-426, 427, and 428
- PRZR level LT-433 is 20% and decreasing
- Containment pressure is 1.5 psig and increasing
- Annunciator 47031-Q CONTAINMENT SUMP A LEVEL HIGH is in alarm
- PRT level is 74% and stable
- PRT temperature is 100°F and stable

Which failure has caused the above events and indications?

a. A PRZR PORV has failed open.

b. A PRZR heater well has ruptured.

c. A PRZR reference leg has ruptured.

d. A PRZR Spray Valve has failed open.

18. Given the following conditions:

- The plant is in HOT SHUTDOWN at 547°F
- RXCP B has just been stopped
- The NCO has placed the controller for PS-1B/CV-31111 PRZR Spray Control Loop B in MANUAL and shut the valve

Why did the operator close PS-1B?

a. Prevent spray flow from bypassing the PRZR.

b. Prevent spray bypass flow from affecting RCS pressure.

c. Prevent rapid depressurization during RXCP B coastdown.

d. Minimize differential temperature between the PRZR and the spray line.

19. Given the following conditions:

- The plant is in HOT SHUTDOWN following a trip from 100% power
- 5 minutes following the trip, both RXCPs tripped

What is the response of core delta-T?

Core delta-T will...

a. remain constant, then drop as natural circulation flow is developed.
b. drop, then return to the no-load delta-T as natural circulation flow is developed.
c. rise, then return to the no-load delta-T as natural circulation flow is developed.
d. rise, then stabilize at the higher delta-T as natural circulation flow is developed.

20. Given the following conditions:

- RCS boron concentration is currently 900 ppm
- RCS leakage is less than 0.1 gpm
- Both Reactor Makeup Pumps are in AUTO with Reactor Makeup Pump A running
- Boric Acid Transfer Pump A is in AUTO and FAST
- Boric Acid Transfer Pump B is in PULLOUT
- Auto makeup to the VCT has actuated due to low VCT level

What will occur if the Boric Acid Transfer Pump trips when VCT level reaches 20% and NO operator action is taken?

VCT level will...

a. continue to rise to 28% and VCT boron concentration will remain the same.
b. continue to rise to 28% and VCT boron concentration will decrease between 250 and 260 ppm.
c. stabilize at approximately 21% and VCT boron concentration will remain the same.
d. stabilize at approximately 21% and VCT boron concentration will decrease between 20 and 30 ppm.

21. Given the following conditions:

- The RCS is solid
- RCS temperature is 185°F
- RHR Train A is operating to maintain RCS temperature
- RHR letdown is aligned through the RHR/CVCS spectacle flange
- LD-10/CV-31099 Letdown Cont Pressure is in AUTO set to 325 psig

What will occur if the controller output for LD-10 fails low (0%)?

LD-10 moves fully...

a. closed to raise the upstream letdown pressure as indicated on PI-155, and RCS pressure decreases.

b. open to raise the upstream letdown pressure as indicated on PI-155, and RCS pressure increases.

c. closed to lower the upstream letdown pressure as indicated on PI-155, and RCS pressure increases.

d. open to lower the upstream letdown pressure as indicated on PI-155, and RCS pressure decreases.

22. Given the following conditions:

- The plant is in COLD SHUTDOWN
- Refueling Level is at 15% for RXCP B seal work
- RHR Pump A is in operation
- Charging Pump A is running
- BOTH SI Pump Control Switches are in PULLOUT

What action is required if feeder breaker 1-502 were to fail on thermal overload actuation?

a. Start RHR Pump B.

b. Start BOTH SI Pumps.

c. Start Charging Pump B.

d. Place SI Pump B Control Switch in AUTO position.

23 . Given the following conditions:

- The plant is in INTERMEDIATE SHUTDOWN
- RCS temperature is 275°F
- RHR Pump A is operating with BOTH RHR Heat Exchangers in service for cooldown
- RHR Pump B is out of service for maintenance
- Annunciator 47024-H CC SURGE TANK LEVEL HIGH/LOW is in alarm
- CC Surge Tank level is verified to be increasing
- R-17 Comp Cooling Liquid Monitor indicates increasing radiation levels

Identify the leak location and the correct operator response for these conditions.

The leak is located in the...

a. RHR Pump A Seal Cooler. Isolate flow to the cooler and continue the cooldown to 201°F using RHR Pump A.

b. Seal Water Heat Exchanger. Isolate flow to the heat exchanger and continue the cooldown to COLD SHUTDOWN using both RHR Heat Exchangers.

c. RHR Heat Exchanger A. Isolate flow to the heat exchanger and continue the cooldown to 201°F using RHR Heat Exchanger B only.

d. RHR Heat Exchanger B. Isolate flow to the heat exchanger and continue the cooldown to COLD SHUTDOWN after aligning the Spent Fuel Pool Cooling Heat Exchanger to the RHR system.

24 . Given the following conditions:

- A cooldown to COLD SHUTDOWN is in progress
- RCS Tavg is 450°F
- RCS pressure is 990 psig
- Both SI Accumulator Discharge Isolation power supply breakers are closed
- SI-20A/MV-32091 Accumulator A Isolation valve is closed with its switch in AUTO
- SI-20B/MV-32096 Accumulator B Isolation valve is open with its switch in AUTO

What is the response of the SI Accumulators if a double-ended pipe break occurs on Loop B Cold Leg at the reactor vessel nozzle (Design Basis LOCA)?

a. Neither SI Accumulator will inject to the reactor vessel.

b. Only SI Accumulator B will inject to the reactor vessel.

c. Only SI Accumulator A will inject to the reactor vessel.

d. Both SI Accumulators will inject to the reactor vessel.

25 . Given the following conditions:

- The plant is at 100% power with normal equipment alignment
- OCBs supplying power to the Reserve Aux Transformer (RAT) open
- Coincident with the loss of power to the RAT, a SI occurs

What is the starting order for the ECCS Pumps?
(Assume the maximum design time to reach load assumption conditions for the respective diesel generators)

a. SI Pump A, RHR Pump A, SI Pump B, RHR Pump B.
b. SI Pump A, SI Pump B, RHR Pump A, RHR Pump B.
c. SI Pump B, RHR Pump B, SI Pump A, RHR Pump A.
d. SI Pump B, SI Pump A, RHR Pump B, RHR Pump A.

26 . The following PRT parameters are noted:

- Temperature is 125°F
- Level is 72%
- Pressure is 5 psig
- Hydrogen concentration is 1.3%

What action should be taken regarding these conditions?

a. Decrease level to less than 67%.
b. Increase pressure to greater than 8 psig.
c. Decrease temperature to less than 120°F.
d. Increase hydrogen content to greater than 2%.

27 . Given the following conditions:

- 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 actuates
  - 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. A RXCP #1 seal failure.

b. A PRZR PORV has stuck open.

c. Failure of a Steam Generator Blowdown piping tap.

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

28 . Given the following conditions:

- Initial PRZR pressure was 2235 psig
- PR-2A/CV-31110 PRZR PORV has popped open
- The operator has just closed PR-1A/MV-32089 PRZR PORV Block Valve
- Current PRZR pressure is 2190 psig
- PRT parameters:
  - Level is 75%
  - Pressure is 6.5 psig
  - Temperature is 123°F

What is the expected temperature indication for TI-438, PORV Outlet temperature, immediately following closure of PR-1A?

a. 650°F.

b. 230°F.

c. 170°F.

d. 125°F.

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29. Given the following conditions:

- The Plant is at 100% power
- Reactor trip breaker testing is being performed with Reactor Trip Bypass Breaker B (52/BYB) racked in and closed
- Both Reactor Trip Breakers (52/RTA and 52/RTB) are closed
- Reactor Trip Bypass Breaker A (52/BYB) is open and racked out

What is the effect on systems operation if Reactor Trip Bypass Breaker B (52/BYB) failed to open on a reactor trip?

a. Only the turbine control valves and intercept valves will close.

b. The Atmospheric Steam Dump valves receive an open signal but do NOT arm.

c. If RCS Low Tavg occurs, FW-7B/CV-31027 S/G B Main FW Valve does NOT close.

d. Following any SI actuation and reset, automatic actuation of SI Train B CANNOT be blocked.

30. Given the following conditions:

- The reactor has just tripped
- Prior to the trip, reactor power was at 30% with all systems in their normal lineup
- PRZR pressure channel 4 (PT-449) previously failed low and was removed from service in accordance with A-MI-87 "Bistable Tripping for Failed Reactor Protection or Safeguards Inst."
- Investigation showed a Reactor Protection System bistable failure (actuation) precipitated the reactor trip

Which of the following bistable failures would have caused the reactor trip?

a. Channel 1 Overtemperature Delta-T 405C OVER TEMP TRIP.

b. Channel 2 Turbine Impulse Pressure 486A TURBINE PRESS P13.

c. Channel 3 Overpower Delta-T 407A OVER POWER TRIP.

d. Channel 4 Nuclear Power Range Instrument Drawer N44A OVERPOWER TRIP HIGH RANGE.

21. Which of the following conditions will require a manual start of the Turbine Driven Auxiliary Feedwater Pump if the automatic start fails?

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a. Train B SI actuation.

b. Trip of both Main Feedwater Pumps at 60% power.

c. Lockout of the RAT coincident with a turbine trip at 100% power.

d. Steam Generator A level at 10% NR and Steam Generator B level at 70%.

32. Given the following conditions:

- A large break LOCA has occurred
- Containment pressure is observed to be 25 psig
- Containment Spray has NOT initiated
- Manual actuation of Containment Spray has been unsuccessful
- All other ESF actuations and components have functioned normally

What is the proper sequence of actions to be taken to manually initiate Containment Spray for Train A?

Manually start the ICS Pump A...

a. Check ICS-5A/MV-32066 and ICS-6A/MV-32067, Cntmt Spray Pump A Discharge Isolation valves automatically open.
Check ICS-201/CV31272 and ICS-202/CV-31273, ICS Recirculation RWST are closed.
Check CI-1001A/CV-31393 and CI-1001B/CV-31394, Caustic Additive to Cntmt Spray valves automatically open.

b. Manually open ICS-5A/MV-32066 and ICS-6A/MV-32067, Cntmt Spray Pump A Discharge Isolation valves.
Check ICS-201/CV31272 and ICS-202/CV-31273, ICS Recirculation to RWST are closed.
Check CI-1001A/CV-31393 and CI-1001B/CV-31394, Caustic Additive to Cntmt Spray valves automatically open.

c. Manually open ICS-5A/MV-32066 and ICS-6A/MV-32067, Cntmt Spray Pump A Discharge Isolation valves.
Check ICS-201/CV31272 and ICS-202/CV-31273, ICS Recirculation to RWST are closed.
Manually open CI-1001A/CV-31393 and CI-1001B/CV-31394, Caustic Additive to Cntmt Spray valves.

d. Manually open ICS-5A/MV-32066 and ICS-6A/MV-32067, Cntmt Spray Pump A Discharge Isolation valves.
Manually close ICS-201/CV31272 and ICS-202/CV-31273, ICS Recirculation to RWST.
Manually open CI-1001A/CV-31393 and CI-1001B/CV-31394, Caustic Additive to Cntmt Spray valves.

33 . Given the following conditions:

- The plant is at 100% power
- Control Bank D rods are at 208 steps withdrawn
- An electrical failure deenergizes Instrument Bus BRB-114

If the NCO attempts to withdraw rods, which of the following prevents rod motion?

a. Overpower Delta-T Rod Stop.

b. Overtemperature Delta-T Rod Stop.

c. Power Range Overpower Rod Stop.

d. Intermediate Range High Flux Rod Stop.

34 . If a turbine runback to 60% power occurs from 100% steady-state power, what is the expected response for Power Range Nuclear Instruments (NIs) to Xenon?
(Consider only Xenon effect)

The NIs will drop to 60%,...

a. rise above 60% over the next 8 hours, and then slowly drop to 60%.

b. drop below 60% over the next 8 hours, and then slowly rise to 60%.

c. drop below 60% and stabilize over the next 72 hours at less than 60%.

d. rise above 60% and stabilize over the next 72 hours at greater than 60%.

35 . If an input to the Subcooling Margin Monitor (SMM) goes out of range due to changing plant conditions with the SMM operating in NORMAL mode, what occurs on the associated SMM digital display?

a. The selected margin value is flashing while tracking its calculation.

b. The ID number for the out-of-range detector is flashing.

c. The value for the out-of range parameter is flashing.

d. Four E's are displayed ("EEEE").

36 . How does a SI Signal affect operation of SW-901B-1/CV-31705, Header B Shroud Clg Coil A/B Bypass valve?

a. It gets a close signal to provide Containment Isolation for the Shroud Cooling Coil A/B.

b. It gets an open signal to ensure adequate cooling flow through Containment Fan Coil Unit B.

c. It gets an open signal to maintain a minimum flow of 75 gpm through Shroud Cooling Coil A/B.

d. It gets a close signal to prevent excessive cooling flow conditions for Containment Fan Coil Unit B.

37 . What is the major concern if the Containment Fan Coil Units Emergency Discharge Dampers RBV-150 A and B both fail open during normal 100% power operations?

a. RXCP B motor stator overheating.

b. Hot air stratification in the Containment Dome.

c. Reverse air flow through the Containment Fan Coil Unit.

d. Damage to the Reactor Vessel Gap and Nuclear Instrumentation.

38 . Given the following conditions:

- A LOCA has occurred
- Containment Spray has actuated
- RWST level currently reads 40%
- Caustic Additive Standpipe level currently reads 100%

What would be the effect of these conditions?

a. Containment radiation levels are higher due to the increased radioactive noble gas production.

b. Containment pressure peaks at higher value due to the reduced heat removal capacity of the ICS spray.

c. Corrosion of components in containment increases due to lower pH value of the containment sump fluid.

d. Removal of hydrogen in the containment atmosphere is lower due to reduce volume of injected sodium hydroxide.

39 . Given the following conditions:

- An accident has occurred
- 20 minutes has elapsed since the event initiation
- Containment Spray has actuated

Which of the following identifies the set of parameters that must be satisfied to secure Internal Containment Spray following actuation during an accident condition?

a. Containment pressure less than 4 psig AND ICS Pump run time greater than 30 minutes.
b. Containment pressure less than 4 psig AND containment radiation level less than 2 R/hr.
c. Containment pressure less than 17 psig AND ICS Pump run time greater than 30 minutes.
d. Containment pressure is less than 17 psig AND containment radiation level less than 2 R/hr.

40 . Given the following conditions:

- A LOCA has occurred
- Containment hydrogen is at 3.4% and increasing

Which of the following describes the appropriate method for reducing the hydrogen concentration per N-RBV-18C "POST LOCA Hydrogen Control"?

a. Use Shield Building Vent System
b. Use the Auxiliary Building Vent System.
c. Use only Containment Purge and Vent System.
d. Use only the Containment 2-inch Vent System.

41 . Given the following conditions:

- A LOCA has occurred
- Venting and filtration of containment atmosphere to control the hydrogen concentration is about to commence

What is the limiting factor in setting the containment atmosphere release flow rate?

The capability of the...

a. Shield Building Ventilation system to maintain a negative pressure in the Shield Building Annulus with respect to the Auxiliary Building.

b. Shield Building Ventilation system to maintain a negative pressure in the Shield Building Annulus with respect to the outside atmosphere.

c. Aux Building Special Ventilation system to maintain a negative pressure in the Auxiliary Building with respect to the outside atmosphere.

d. Aux Building Special Ventilation system to maintain a negative pressure in the Auxiliary Building with respect to the Shield Building Annulus.

42 . Given the following conditions:

- The plant is in COLD SHUTDOWN for an outage
- Containment Vent via the 36" RBV valves is being initiated
- The operator opened the Purge/Vent Exhaust Valves and dampers, and opened the Purge/Vent Supply valves and dampers
- At 0702 the operator took the Containment Vent Exhaust Fan & Damper control switch to ON
- The operator then had to respond to 47052-I STATION & INST AIR COMPRESSOR FAULT alarm
- At 0704 the operator took the Containment Purge/Vent Supply Fan & Damper control switch to ON

What was the flow rate for the air exhausted to the Reactor Bldg Discharge Vent when the operator completed these actions?

a. 0 cfm.

b. 4,000 cfm.

c. 33,000 cfm.

d. 37,000 cfm.

43. Given the following conditions:

- The plant is at 100% power
- Annunciator 47055-N SPENT FUEL POOL ABNORMAL is in alarm
- Spent Fuel Pool level is reported to be decreasing very rapidly

Which of the following would be the appropriate source to use for SFP makeup using the highest possible flow capacity?

a. RWST.

b. Blender.

c. Service Water.

d. Reactor Makeup Water.

44. Given the following conditions:

- The plant is at 75% power
- Failure of automatic control results in FW-7A, S/G A Main Feedwater valve going closed
- The operator takes manual control and rapidly opens the valve to near its previous position

Which of the following is an immediate result of re-opening the valve?

a. Pressurizer level increases due to increase in RCS Tavg.

b. Rods step out due to the rapid increase in reactor power.

c. S/G level shrinks due to the rapid addition of colder feedwater.

d. S/G level increases rapidly with higher moisture carryover in steam to the turbine.

45. Which of the following conditions does NOT result in an inoperable Turbine Driven (TD) AFW pump with reactor power at 20%?

a. MS-100A or MS-100B S/G Steam Supply to TD AFW Pump Isol is NOT fully open.

b. AFW-10A or AFW-10B AFW Train Crossover Valve is NOT fully open.

c. MS-1A or MS-1B S/G Main Steam Isolation Valve is NOT fully open.

d. MS-102 T/D AFW Pump Main Steam Isol is in PULLOUT.

46 . Given the following conditions:

- The plant is at 8% power
- Main turbine rollup completed at 1800 rpm
- Main Feedwater Pump A is operating and the Main Feed Control Bypass valves are in AUTO
- Steam Dump Control is in Steam Pressure mode with AUTO setpoint at 1005 psig

If main steam line pressure transmitter PT-484 fails high, which of the following will occur?

a. All Steam Dump Valves remain closed.

b. All Condenser Steam Dumps open but reclose when RCS temperature falls to 540°F.

c. All Condenser and all Atmospheric Steam Dump valves open but reclose when the SI signal occurs.

d. All Atmospheric Steam Dump valves open and pressure continues to fall until automatic MSIV closure occurs.

47 . With the plant at 98% power, what is the plant response to ONE Condenser Steam Dump valve failing open?

a. An increase in steam flow resulting in an increase in turbine load.

b. A decrease in Tavg resulting in control bank D rods stepping in.

c. A decrease in reactor power and an increase in S/G levels.

d. An increase in reactor power and a decrease in PRZR level.

48. Given the following conditions:

- The plant is at 100% power
- Both Heater Drain Pump Motors have uncoupled due to a fault in the HD Pump Control Cabinet.
- The following timeline for the event was recorded:
  01:00:00 - Annunciator 47063-P FEEDWATER HTR BYPASS ALERT alarmed
  01:00:20 - Annunciator 470461-G FWP A/B SUCTION PRESS LOW alarmed
  01:01:25 - Annunciator 470461-G cleared
  01:02:30 - Annunciator 47063-P cleared
  01:03:50 - Feedwater suction pressure stabilized at 270 psig

What Operator action should be taken?

a. Open C-701, Condenser Recirc Cont valve.

b. Trip ONE Feedwater Pump to initiate a VPL runback of the Turbine.

c. Verify C-13, Condensate Bypass LP FW Heaters valve, has opened.

d. Reduce Turbine load to less than 60% and stop ONE Feedwater Pump.

49. Given the following conditions:

- The plant was originally at 100% power
- Rod Control is selected to MANUAL
- BS-100A/CV-31167, Heater Bleed Steam Supply to FW Heater 15A and 15B valve fails closed

Which of the following describes the initial affect on the given plant parameters, and the actions the operator would take?

	Plant Efficiency	Action
a.	Increases,	NO action required.
b.	Decreases,	Increase power to 100% per plant procedures.
c.	Increases,	Decrease power to 100% power or less per plant procedures.
d.	Decreases,	Decrease power to 100% power or less per plant procedures.

50 . Given the following conditions:

- A plant startup is in progress at 12% reactor power
- Turbine power is at 63 MW (11%)
- Feedwater Control transfer to Main Feedwater Flow Control Valves has been completed and FW-7A and FW-7B controllers are in AUTO maintaining program level
- S/G A and B Bypass Flow Control Valve controllers are set at 35% level setpoint

What is the IMMEDIATE response of the feedwater control system if PT-485, Turbine Impulse Pressure instrument, fails low?

FW-7A/7B throttle to control S/G level at...

a. 33%, and NO annunciators alarm.

b. 33%, and only annunciators 47062-A[47062-D], S/G A[B] PROGRAM LEVEL DEVIATION alarm.

c. 44%, and only annunciators 47062-B[47062-E] S/G A[B] BYPASS CV LEVEL DEVIATION alarm.

d. 44%, and annunciators 47062-B[47062-E] S/G A[B] BYPASS CV LEVEL DEVIATION and 47062-A[47062-D], S/G A[B] PROGRAM LEVEL DEVIATION alarm.

51 . Given the following conditions:

- The plant is at HOT SHUTDOWN 48 hours after a plant trip
- ALL of the below conditions were identified immediately following the Trip and have NOT been corrected

Which of these conditions, if it continued, would require that the plant be cooled to below 350°F due to entry into a Limiting Condition for Operation?

a. AFW-10A, AFW Train A Crossover Valve, is closed.

b. ONE Turbine Overspeed Protection System is inoperable.

c. SW-601A, SW to Aux. Feedwater Pump 1A, is inoperable.

d. 39,000 gallons of water is available in the Condensate Storage Tanks.

52 . Given the following conditions:

- The plant is at 55% power
- S/G level channel LT-473 is removed from service per A-MI-87

If S/G level channel LT-471 fails high, what would be the status of feed for the S/Gs?

a. Both S/Gs are being fed from the motor-driven AFW Pumps.

b. Both S/Gs are being fed from the turbine-driven AFW Pump only.

c. Feed to both S/Gs increases as FW-7A/B, their respective S/G Main Feed valves, throttle open.

d. Feed to S/G B lowers due to throttling close of FW-7B, S/G B Main Feed valve. Feed to S/G A remains normal.

53 . Which of the following describes the electrical power sources to the Instrument Bus Inverters in order of priority?

	Most Preferred	Next Preferred	Least Preferred
a.	480 VAC,	125 VDC,	120 VAC.
b.	480 VAC,	120 VAC,	125 VDC.
c.	120 VAC,	480 VAC,	125 VDC.
d.	120 VAC,	125 VDC,	480 VAC.

54 . Given the following conditions:

- The plant is in INTERMEDIATE SHUTDOWN with the spare Charger off-site for repair.
- Annunciators 47105A BATTERY A ABNORMAL and 47104-A BATTERY A CHARGER TROUBLE are in alarm
- Both trains of RHR are operable with RHR in service controlling RCS temperature at 300°F
- S/G B is operable
- Battery A Charger is found to have tripped and must be replaced.

What action is required to provide charging capability to both batteries?

a. Install the charger from Battery C or D.

b. Align 250 VDC battery charger BRE-108 to Bus BRA-104.

c. If grounds do NOT exist on either train, cross-connect the 125 VDC Buses BRA-102 and BRB-102.

d. Take the plant to REFUELING shutdown conditions, and if grounds do NOT exist on either train, then cross-connect the 125 VDC Buses BRA-102 and BRB-102.

55 . Given the following conditions:

- Diesel Generator A (DG A) was running paralleled to its associated 4160 VAC Bus per SP-42-312A "Diesel Generator A Available Test"
- The Control Room operator was adjusting load and voltage when the Speed Control switch sticks in the LOWER position

Which of the following describes the FIRST action the operator should take?

a. Take the Control Room Diesel Engine A Control Switch to STOP/PULLOUT.

b. Take the Control Room DG A to Bus 5 supply breaker 1-509 to TRIP/PULLOUT.

c. Direct the local operator to take the DG Excitation and Control Panel Governor switch to RAISE.

d. Direct the local operator to take the DG Excitation and Control Panel (Voltage Control) Mode Selector switch to OFF.

56 . Given the following conditions:

- A LOCA has occurred, followed by a loss of offsite power
- Diesel Generator (DG) A is operating at 60.8 Hz
- The load on DG A is 3025 KW
- Bus 5 voltage is 4250 VAC

Which of the following actions would reduce the KW loading on D/G A by the largest amount?

a. Raising Bus 5 voltage and lowering DG speed.

b. Raising DG speed and stopping Containment Fan Coil Unit C.

c. Lowering Bus 5 voltage and raising DG speed.

d. Lowering Bus 5 voltage and stopping Containment Spray Pump A.

57 . During a radioactive waste release of the CVCS monitor tank to the circulating water system, R-18 Waste Discharge Liquid monitor alarms. The operator notes that the release is terminated.

How was the release terminated?

a. Liquid waste discharge valves WD-18 and WD-19 both auto closed.

b. Liquid waste discharge valve WD-19 only auto closed.

c. CVCS MT discharge valve CVC-918 auto closed.

d. The running CVCS Monitor Tank pump tripped.

58 . Given the following conditions:

- The plant is at 100% power
- S/G blowdown is in service in Mode II
- Condenser air removal is aligned for normal operation
- R-19 S/G Blowdown Liquid monitor detector has failed (readout low)
- Condenser Air Inleakage was in progress prior to the detector failure
- The NCO positions R-19 keyswitch to the OFF position

Which of the following describes the effect of the operator actions?

a. Bleed Steam supply to Heating Steam system isolates if the Heating Boiler is operating.

b. Condenser Air Ejector discharge is routed to the Aux. Building vent stack.

c. Blowdown flowpath switches to Mode I alignment.

d. Blowdown continues in its Mode II alignment.

59 . When the Waste Gas Decay Tank C is being discharged to atmosphere, what condition will automatically close WG-36/CV-31215, Gas Decay Tanks to Plant Vent?

a. Both Waste Gas Compressors trip.

b. Both Auxiliary Building Exhaust Fans trip.

c. P-21107, Waste Gas Vent Header pressure low.

d. R-13, Aux Building Vent Exhaust Monitor high alarm.

60 . Given the following conditions:

- Waste Gas Decay Tank C is aligned for cover gas
- Waste Gas Decay Tank B is aligned for fill
- Waste Gas Decay Tank A is selected for standby service

Which of the following conditions will cause the Waste Gas System to isolate the Waste Gas Decay Tank aligned for fill?

a. The Standby Selector Switch is placed in the "B" position.

b. Gas Decay Tank B pressure rises above 110 psig.

c. Gas Decay Tank C pressure falls below 8 psig.

d. Both Waste Gas Compressors trip.

61 . Which of the following describes an immediate action of A-RM-45, "Abnormal Radiation Monitoring System Operation", for the failure of R-1, Control Room Area radiation monitor, that results in a high alarm?

a. Direct unnecessary personnel to exit the Control Room.

b. Implement the actions of E-0-06 "Fire In Alternate Fire Zone".

c. Verify closed ACC-4, Control Room A/C Normal Recirc Damper.

d. Verify Control Room Ventilation shifts to the Post Accident Recirc mode of operation.

62 . Why is Air Compressor 1A stopped if cooling water flow to the compressor is lost?

The air compressor will trip...

a. due to overheating of the compressor motor resulting in overload.

b. due to seal leakage resulting in low air discharge pressure.

c. when the limit for air outlet temperature is exceeded.

d. when the limit for oil temperature is exceeded.

63 . Given the following conditions:

- 0100:00 - FP-331/CV-31377 Turbine Lube Oil Storage Tank Deluge valve failed opened (NO fire present)
- 0100:10 - Fire header pressure read 105 psig
- 0100:12 - Fire header pressure read 98 psig
- 0102:00 - FP-330 Manual Isolation for FP-331 was closed
- 0102:05 - Fire header pressure restored to normal

Which of the following describes the steps that must be taken to restore the Fire Pumps to their normal status once FP-331 is reset?

a. Fire Pump A and Fire Pump B control switches must be momentarily taken to STOP/SI RESET.

b. Fire Pump A and Fire Pump B control switches must be momentarily taken to STOP/SI RESET, and Fire Pump A local STOP pushbutton must be depressed.

c. Only Fire Pump A local STOP pushbuttons must be depressed.

d. Both Fire Pump A and Fire Pump B local STOP pushbuttons must be depressed.

64 . Which of the following will result in automatic actuation with IMMEDIATE discharge of its associated fire protection medium (i.e., water, halon, carbon dioxide)?

a. ONE temperature switch/thermostat actuates in Zone 103 Diesel Generator Room A.

b. TWO ionization (smoke) detector actuates in Zone 602 the Aux. Bldg. Record Storage Room.

c. ONE temperature rise detector (thermal pneumatic) actuates in Zone 1203 Materials Work Storage.

d. ONE photo-electric detector and ONE ionization (smoke) detector actuates in Zone 1103 QA/QC Vault.

65 . Given the following conditions:

- The plant is at 85% power (steady state) with burnup at 10,000 MWD/MTU
- Tavg is currently on program
- RCS boron is 800 ppm
- When the NCO initiated rod withdrawal for load pickup, TLA-1 ROD SUPERVISORY ALARM actuated
- Control Bank D Rod G-3 indicated 170 steps with the remaining Bank D rods indicating 190 steps
- Rod G-3 is found to be stuck at its current position

If the NCO inserts Control Bank D until TLA-1 just clears, how much dilution (in gallons) would need to be added to maintain current RCS temperature?
(Assume total bank worth when adjusting rods.)

a. 178-222 gallons.

b. 223-267 gallons.

c. 537-581 gallons.

d. 582-627 gallons.

66 . Given the following conditions:

- The plant is operating at 18% power
- The high pressure tap to RCS flow instrument FT-411 on loop A fails

What is the resulting plant condition, if NO operator action is taken?

a. All loop A flow indicators will read low, and the reactor trip is generated on RCS loop low flow.

b. All loop A flow indicators will read low, but the reactor trip is generated on low PRZR pressure.

c. Only FI-411 RCS flow indication will read low, and the reactor trip is generated on RCS loop low flow.

d. Only FI-411 RCS flow indication will read low, but the reactor trip is generated on low PRZR pressure.

67 . During a small break LOCA on a cold leg, a phase is reached where the vessel level continues to decrease below the hot leg penetrations and boiling in the core is the means of transporting the core heat to the bubble above the core. A fixed pressure differential exists between the core and the break and is maintained by the loop seal (in the intermediate leg). Natural circulation flow as a heat removal mechanism for the RCS has been lost.

Which of the following describes the main heat removal mechanism for the RCS?

a. Slug flow occurs via the cold legs through the loop seal and flashing occurs across the cold leg break.

b. Steam from the bubble condenses on the hot leg side of the S/G U-tubes which then drains back to the core via the hot legs.

c. Partial natural circulation flow, characterized by liquid pulses, flows from the cold leg over the S/G U-tubes and into the hot legs.

d. Steam from the bubble condenses in the reactor vessel head, which is cooled by fans in the containment, and drains back to the core.

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68 . Given the following conditions:

- A LOCA has occurred
- RCS pressure is 125 psig
- RCS Core Exit TCs read 380°F
- SI Pump A is running providing 325 gpm flow
- RHR Pump A is running providing 1150 gpm flow

What is the appropriate action taken in response to the above conditions?

Entry into FR-P.1 "Response to Pressurized Thermal Shock Condition" is...

a. NOT required since RCS pressure is below 350 psig.

b. made but NO actions are implemented before returning to procedure in effect.

c. made and a RCS temperature soak for a ONE hour period will be completed.

d. made and cooldown will continue within a limit of 50°F in any 60 minute period.

69 . Given the following conditions:

- A LOCA has occurred
- The crew has transitioned from E-1 "Loss of Reactor or Secondary Coolant" to FR-C.1 "Response to inadequate Core Cooling" due to a RED path on Status Trees
- Actions of FR-C.1 have been completed through the depressurization of both S/Gs
- NO RXCP is running
- S/G levels are at 18% NR

Which of the following conditions directly requires starting a RXCP even if normal support conditions are NOT established?

a. Level in both S/Gs fall to 15% and feed flow is lost.

b. Neither SI Pump flow NOR RHR pump flow is indicated.

c. Both PRZR PORVs and the RCS vents have been opened.

d. Core Exit TC temperature indications read greater than 1200°F.

70 . Given the following conditions:

- The plant is in HOT SHUTDOWN
- All control rods are inserted
- Reactor trip breakers are open.

Which of the following would result in the earliest required shutdown of the running RXCP B?

a. CC-610B/CV-31128, RXCP B Thermal Barr Comp Cooling Return, goes closed.

b. CC-601B/MV-32085, Component Cooling to RXCP B, is inadvertently closed and CANNOT be reopened.

c. RXCP B No. 1 seal leakoff flow is 6.0 gpm with RXCP bearing water temperature at 215°F and decreasing.

d. Seal Injection flow to the RXCP is lost due to a blocked seal injection filter with #1 seal outlet temperature rising to 228°F.

71 . Given the following conditions:

- The reactor is at 9% power during a startup
- Turbine load is 7% (37 MW)
- Steam Dumps are in TAVG Mode
- RXCP A trips
- The reactor is NOT tripped

Which of the following sets of conditions describes the expected condition for the parameters listed below?

	Actual Reactor Power	Steam Flow for S/G A	Steam Flow for S/G B
a.	DECREASE,	DECREASE,	DECREASE.
b.	CONSTANT,	INCREASE,	INCREASE.
c.	CONSTANT,	DECREASE,	INCREASE.
d.	DECREASE,	DECREASE,	CONSTANT.

72 . Given the following conditions:

- The plant is in HOT SHUTDOWN following a reactor trip from 100% power
- All primary and secondary parameters are being maintained at their no-load values
- Charging Pump relief valve lifts causing a loss of charging and seal injection flows

What occurs within the given time frame if the operator fails to take action to stop letdown flow?

a. LD-2/CV-31104 and LD-3/CV-31108, Letdown Isolation valves, will go closed in approximately 3 minutes.

b. Annunciator alarm 47043-C PRESSURIZER LEVEL DEVIATION will actuate in approximately 10 minutes.

c. The operator will be required to place all PRZR heaters in OFF position in approximately 11 minutes.

d. The operator will be required to manually initiate Safety Injection in approximately 16 minutes.

73 . Given the following plant conditions:

- The plant is at 50% power
- Seal injection flow has been lost to the RXCPs
- CVC-207A, RXCP A #1 Seal Leakoff Isolation has been closed due to a high RXCP seal leakoff flow alarm
- CVC-204A RXCP Seal Supply Line Throttle as been closed
- After 15 minutes, the line blockage was cleared and seal injection is ready to be restored to RXCP A
- The operator then reopens CVC-204A to its original position.

What is the result of his actions?

The RXCP...

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

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

c. seal package may be damaged as it undergoes a great than 1°F/minute cool down.

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

74 . Given the following conditions:

- The plant is at HOT SHUTDOWN, 12000 MWD/MTU
- RCS Boron is 940 ppm
- 47042G, AUCTIONEERED TAVG-TREF DEVIATION, has alarmed
- 47043H, RCS LOOP 1A TAVG LO-LO, alarms
- 47043G, RCS LOOP 1B TAVG LO-LO, alarms
- All RCS temperature indications are decreasing due to an uncontrolled cooldown
- Boric Acid Tank A level indication is 56% as indicated on LI-172

Following completion of the emergency boration, what would the tank level read?

a. 53%.

b. 50%.

c. 42%.

d. 28%.

75 . When performing an emergency boration in accordance with FR-S.1, "Response to Nuclear Power Generation/ATWS", the operator is directed to manually start SI Pumps if charging flow is inadequate.

What is the PRIMARY reason Safety Injection is NOT manually initiated?

a. Containment process radiation monitors would be isolated.

b. Instrument Air to containment would be isolated.

c. The Main Feedwater Pumps would be tripped.

d. RXCP seal return flow would be interrupted.

76 . Given the following conditions:

- RCS temperature is 118°F.
- The reactor vessel head is removed.
- Reactor vessel upper internals are installed in the reactor vessel.
- Refueling level is 10.25%
- RCS draining is proceeding at 10 gpm
- RHR Pump A is running with indicated flow of 2000 gpm
- RHR Pump A begins to exhibit indications of cavitation

Which of the following explains why cavitation of the RHR Pump occurred in this situation?

a. Steam binding of the RHR pump occurred due to low recirculation flow.

b. Draining of water from the S/G tubes has increased the RHR suction temperature.

c. Draining with the upper internals in place has reduced the RHR discharge pressure.

d. Air entrainment at the RHR suction inlet has occurred due to the high flow conditions.

77 . Given the following conditions:

- A loss of all AC power has occurred
- After 10 minutes, power was restored to Busses 5 and 6
- The actions of ECA-0.1 are being performed to start a CC Pump

Why are CC-613A and B, RXCP CC Return Manual Isolation Valves verified closed prior to restarting the CC Pump?

a. Reduce CC heat loads to the minimum based on SW loads.

b. Protect CC availability by precluding steam formation in the CC piping.

c. Prevent damage to the RXCP bearings due to excessive cooldown rate.

d. Maximize flow to the CVCS components for reestablishing charging, letdown and seal return.

78 . Given the following conditions:

- The plant is at 25% power
- The only available Feedwater Pump tripped
- The reactor and turbine failed to trip manually
- S/G narrow range levels are now offscale low
- Control rods are being inserted manually
- Unloading rate is 20%/minute

Which of the following correctly describes of the expected response of the ATWS Mitigation System Actuation Circuitry (AMSAC)?

AMSAC...

a. will NOT trip the turbine nor actuate diverse circuits to trip the reactor trip breakers because reactor power will be below P-10, but will start only the motor-driven AFW Pumps.

b. will NOT trip the turbine nor actuate the circuits to remove excitation from the Rod Drive MG sets because power will be below P-7, but will start all AFW Pumps.

c. will actuate diverse circuits to trip the reactor trip breakers, trip the turbine and start only the motor-driven AFW Pumps.

d. will trip the turbine, actuate circuits to remove excitation from the Rod Drive MG sets, and start all AFW Pumps.

79 . Given the following conditions:

- Reactor power is 100%
- Reactor trip breaker testing is being performed with Reactor Trip Bypass Breaker A (52/BYA) racked in and closed
- Both Reactor Trip Breakers (52/RTA and 52/RTB) are closed
- The NAO racks in and closes Reactor Trip Bypass Breaker B (52/BYB)
- Breakers 52/RTB and 52/BYA open

Which of the following describes the response to this condition?

The reactor is...

a. NOT tripped, and the NCO should manually trip the reactor.

b. NOT tripped as this is the expected response when 52/BYB was closed.

c. tripped and the NCO should direct the NAO to locally open both 52/RTA and 52/BYB.

d. tripped and the NCO should manually trip the reactor as directed by E-0 "Reactor Trip Or Safety Injection".

80 . Given the following conditions:

- Refueling operations are in progress
- An irradiated fuel assembly is located in the Spent Fuel Pool (SFP) upender
- The conveyor car & fuel assembly container is in the UP position in preparation for relocation of the assembly to the Spent Fuel Pool storage racks.
- The assembly has been latched with the SFP tool and raised from the fuel assembly container
- Water level in the reactor cavity and SFP is rapidly decreasing
- The following annunciators have alarmed:
  - 47031-R REACTOR CAVITY SUMP LEVEL HIGH/LOW
  - 47031-Q CONTAINMENT SUMP A LEVEL HIGH
  - 47031-P CONTAINMENT SUMP A LEVEL HI-HI
  - 47055-N SPENT FUEL POOL ABNORMAL

According to E-FH-53B, "Loss of Reactor Cavity Inventory," what are the actions required in the Spent Fuel Pool area for this event?

a. Evacuate all personnel.
   Stop the SFP ventilation systems.

b. Move the assembly to the nearest empty spent fuel rack.
   Lower the fuel assembly in the rack location.
   Close the Fuel Transfer System Gate Valve.
   Evacuate all personnel.

c. Evacuate non-essential personnel.
   Move the fuel assembly to the area north of the conveyor.
   Lower the fuel assembly to the floor.
   Close the Fuel Transfer System Gate Valve.
   Evacuate remaining personnel.

d. Evacuate non-essential personnel.
   Unlatch the assembly.
   Lower the upender frame.
   Transfer the conveyer car to the containment.
   Evacuate remaining personnel.

81 . Given the following conditions:

- A S/G tube rupture has occurred in S/G A
- SI is actuated
- The required RCS cooldown has been completed per E-3 "Steam Generator Tube Rupture"
- RCS subcooling is 75°F
- PRZR pressure is 1600 psig
- PRZR level is stable at 20%
- PRZR backup heaters are in PULLOUT
- Charging Pump A is running in MAN
- Charging flow is 52 gpm
- S/G A pressure is 1020 psig
- S/G B pressure is 675 psig

What action should be performed at these conditions to minimize leakage flow from the RCS to the Steam Generator?

a. Terminate SI and stop SI Pumps.

b. Manually throttle open PRZR sprays.

c. Increase feedwater flow to the S/G A.

d. Lower the Steam Dump steam release pressure setpoint

82 . Given the following conditions:

- A plant startup is in progress
- S/G levels are at 33%
- Reactor power is at 8%
- One Main Feedwater Pump is running
- Feedwater Bypass Valves are in AUTO maintaining S/G levels

If a steamline leak results in a 50% step increase in steam flow, what is the FIRST indication the NCO will observe?

a. S/G pressure indications will rise.

b. Feedwater flow indications will lower.

c. S/G narrow range level indications will drop.

d. Feedwater temperature indications will increase rapidly.

83 . Given the following conditions:

- An SI has occurred due to multiple steam leaks
- Train B SI failed to initiate either automatically or manually
- The crew has started the necessary Train B equipment as directed by E-0 "Reactor Trip or Safety Injection"
- Selected plant parameters read:

Instrument Channel -	I	II	III	IV
- RCS Tavg:	541°F	538°F	541°F	539°F
- PRZR pressure:	1834 psig	1830 psig	1832 psig	1828 psig
- Cntmt pressure:	5.2 psig	5.3 psig		5.5 psig
		5.2 psig	60.0 psig	5.3 psig
- S/G A pressure:	800 psig	795 psig	800 psig	
- S/G A steam flow:	0.725×10^6 lbs/hr	0.730×10^6 lbs/hr		
- S/G B pressure:	785 psig		780 psig	770 psig
- S/G B steam flow:			0.751×10^6 lbs/hr	0.745×10^6 lbs/hr

Assuming NO operator action has been taken concerning the MSIVs, which of the following describes their expected positions under these conditions?

	MSIV A	MSIV B
a.	Open	Open.
b.	Open	Closed.
c.	Closed	Open.
d.	Closed	Closed.

84 . Given the following conditions:

- Plant has been operating at 75% power
- The following annunciators go into alarm
  - 47061-D FEEDWATER PUMP B TRIP
  - 47062-A S/G A PROGRAM LEVEL DEVIATION
  - 47062-C S/G A LEVEL LOW
  - 47062-D S/G B PROGRAM LEVEL DEVIATION
  - 47062-F S/G B LEVEL LOW
  - 40763-F FEEDWATER PUMP B ABNORMAL
  - 47064-D S/G B LEVEL LOW LOW
- S/G A levels read 29%, 31% & 31% across the board
- S/G B levels read 0%, 29% & 29% across the board
- Turbine load has stabilized at 60%

Which of the following identifies the action the NCO is required to take?

a. Depress CV LOWER pushbutton until plant load is reduced to less than 50%.

b. Manually trip the reactor and enter E-0 "Reactor Trip Or Safety Injection".

c. Control FW-7B Main Feedwater Control Valve in MAN.

d. Verify BOTH motor-driven AFW pumps have started.

85 . Given the following conditions:

- A loss of offsite power has occurred
- The plant is in HOT SHUTDOWN
- PRZR pressure is at 2000 psig
- Bus 5 and Bus 6 are deenergized due to failures of their respective Diesel Generators
- System Operating has cleared all faulted lines and is ready to restore power to Kewaunee switchyard
- Power will be restored through Q-303 from Point Beach
- NO electrical faults exist on Kewaunee plant equipment

What is the sequence of reenergizing the transformers and repowering the 4160 KV buses?

a. The Tertiary Auxiliary Transformer (TAT) is energized.
Bus 5 is aligned to the TAT.
Bus 6 is aligned to the TAT.
The Main Auxiliary Transformer (MAT) is energized.
Buses 1, 2, 3 and 4 are aligned to the MAT.

b. The Tertiary Auxiliary Transformer (TAT) is energized.
The Main Auxiliary Transformer (MAT) is energized.
Bus 5 is aligned to the TAT.
Bus 6 is aligned to the TAT.
Buses 1, 2, 3 and 4 are aligned to the MAT.

c. The Tertiary Auxiliary Transformer (TAT) is energized.
Bus 5 is aligned to the TAT.
Bus 6 is aligned to the TAT.
The Reserve Auxiliary Transformer (RAT) is energized.
Buses 1, 2, 3 and 4 are aligned to the RAT.

d. The Tertiary Auxiliary Transformer (TAT) is e.nergized.
The Reserve Auxiliary Transformer (RAT) is energized.
Bus 5 is aligned to the TAT.
Bus 6 is aligned to the RAT.
Buses 1, 2, 3 and 4 are aligned to the RAT.

86 . Given the following conditions:

- A loss of offsite power has occurred
- Diesel Generator B is out of service
- Diesel Generator A tripped
- Annunciator 47091-B DIESEL GEN A MECH LOCKOUT has actuated
- The local operator reports each of the of the alarms listed below are lit

Which of the following, if it alone were cleared, would allow a restart of Diesel Generator A?

a. Diesel generator annunciator DR101-11, HIGH WATER TEMP.

b. Diesel engine control panel D-1A L7, JACKET WATER PRESSURE lamp.

c. Diesel generator annunciator DR101-21, LOW LUBE OIL LEVEL.

d. Diesel engine control panel D-1A L10, LOW AIR PRESSURE SYSTEM #2 (PRIMARY) lamp.

87 . Given the following conditions:

- A fire has occurred in the Control Room
- The actions of E-0-06 "Fire In Alternate Fire Zone" are being performed:
  - The CRS has just started Diesel Generator 1A
  - NCO A has transferred controls to the Dedicated Shutdown Panel (DSP)
  - SD3A/CV-31170 controller at the DSP is set to 0% demand
  - All other local actions are complete
  - A loss of offsite power then occurs

What would NCO A expect to see concerning RCS temperature control following the loss of offsite power?
(Assume the fire has NOT caused any spurious actuations)

a. Both S/G PORVs open to maintain RCS temperature about 549°F.

b. Only S/G PORV SD-3B/CV-31174 opens to maintain RCS temperature about 549°F.

c. Only S/G PORV SD-3A/CV-31170 opens to maintain RCS temperature about 552°F.

d. The first set of Main Steam Safety Valves opens to maintain RCS temperature about 556°F.

88 . Given the following conditions:

- Plant has tripped and an SI signal has been generated
- The Train A (slave) relays (ESFAS) to the Service Water System valves have failed to operate (SW was only system affected)

What is the status of cooling to Component Cooling Heat Exchanger A, if NO operator action is taken?

a. There is NO Service Water flow through the heat exchanger.

b. Flow will be lower than expected but will be at a set constant value.

c. Flow through the CCW heat exchanger will be at its post-accident expected value.

d. Flow will be lower than expected and controlled by the CCW outlet header temperature.

89 . Given the following conditions:

- Plant has been tripped due to a loss of Instrument Air pressure.
- Air pressure has been restored.
- Equipment restoration is in progress.

Following the realignment of Internal Containment Spray (ICS) suction piping, what additional action should be taken?

a. Flush the ICS suction piping.

b. Drain ICS pump discharge piping.

c. Verify flow through recirculation lines.

d. Stroke test CI-1001A and B, Caustic Line Control Valves.

90 . Given the following conditions:

- The plant is in HOT SUTDOWN
- E-0-06 "Fire In Alternate Fire Zone" has been implemented due to a fire
- Control Operator A is ready to establish charging flow from the Dedicated Shutdown Panel (DSP)
- All DSP controls are in their expected position (control aligned to DSP; those controls NOT already positioned by operator are in the position designated in N-MI-87 CLA, "Dedicated Shutdown System Periodic Checklist"
- Fire conditions have NOT affected any component operation.

Prior to Control Room evacuation the following were the charging conditions:

- Charging Pump A was running in MAN
- Charging Pump C was running in MAN
- Total charging flow (regen HX and seal injection) was 86 gpm
- Charging Pump A was providing 40% of the total flow

What is the charging flow expected to be when the Charging Pump is started from the DSP?

a. 15 gpm.

b. 49 gpm.

c. 52 gpm.

d. 60 gpm.

91 . Given the following conditions:

- A LOCA is in progress
- The reactor is tripped
- Core exit thermocouples read 1205°F
- RCS pressure is 900 psig

Assuming each of the following is available, what is the preferred method that provides the most effective means of cooling the core?

a. Starting a Reactor Coolant Pump.

b. Establishing high pressure safety injection flow.

c. Injecting the SI Accumulators by opening the PRZR PORVs to reduce RCS pressure.

d. Injecting the SI Accumulators by dumping steam from the S/Gs to reduce RCS pressure.

92 . What is the detector type and sensing location for R-9, RCS Letdown Radiation monitor?

a. A scintillation detector located near the letdown line upstream of the reactor coolant filter.

b. A scintillation detector located near the letdown line downstream of the reactor coolant filter.

c. A Geiger Mueller detector located near the letdown line upstream of the reactor coolant filter.

d. A Geiger Mueller detector located near the letdown line downstream of the reactor coolant filter.

93 . Given the following conditions:

- A turbine trip resulted in premature opening of one PRZR safety valve and a steamline break inside the reactor containment.
- The operators have performed all the actions of the following
  - E-0 "Reactor Trip Or Safety Injection"
  - E-2 "Faulted Steam Generator Isolation"
  - ES-1.3 "Transfer to Containment Sump Recirculation"
- The faulted S/G is at ZERO psig
- The primary safety reclosed after a period of time
- RCS pressure is 1950 psig and stable
- PRZR level is at 32%
- RCS subcooling is 125°F
- Intact S/G level is 22% with AFW flow controlled by the operator to maintain this level
- Containment pressure rose to 24 psig and is now at 8 psig

When the crew transitions from E-1 "Loss of Reactor Or Secondary Coolant" to ES-1.1 "SI Termination", what actions will be taken in securing the ECCS Pumps?

a. The SI and RHR pumps will be stopped first, charging flow established and then the ICS stopped.

b. The SI and RHR pumps will be stopped, charging flow established but the ICS pumps will continue to run.

c. The SI pumps will be stopped, charging will be established, the ICS pumps and the RHR pumps will continue to run.

d. The SI pumps will be stopped, charging will be established, the ICS pumps will be stopped and then the RHR pumps will be stopped.

94 . Why is SI-5A/MV-32107, SI Pump A Suction Isolation valve, closed prior to opening RHR-299A/MV-32134, Residual Heat Exchanger Outlet to Safety Injection Pump 1A valve, during the recirculation phase of a LOCA?

a. Reduces system pressure to the SI Pump suction header resulting increased SI flow due to decreased RHR Pump NPSH requirement.

b. Prevents tripping of the SI pumps on high discharge flow rate resulting from increased SI Pump recirculation flow.

c. Protects the SI Pump suction header from overpressurization if the RHR pump was aligned to the RCS hot leg.

d. Prevents the RHR Pump from recirculating contaminated Sump water directly to the RWST.

95 . Given the following conditions:

- An inadvertent Safety Injection signal tripped the reactor
- All equipment responded as required with the following exception:
  LD-4B/CV-31232, Letdown Orifice B Isolation, failed to close.
- PRZR level fell to 20%; is now at 23% and rising.

If the CVCS letdown line were to break just outside of the containment penetration, which of the following conditions could be expected if the operators failed to respond?

a. Rapid core uncovery with fuel damage would occur.

b. SI injection flow to RCS Loop B Cold Leg would be lost.

c. Recirculation capability from the Containment Sump B would be lost.

d. The charging line penetration to the RCS would undergo thermal shock.

96 . Given the following conditions:

- A LOCA outside containment has occurred
- SI was manually actuated.
- The crew has completed ECA-1.2, "LOCA Outside Containment," and transitioned ECA-1.1, "Loss of Emergency Coolant Recirculation"

Why is subcooling minimized once cooldown has been started?

a. It allows the operator to stop all ECCS pumps.

b. It allows RHR to be placed in service in cooldown mode earlier.

c. This conserves the inventory of the RWST during the injection phase.

d. This lowers the RCS pressure reducing the amount of RCS inventory loss.

97 . Given the following conditions:

- A main steam line break occurred inside containment
- The actions of FR-P.1, "Response To Imminent Pressurized Thermal Shock Condition" are being performed
- A RCS temperature soak is required and has been initiated
- NO RXCPs are running

Which evolution would be permitted during the soak period?

a. Energize all PRZR heaters.

b. Place PRZR auxiliary spray in service.

c. Raise the non-faulted S/G level from 18% to 40%.

d. Lower the S/G PORV controller pressure setpoint 25 psi.

98 . Given the following conditions:

- At 0630 hours a reactor trip with SI occurred
- At 0700 hours the crew enter ECA-1.1, "Loss of Emergency Coolant Recirculation"
- At 0730 hours the following conditions exist:
  - RCS subcooling is 52°F
  - RCS pressure is 2100 psig
  - RCS is on natural circulation
  - SI Pump A is running

How would operators establish the required SI flow to the RCS to meet the requirements of Figure ECA-1.1-1? (For USAR Figure 14.2.4-1, assume the SI pumps provide equal flow at the given pressure and assume 8.3 lbs/gallon.)

a. Throttle the running SI Pump discharge valve.

b. Start the second SI Pump and operate at full flow.

c. Start the second SI Pump and throttle its discharge valve.

d. With the ONE SI Pump running, NO further action is required.

99 . Given the following conditions:

- The plant has tripped from 100% power
- Following the trip, 47011B, RADIATION INDICATION HIGH annunciator actuated
- The SER indicates R-13 Aux Building Vent Radiation High

Which of the following describes the plant response?

a. All running Aux Building Supply and Exhaust Fans stop.
SFP Charcoal Filters outlet damper opens and bypass dampers close.
R11/12 sample discharge is routed to containment.
Train A Zone SV Exhaust Fan and Train A Safeguards Fan Coils Units start

b. All running Aux Building Supply and Exhaust Fans stop.
SFP Charcoal Filters outlet damper opens and bypass dampers close.
Train B Zone SV Exhaust fan and Train B safeguards fan coils units start.

c. Aux Building Supply Fan A and Aux Building Exhaust Fan A only stop.
SFP Exhaust Fan A starts.
R11/12 sample discharge is routed to containment.
Train A Zone SV Exhaust Fan and Train A Safeguards Fan Coils Units start.

d. All running Aux Building Supply and Exhaust Fans stop.
Both SFP Exhaust Fan and the SFP Supply Fan start.
Both Trains of Zone SV Exhaust Fans and Safeguards Fan Coils Units start.

100 . Given the following conditions:

- A LOCA has occurred
- The crew is performing cooldown as directed by ES-1.2 " Post LOCA Cooldown And Depressurization"
- ECCS Pumps are still operating in injection phase
- The ICS system has been stopped
- TWO Containment Cooling Fan Coil Units are running
- Containment pressure is stable at 2.2 psig
- The CRS transitions to FR-Z.3 "Response to High Containment Radiation Level" in response to a YELLOW path condition

In addition to running ICS, what action does the CRS direct in FR-Z.3 in order to help reduce containment radiation levels?

a. Initiate Containment Post Accident Vent.

b. The idle Containment Cooling Fan Coil Units are started.

c. A RHR Pump is started and aligned to supply the associated ICS header.

d. ONE train of venting and filtering containment atmosphere through Shield Building Vent is initiated.

Reactor Operator Examination

1 . a		RO Value	SRO Value			
KA	GENERIC 2.1.1	3.7	3.8	Ref. Section	Ref. Page	Ref. Revision
Refs.	Shift Operation and Turnover		NAD 3.17	5.1.5	3	C

Objs. 1190020301K01 - EXPLAIN THE SHIFT ORGANIZATION DURING ALL PLANT CONDITIONS IN ACCORDANCE WITH NAD 3.17.

1190060302K01 - Explain the shift organization during all plant conditions in accordance with NAD 3.17.

2 . a		RO Value	SRO Value			
KA	GENERIC 2.1.21	3.1	3.2	Ref. Section	Ref. Page	Ref. Revision
Refs.	Procedure Use and Adherence		GNP 03.01.03	6.1.3	2	C
	Waste Gas System Leakage Test		RT-GWP-32B	1.1	1	C

Objs. 1190180304K02 - Explain the expectations for the implementation of Operation Procedures in accordance with Operations Department Instructions.

3 . d		RO Value	SRO Value			
KA	GENERIC 2.1.30	3.9	3.4	Ref. Section	Ref. Page	Ref. Revision
Refs.	Fire In Alternate Fire Zone		E-0-06	Steps 21, 28, 29 & 42	19-20, 26, 27-28, 34	M
	Charging and Volume Control Prestartup Checklist		N-CVC-35B-CL	Section 4.4	8-10	AG

Objs. 0350000001K05 - LOCATE ALL SYSTEM CONTROLS AND INDICATIONS WHILE OPERATING THE CHEMICAL AND VOLUME CONTROL - CHARGING AND LETDOWN SYSTEM FROM THE CONTROL ROOM AND THE DEDICATED SHUTDOWN PANEL.

0350000001K06 - DISCUSS THE DESIGN CHARACTERISTICS OF THE CVC SYSTEM FOR ANY MODE OF OPERATION PER SYSTEM DESCRIPTION 35 AND USAR SECTIONS 9.2, 10A.5, AND 14.1.

4 . b		RO Value	SRO Value			
KA	GENERIC 2.2.1	3.7	3.6	Ref. Section	Ref. Page	Ref. Revision
Refs.	Reactor Startup		N-CRD-49B	21.c	10	X
	Reactor Data Manual			RD 5.1.1.1	1	May 19, 2000

Objs. 0490010101A05 - Withdraw the Control Banks during a reactor startup, in accordance with N-CRD-49B.

5 . a		RO Value	SRO Value			
KA	GENERIC 2.2.30	3.5	3.3	Ref. Section	Ref. Page	Ref. Revision
Refs.	Reactor Cavity Draining with Fuel or Upper Internals Installed		N-FH-53E	2.6 & 3.8	1-2	H
	KNPP Technical Specifications			3.8.3	TS 3.8-1	Amend 132

Objs. 0530000001K05 - LOCATE ALL SYSTEM CONTROLS AND INDICATIONS WHILE OPERATING IN SUPPORT OF THE REFUELING FROM THE CONTROL ROOM.

0530080101A01 - GIVEN FUEL OR UPPER INTERNALS INSTALLED, DRAIN THE REACTOR CAVITY IN ACCORDANCE WITH N-FH-53E.

6 . d		RO Value	SRO Value			
KA	GENERIC 2.2.33	2.5	2.9	Ref. Section	Ref. Page	Ref. Revision
Refs.	Rod Control and Rod Position Indication (CRD)		System Description Number 49	3.6.4 & 3.6.5	49-19 & 20	A

Objs. 0490000001K02 - DISCUSS THE DESIGN CHARACTERISTICS OF THE ROD CONTROL SYSTEM, FOR ANY MODE OF OPERATION, PER SYSTEM DESCRIPTION 49 AND USAR SECTIONS 1.2, 1.3, 1.8, 3.1, 3.2, 7.2, 7.3, 14.1 AND 14.2.

0490000001K04 - EXPLAIN THE RESPONSE TO THE OPERATION OF ALL ROD CONTROL SYSTEM CONTROLS IN ACCORDANCE WITH THE FOLLOWING LOGIC DIAGRAMS/SCHEMATICS: XK100-151, XK100-552, XK100-553, E2044, AND THE SYSTEM DESCRIPTION BLOCK DIAGRAMS.

7 . a		RO Value	SRO Value			
KA	GENERIC 2.3.1	2.6	3.0	Ref. Section	Ref. Page	Ref. Revision
Refs.	Radiation Work Permit		NAD 8.3	3.3	1	F
	Radiation Worker Training		T-GET-LP RWT	RWP - II.C	71-72	B

Objs. T-GET-LP RWT: Enabling Objective 10. - Interpret and apply information found on a RWP to a task in a radiologically controlled area

T-GET-LP RWT: Learning Objectives for Entering and Exiting the Radiologically Controlled Area: RWP - *State the types of RWPs and the function of each.

8 . d		RO Value	SRO Value			
KA	GENERIC 2.3.2	2.5	2.9	Ref. Section	Ref. Page	Ref. Revision
Refs.	ALARA Program		NAD-01.23	5.5 & 5.7	2	B
	Radiation Worker Training		T-GET-LP RWT	ALARA - III.	31-35	B

Objs. T-GET-LP RWT: Enabling Objective 6. - Apply basic methods to minimize radiation exposure to a given scenario

T-GET-LP RWT: Learning Objectives for ALARA: Calculate stay time given a dose rate, current exposure, and an exposure limit.

9 . d

		RO Value	SRO Value	Ref. Section	Ref. Page	Ref. Revision
KA	GENERIC 2.3.11	2.7	3.2			
Refs.	Offsite Dose Calculation Manual			Specification 3.1 & Table 3.1	3-2, 3-13	8
	Radiological Liquid Discharges (Batch Mode)	SP 32A-136		3.4	2	U

Objs. 32A0110104A01 - WHEN DIRECTED, DISCHARGE THE WASTE CONDENSATE TANKS TO THE AUXILIARY BUILDING STANDPIPE IN ACCORDANCE WITH N-LWP-32A-3.

10 . d

		RO Value	SRO Value	Ref. Section	Ref. Page	Ref. Revision
KA	GENERIC 2.4.12	3.4	3.9			
Refs.	User's Guide for Integrated Plant Emergency Operating Procedures	UG-0		k.5, n	3, 5	C

Objs. E000020501K03 - LIST THE MAJOR STEPS IN THE HIGH LEVEL ACTION SUMMARY FOR E-0, "REACTOR TRIP OR SAFETY INJECTION," PER THE IPEOP BACKGROUND DOCUMENT.

E000020501K04 - LIST THE IMMEDIATE OPERATOR ACTIONS, GIVEN A REACTOR TRIP, IN ACCORDANCE WITH E-0.

11 . d

		RO Value	SRO Value	Ref. Section	Ref. Page	Ref. Revision
KA	GENERIC 2.4.21	3.7	4.3			
Refs.	User's Guide for Integrated Plant Emergency Operating Procedures		UG-0	I.1, 2, 5 & 9	4	C
	Heat Sink		F-0.3		1	D

Objs. 46A0000001K02 - Discuss the design characteristics of the Honeywell Plant Computer, for any mode of operation, per System Description 46A.

12 . a

		RO Value	SRO Value	Ref. Section	Ref. Page	Ref. Revision
KA	GENERIC 2.4.26	2.9	3.3			
Refs.	KNPP Fire Plan			12.5	35-36	3
	Fire Plan Operability, Surveillance & Contingency Requirements		FPP 08-01	Table 1 (Bottom of page)	7	C

Objs. 0080000004K08 - Identify the components that affect the Fire Protection system operability per the Fire Plan

13 . a

		RO Value	SRO Value	Ref. Section	Ref. Page	Ref. Revision
KA	GENERIC 2.4.35	3.3	3.5			
Refs.	POST-LOCA Hydrogen Control		N-RBV-18C	4.1.1.b	2	K

Objs. 0180000004K05 - LOCATE ALL LOCAL CONTROLS AND INDICATIONS FOR THE RBV SYSTEM PER M-547 AND M-602.

0180010104A01 - WHEN DIRECTED, STARTUP THE POST-LOCA HYDROGEN CONTROL SYSTEM FOR HYDROGEN DILUTION OF CONTAINMENT WITH INSTRUMENT AIR AVAILABLE IN ACCORDANCE WITH N-RBV-18C.

14 . b

			RO Value	SRO Value			
KA	001	K2.05	3.1*	3.5	Ref. Section	Ref. Page	Ref. Revision
Refs.	System Description - Rod Control Number 49 & Rod Position Indication (CRD)				3.2	4	A
	Circuit Diagram - 4160 & 480 V Power Sources			E-240			AQ
	Circuit Diagram - DC Aux. And Emergency AC			E-233			AP

Objs. 0490000001K02 - Discuss the design characteristics of the Rod Control System, for any mode of operation, per System Description 49 and USAR sections 1.2, 1.3, 1.8, 3.1, 3.2, 7.2, 7.3, 14.1 AND 14.2.

0490000004K06 - List the power supplies for the following components per N-CRD-49-CL: Rod Drive MG Sets

Contributor to Core Damage Sequence: Loss of AC Bus – 2%

15 . a

			RO Value	SRO Value			
KA	001	K3.02	3.4*	3.5	Ref. Section	Ref. Page	Ref. Revision
Refs.	Continuous Rod Withdrawal			E-CRD-49B	2.0, 3.2	1, 2	H
	KNPP USAR				14.1.2	14.1-5 - 7	15

Objs. 0490000001K02 - DISCUSS THE DESIGN CHARACTERISTICS OF THE ROD CONTROL SYSTEM, FOR ANY MODE OF OPERATION, PER SYSTEM DESCRIPTION 49 AND USAR SECTIONS 1.2, 1.3, 1.8, 3.1, 3.2, 7.2, 7.3, 14.1 AND 14.2.

16 . d

			RO Value	SRO Value	Ref. Section	Ref. Page	Ref. Revision
KA	002	A3.03	4.4	4.6			
Refs.	Integrated Logic - RCS			E-2038			Z
	Logic Diagram - PRZR Pressure & Level Control			XK100-154			4

Objs. 0360000001K17 - EXPLAIN THE RESPONSE TO THE OPERATION OF ALL PRESSURIZER PRESSURE CONTROL SYSTEM CONTROLS IN ACCORDANCE WITH LOGIC DIAGRAMS/SCHEMATICS E2037, E2038, E2039, E2042, XK100-148, XK100-150, XK100-154, XK100-155, AND XK100-546.

17 . b

			RO Value	SRO Value	Ref. Section	Ref. Page	Ref. Revision
KA	002	K3.03	4.2	4.6			
Refs.	Flow Diagram - RCS			OP XK100-10			BE
	Reactor Coolant Leak			A-RC-36D	4.3	5	AB

Objs. 0360030401A01 - GIVEN AN UNIDENTIFIED RCS LEAK, RESPOND IN ACCORDANCE WITH A-RC-36D.

0360000001K32 - DISCUSS THE DESIGN CHARACTERISTICS OF THE REACTOR COOLANT SYSTEM FOR ANY MODE OF OPERATION PER SYSTEM DESCRIPTION 36 AND USAR SECTIONS 4.0 AND 6.5.

Contributor to Core Damage Sequence: Small LOCA – 7%

18 . a			RO Value	SRO Value			
KA	003	A1.06	2.9	3.1	Ref. Section	Ref. Page	Ref. Revision
Refs.	Reactor Coolant Pump Operation			N-RC-36A	4.3.2	4	Y
	System Description - Reactor Coolant (RC)			Number 36	3.6.3	36-27	A

Objs. 0360000001K32 - DISCUSS THE DESIGN CHARACTERISTICS OF THE REACTOR COOLANT SYSTEM FOR ANY MODE OF OPERATION PER SYSTEM DESCRIPTION 36 AND USAR SECTIONS 4.0 AND 6.5.

0360000001K35 - DISCUSS THE DESIGN CHARACTERISTICS OF THE PRESSURIZER PRESSURE CONTROL SYSTEM FOR ANY MODE OF OPERATION PER SYSTEM DESCRIPTION 36 AND USAR SECTIONS 7.2 AND 7.3.

19 . d			RO Value	SRO Value			
KA	003	K1.10	3.0	3.2	Ref. Section	Ref. Page	Ref. Revision
Refs.	Accident & Transient Analysis				VI.5	VI-18 - 20	1987
	PLANT SPECIFIC BACKGROUND INFORMATION FOR KNPP IPEOP ES-0.1			ES-0.1	4. Step 9	22-23	I

Objs. 2030000001K06 - Describe the initial trends of various primary and secondary parameters following a DRCF event.

E000040501K06 - DISCUSS THE CONDITIONS REQUIRED TO SUPPORT OR INDICATE NATURAL CIRCULATION, GIVEN A REACTOR TRIP RECOVERY, IN ACCORDANCE WITH ES-0.1.

20 . d

			RO Value	SRO Value			
KA	004	A3.01	3.5	3.7	Ref. Section	Ref. Page	Ref. Revision
Refs.	Control Room Alarm Response Procedures			CVC-35	47044L		A
	Integrated Logics - CVCS			E-2024			
	Integrated Logics - CVCS			E-2023			X

Objs. 0350260101A01 - PERFORM AN AUTOMATIC MAKEUP, WHILE CONTROLLING RCS BORON CONCENTRATION IN ACCORDANCE WITH N-CVC-35A.

0350000001K02 - EXPLAIN THE RESPONSE TO THE OPERATION OF ALL CHEMICAL AND VOLUME CONTROL - CHARGING AND LETDOWN SYSTEM CONTROLS IN ACCORDANCE WITH LOGIC DIAGRAMS E2020, E2023 THROUGH E2028, E2030, E3000, AND E2039.

21 . d

			RO Value	SRO Value			
KA	004	K6.24	2.5	2.6	Ref. Section	Ref. Page	Ref. Revision
Refs.	System Description - Chemical and Volume Control (CVC)			Number 35	3.2.8	35-16, 17	Orig
	Integrated Logic - CVCS			E-2023			X
	Flow Diagram - CVCS			OP XK100-36			AS

Objs. 0350000001K06 - DISCUSS THE DESIGN CHARACTERISTICS OF THE CVC SYSTEM FOR ANY MODE OF OPERATION PER SYSTEM DESCRIPTION 35 AND USAR SECTIONS 9.2, 10A.5, AND 14.1.

0350000001K02 - EXPLAIN THE RESPONSE TO THE OPERATION OF ALL CHEMICAL AND VOLUME CONTROL - CHARGING AND LETDOWN SYSTEM CONTROLS IN ACCORDANCE WITH LOGIC DIAGRAMS E2020, E2023 THROUGH E2028, E2030, E3000, AND E2039.

22 . a

			RO Value	SRO Value			
KA	005	K2.01	3.0	3.2	Ref. Section	Ref. Page	Ref. Revision
Refs.	Circuit Diagram - 4160V & 480V Power Sources			E-240			AQ
	RHR Operation At A Reduced Inventory Condition			N-RHR-34C	2.0 & 4.13	1, 6	I

Objs. 0340000004K07 - LIST THE POWER SUPPLIES FOR THE FOLLOWING COMPONENTS PER N-RHR-34-CL:1) RHR PUMPS A & B2) RHR-2A/2B3) RHR-11

0340030104A01 - Align the RHR System for a Reduced Inventory condition in accordance with N-RHR-34C

PRA SYSTEM Importance: Bus 52 or 62 – 1; Bus 5 or 6 – 2

23 . c

			RO Value	SRO Value			
KA	005	K6.03	2.5	2.6	Ref. Section	Ref. Page	Ref. Revision
Refs.	Residual Heat Removal System Operation			N-RHR-34	2.3, 2.12, 4.2	1, 3, 8	AM
	Leakage into Component Cooling System			A-CC-31B	4.8.3	3	I
	Flow Diagram Aux. Coolant			OP XK100-18			AK

Objs. 0340000001K02 - DISCUSS THE DESIGN CHARACTERISTICS OF THE RESIDUAL HEAT REMOVAL SYSTEM, FOR ANY MODE OF OPERATION, PER SYSTEM DESCRIPTION 34 AND USAR SECTION 6.1, 6.2, 6.5, AND 9.3.

0340000004K08 - IDENTIFY THE COMPONENTS THAT AFFECT THE RHR SYSTEM OPERABILITY PER TECHNICAL SPECIFICATIONS SECTIONS 3.1 AND 3.3.

0310010401A01 - Given leakage into the Component Cooling System, identify the source in accordance with A-CC-31B.

24 . c

			RO Value	SRO Value	Ref. Section	Ref. Page	Ref. Revision
KA	006	A3.01	4.0*	3.9			
Refs.	KNPP USAR				6.2.2 Accumulators	6.2-10	15
	KNPP USAR				14.3.2	14.3-8	15
	Integrated Logic - SI			E-2034			R

Objs. 0330000001K02 - DISCUSS THE DESIGN CHARACTERISTICS OF THE SAFETY INJECTION SYSTEM, FOR ANY MODE OF OPERATION, PER SYSTEM DESCRIPTION 33 AND USAR SECTIONS 1.3, 1.5, 1.8, 6.2, 7.2, 7.5, 14.2, 14.3.

0330000001K04 - EXPLAIN THE RESPONSE TO THE OPERATION OF ALL SAFETY INJECTION SYSTEM CONTROLS IN ACCORDANCE WITH LOGIC DIAGRAMS/SCHEMATICS E1635, E2032, E2033, E2034, E2035, XK100-148 AND XK100-150.

Contributor to Core Damage Sequence: Large LOCA – 9%; PRA SYSTEM Importance: SI Accumulator – 5

25 . a

			RO Value	SRO Value	Ref. Section	Ref. Page	Ref. Revision
KA	006	K2.01	3.6	3.9			
Refs.	4160V AC Supply and Distribution System Operation			N-EHV-39	4.2.2	5	N
	System Description - Diesel Generator Electrical (DGE)			Number 42	3.12.3; 3.14.1&2	42-37, 39; 42-44, 46	ORIG
	Integrated Logic - DG Electric			E-1637; E-2000			U; W

Objs. 0330000004K07 - LIST THE POWER SUPPLIES FOR THE FOLLOWING COMPONENTS PER N-SI-33-CL:1) SI PUMPS A & B2) SI-20A/20B

0340000004K07 - LIST THE POWER SUPPLIES FOR THE FOLLOWING COMPONENTS PER N-RHR-34-CL:1) RHR PUMPS A & B2) RHR-2A/2B3) RHR-11

0420000004K04 - IDENTIFY THE MAJOR EQUIPMENT THAT STARTS ON EACH STEP OF THE SI/BO LOADING SEQUENCE IN ACCORDANCE WITH E1637, E1638, E2000, E2001, E2002.

26 . c

			RO Value	SRO Value			
KA	007	A1.03	2.6	2.7	Ref. Section	Ref. Page	Ref. Revision
Refs.	Pressurizer Relief Tank Operation			N-RC-36B	2.0	1, 2	N

Objs. 0360000001K34 - DISCUSS THE DESIGN CHARACTERISTICS OF THE PRESSURIZER AND PRT SYSTEM FOR ANY MODE OF OPERATION PER SYSTEM DESCRIPTION 36 AND USAR SECTION 4.2.

0360230101A01 - WHEN DIRECTED, ESTABLISH NORMAL OPERATING CONDITIONS IN THE PRT IN ACCORDANCE WITH N-RC-36B.

27 . b

			RO Value	SRO Value			
KA	007	K3.01	3.3	3.6	Ref. Section	Ref. Page	Ref. Revision
Refs.	System Description -Reactor Coolant (RC)			Number 36	3.6.8	36-30	A
	Flow Diagram - RCS			OP XK100-10			BE

Objs. 0360000001K34 - DISCUSS THE DESIGN CHARACTERISTICS OF THE PRESSURIZER AND PRT SYSTEM FOR ANY MODE OF OPERATION PER SYSTEM DESCRIPTION 36 AND USAR SECTION 4.2.

Contributor to Core Damage Sequence: Medium LOCA – 13%

28 . b

			RO Value	SRO Value			
KA	010	K5.02	2.6	3.0*	Ref. Section	Ref. Page	Ref. Revision
Refs.	Steam Tables						

Objs. 0360000001K34 - DISCUSS THE DESIGN CHARACTERISTICS OF THE PRESSURIZER AND PRT SYSTEM FOR ANY MODE OF OPERATION PER SYSTEM DESCRIPTION 36 AND USAR SECTION 4.2.

DELETED DAP

29 . d

			RO Value	SRO Value			
KA	012	K1.05	3.8*	3.9	Ref. Section	Ref. Page	Ref. Revision
Refs.	Logic Diagram - Reactor Trip Signals			Xk100-144			5C
	Integrated Logic - DG Electric			E-1635			O

Objs. 0470000001K02 - Explain the response to the operation of all Reactor Control & Protection system controls in accordance with Logic Diagrams/Schematics E2042, E2043, E2044, E2051-1, E2051-2, XK100-144...

0490000004K03 - Explain the basic principles of operation for the following major components of the CRD System per System Description 49:1) Rod Drive MG Sets2) Reactor Trip and Bypass Breakers

30 . a

			RO Value	SRO Value			
KA	012	K6.01	2.8	3.3	Ref. Section	Ref. Page	Ref. Revision
Refs.	Bistable Tripping for Failed Reactor Protection or Safeguards Inst.			A-MI-87	Attachment I PT-449	50	M
	Instrument Block Diagram - PRZR Press						XK100-546 2R
	Instrument Block Diagram - dT-Tavg			XK100-551			1N

Objs. 0470000001K06 - Discuss the purposes/functions of the Reactor Control & Protection system and it's components for any mode of operation per System Description 47 and USAR sections 1.3, 7.2.

0470000001K02 - Explain the response to the operation of all Reactor Control & Protection system controls in accordance with Logic Diagrams/Schematics E2042, E2043, E2044, E2051-1, E2051-2, XK100-144,..., XK100-546 through XK100-559

DELETED OKP

31 . C

			RO Value	SRO Value			
KA	013	A1.04	3.4	3.6	Ref. Section	Ref. Page	Ref. Revision
Refs.	Integrated Logic Diagram Auxiliary Feedwater System			E-1602			AW

Objs. 05B0000001K02 - Explain the response to the operation of All Auxiliary Feedwater System Controls in accordance with logic diagrams/schematics: E1602, E2802, XK100-149, XK100-157.

Contributor to Core Damage Sequence: Loss of AC Bus – 2%; PRA SYSTEM Importance: TD AFW – 9

32 . C

			RO Value	SRO Value			
KA	013	A4.01	4.5	4.8	Ref. Section	Ref. Page	Ref. Revision
Refs.	Response to High Containment Pressure			FR-Z.1	Step 3	3-4	
	Integrated Logic Diagram ICS Sys			E-1604			U
	Integrated Logic Diagram ICS System			E-2012			K

Objs. 0230000004K06 - DESCRIBE THE ICS SYSTEM'S LOCAL COMPONENT OPERATION USING THE FOLLOWING LOGIC DIAGRAMS: E1604 AND E2012.

0550000001K03 - EXPLAIN THE RESPONSE TO THE OPERATION OF ALL ENGINEERED SAFETY FEATURES SYSTEM CONTROLS IN ACCORDANCE WITH LOGIC DIAGRAMS/SCHEMATICS

FRZ0010501K04 - GIVEN A HIGH CONTAINMENT PRESSURE CONDITION, EXPLAIN THE BASIS FOR ACTIONS TAKEN, PER FR-Z.1 BACKGROUND DOCUMENT.

33 . c

			RO Value	SRO Value			
KA	015	K2.01	3.3	3.7	Ref. Section	Ref. Page	Ref. Revision
Refs.	Control Room Alarm Response Procedures			NI-48	47031-L		A
	Circuit Diagram - DC Aux & Emergency AC			E233			AP

Objs. 0480000001K02 - DISCUSS THE DESIGN CHARACTERISTICS OF THE NUCLEAR INSTRUMENTATION SYSTEM, FOR ANY MODE OF OPERATION, PER SYSTEM DESCRIPTION 048 AND USAR SECTIONS 7.2 AND 7.4.

0480110401A01 - GIVEN A POWER RANGE OVERPOWER ROD STOP ANNUNCIATOR, RESPOND IN ACCORDANCE WITH ALARM RESPONSE PROCEDURE 47031-L.

34 . b

			RO Value	SRO Value			
KA	015	K5.15	3.3	3.7	Ref. Section	Ref. Page	Ref. Revision
Refs.	Kewaunee Core Control Theory				Chapter 3 E & Fig 3-10	3-15	

Objs. 05A0020401A01 - GIVEN A FEEDWATER PUMP TRIP, RESPOND IN ACCORDANCE WITH A-FW-05A.

35 . a

			RO Value	SRO Value	Ref. Section	Ref. Page	Ref. Revision
KA	017	K4.01	3.4	3.7			
Refs.	Abnormal Inadequate Core Cooling Monitoring (ICCM) System			A-II-50	4.1	2	D
	System Description - Incore Instr. & Inadequate Core Cooling Monitor Loose Parts Monitoring (IE)			Number 50	3.5.9	50-29, 30	A

Objs. 0500000001K04 - EXPLAIN THE RESPONSE TO THE OPERATION OF ALL INCORE INSTRUMENTATION, ICCMS AND LOOSE PARTS MONITORING SYSTEM CONTROLS IN ACCORDANCE WITH LOGIC DIAGRAMS/SCHEMATICS: E3416, E3417 AND THE SYSTEM DESCRIPTION BLOCK DIAGRAMS.

0500010401A01 - RESPOND TO ABNORMAL ICCMS OPERATION, GIVEN RVLIS LEDs, SMM, OR CET DISPLAY FLASHING IN ACCORDANCE WITH A-II-50.

36 . b

			RO Value	SRO Value	Ref. Section	Ref. Page	Ref. Revision
KA	022	A4.04	3.1*	3.2			
Refs.	System Description - Reactor Building Vent. & Post LOCA Hydrogen Control (RBV)			Number 18	3.6	18-27	Orig
	KNPP USAR				9.6.2	9.6.1, 3	15
	Integrated Logic - Shroud Cooling			E3174			E

Objs. 0180000001K02 - DISCUSS THE DESIGN CHARACTERISTICS OF THE REACTOR BUILDING VENTILATION SYSTEM, FOR ANY MODE OF OPERATION, PER SYSTEM DESCRIPTION 18 AND USAR SECTIONS 5.3, 5.4, 6.3, AND 14.3.9.

0180000001K04 - EXPLAIN THE RESPONSE TO THE OPERATION OF ALL REACTOR BUILDING VENTILATION SYSTEM CONTROLS IN ACCORDANCE WITH LOGIC DIAGRAMS/SCHEMATICS E1608, E1609, E3104, E3174 AND E3310.

37 . a

			RO Value	SRO Value			
KA	022	2.1.32	3.4	3.8	Ref. Section	Ref. Page	Ref. Revision
Refs.	Abnormal Reactor Building Ventilation System Operation			A-RBV-18	4.1.4	2-3	N
	Flow Diagram - Reactor & Shield Bldg. Ventilation			OP M-602			AW

Objs. 0180000001K02 - DISCUSS THE DESIGN CHARACTERISTICS OF THE REACTOR BUILDING VENTILATION SYSTEM, FOR ANY MODE OF OPERATION, PER SYSTEM DESCRIPTION 18 AND USAR SECTIONS 5.3, 5.4, 6.3, AND 14.3.9.

0180010401A01 - RESPOND TO ABNORMAL RBV SYSTEM OPERATION, GIVEN CFCU EMERGENCY DISCHARGE DAMPER(S) OPEN IN ACCORDANCE WITH A-RBV-18.

38 . c

			RO Value	SRO Value			
KA	026	A1.05	3.1	3.4	Ref. Section	Ref. Page	Ref. Revision
Refs.	Technical Specifications for Kewaunee Nuclear Plant				Basis (3.3)	TS B3.3-3	Amend 143
	System Description - Internal Containment Spray (ICS)			Number 23	3.7	23-10	B

Objs. 0230000004K02 - IDENTIFY AND EXPLAIN THE PURPOSE OF THE MAJOR COMPONENTS OF THE ICS SYSTEM PER USAR SECTIONS 6.3 AND 6.4 AND SYSTEM DESCRIPTION 23.

Contributor to Core Damage Sequence: Large LOCA – 9%

39 . b			RO Value	SRO Value			
KA	026	2.1.23	3.9	4.0	Ref. Section	Ref. Page	Ref. Revision
Refs.	Loss Of Reactor Coolant Or Secondary Coolant			E-1	Step 13	8	M

Objs. 0230000001K02 - DISCUSS THE DESIGN CHARACTERISTICS OF THE CONTAINMENT SPRAY SYSTEM, FOR ANY MODE OF OPERATION, PER SYSTEM DESCRIPTION 23 AND USAR SECTIONS 6.4 AND 14.3.

E010010501K03 - Explain the applicability and entry conditions of E-1, Loss of Reactor or Secondary Coolant.

40 . a			RO Value	SRO Value			
KA	028	A2.02	3.5	3.9	Ref. Section	Ref. Page	Ref. Revision
Refs.	POST-LOCA Hydrogen Control			N-RBV-18C	2.4	1	K

Objs. 0180000001K09 - DISCUSS THE DESIGN CHARACTERISTICS OF THE POST LOCA HYDROGEN CONTROL SYSTEM FOR ANY MODE OF OPERATION PER SYSTEM DESCRIPTION 18 AND USAR SECTION 14.3.9.

0180040104A01 - WHEN DIRECTED, STARTUP THE POST-LOCA HYDROGEN CONTROL SYSTEM TO VENT AND FILTER THE CONTAINMENT ATMOSPHERE THROUGH SHIELD BUILDING VENT IN ACCORDANCE WITH N-RBV-18C.

Contributor to Core Damage Sequence: Medium LOCA – 13%

41 . a

			RO Value	SRO Value			
KA	028	K1.01	2.5*	2.5	Ref. Section	Ref. Page	Ref. Revision
Refs.	POST-LOCA Hydrogen Control			N-RBV-18C	CAUTION step 4.1.6.4	8	K
	KNPP USAR				5.5.1	5.5.2	9

Objs. 0240000001K01- Discuss the design characteristics of the Shield Building Ventilation System, for any mode of operation, per System Description 24 and USAR Section 5.5.

0180040104A01 - WHEN DIRECTED, STARTUP THE POST-LOCA HYDROGEN CONTROL SYSTEM TO VENT AND FILTER THE CONTAINMENT ATMOSPHERE THROUGH SHIELD BUILDING VENT IN ACCORDANCE WITH N-RBV-18C.

Contributor to Core Damage Sequence: Medium LOCA – 13%

42 . a

			RO Value	SRO Value			
KA	029	A4.01	2.5	2.5	Ref. Section	Ref. Page	Ref. Revision
Refs.	Reactor Bldg Vent System Cold Operation and Making Releases			N-RBV-18B	NOTE 4.1.1.m	3	U
	Gaseous Radwaste Effluents - Batch Release			SP-32B-116	5.4	4	T
	System Description - Reactor Building Vent. & Post LOCA Hydrogen Control (RBV)			Number 18	3.8.2	18-34	Orig

Objs. 0180000004K03 - Explain the basic principles of operation for the RBV System and the major components and equipment per System Description 18.

43 . C

			RO Value	SRO Value			
KA	033	A1.01	2.7	3.3	Ref. Section	Ref. Page	Ref. Revision
Refs.	Abnormal Spent Fuel Pool Cooling and Cleanup System Operation			A-SFP-21	4.2.2	3, 4	N
	Emergency Spent Fuel Pool Cooling and Cleanup System (SFP)			E-SFP-21	2.0	1	N

Objs. 0210000001K02 - Discuss the design characteristics of the Spent Fuel Pool Cooling and Clean Up System, for any mode of operation, per System Description 21 and USAR Section 9.3.

0210020401A02 - GIVEN A SPENT FUEL POOL LEVEL LOW CONDITION, RESPOND IN ACCORDANCE WITH A-SFP-21

44 . C

			RO Value	SRO Value			
KA	035	K5.03	2.8	3.1	Ref. Section	Ref. Page	Ref. Revision
Refs.	Kewaunee Thermodynamics Theory				Chapter 12	12-30, 31	

Objs. 05A0090401A01 - GIVEN A S/G PROGRAM LEVEL DEVIATION ANNUNCIATOR, RESPOND IN ACCORDANCE WITH ALARM RESPONSE PROCEDURE 47062-A(D).

45 . c

			RO Value	SRO Value	Ref. Section	Ref. Page	Ref. Revision
KA	039	K1.07	3.4*	3.4*			
Refs.	Auxiliary Feedwater System			N-FW-05B	2.1	1	AB
	Technical Specifications for Kewaunee Nuclear Plant				3.4.b.1 & 6	TS 3.4-2, 3	Amend 123

Objs. 0060000004K04 - DESCRIBE THE SYSTEM ARRANGEMENT AND FLOW PATHS WHICH ENABLE THE MS SYSTEM TO FULFILL ITS FUNCTION PER M-203.

05B0000004K08 - IDENTIFY THE COMPONENTS THAT AFFECT THE AFW SYSTEM OPERABILITY PER TECHNICAL SPECIFICATIONS SECTION 3.4.

PRA SYSTEM Importance: TD AFW – 9

46 . b

			RO Value	SRO Value	Ref. Section	Ref. Page	Ref. Revision
KA	039	K3.06	2.8*	3.1			
Refs.	System Description - Main Steam & Steam Dump (MS)			Number 6	3.11	6-22, 23	B
	Integrated Logic - Main Steam & Steam Dump			E-1626			U

Objs. 0060000001K15 - EXPLAIN THE RESPONSE TO THE OPERATION OF ALL STEAM DUMP SYSTEM CONTROLS IN ACCORDANCE WITH LOGIC DIAGRAM/SCHEMATIC E1626 AND XK100-153.

47 . d

			RO Value	SRO Value	Ref. Section	Ref. Page	Ref. Revision
KA	041	K5.07	3.1*	3.6			
Refs.	KNPP USAR				14.1.7	14.1-20	15
	Accident and Transient Analysis				IV.5	IV-17, 18	

Objs. 0060010401A02 - GIVEN A RCS LOOP LO-LO TAVG CONDITION, RESPOND IN ACCORDANCE WITH A-MS-06.

48 . c

			RO Value	SRO Value			
KA	056	2.4.50	3.3	3.3	Ref. Section	Ref. Page	Ref. Revision
Refs.	Control Room Alarm Response Procedures			CD-03	47063-P		A
	Control Room Alarm Response Procedures			FW-05A	47061-G		Orig
	Condensate System Abnormal Operation			A-CD-03	4.4	4-5	M

Objs. 0030000001K04 - Explain the response to the operation of all Main Condensate System Controls in accordance with logic diagrams/schematics E1615

0030070401A01 - GIVEN A FEEDWATER BYPASS ALERT ANNUNCIATOR, RESPOND IN ACCORDANCE WITH ALARM RESPONSE PROCEDURE 47063-P.

49 . d

			RO Value	SRO Value			
KA	059	A2.06	2.7*	2.9*	Ref. Section	Ref. Page	Ref. Revision
Refs.	Abnormal Heater & MS Drain System & Bleed Steam System			A-HD-11	4.2	4	H
	Kewaunee Thermodynamics Theory				Increasing Thermodynamic Efficiency...	8-51, 52	
	Shift Instrument Channel Checks - Operating				SP 87-125	2.1	1 AZ

Objs. 2060000001K01 - List the various transients and accidents leading to either positive or negative reactivity insertion and their possible causes.

50 . b

			RO Value	SRO Value	Ref. Section	Ref. Page	Ref. Revision
KA	059	A3.02	2.9	3.1			
Refs.	Instrument Block Diagram - RX Protection & Control (SGWLC)			XK100-554			1V
	Control Room Alarm Response Procedures			FW05	47062-A/D		A
	Control Room Alarm Response Procedures			FW05	47062-B/E		Orig

Objs. 05A0000001K04 - EXPLAIN THE RESPONSE TO THE OPERATION OF ALL MAIN FEEDWATER SYSTEM CONTROLS IN ACCORDANCE WITH LOGIC DIAGRAMS/SCHEMATICS: E1624, E1625, E2006.

05A0080401A01 - GIVEN A S/G BYPASS CV LEVEL DEVIATION ANNUNCIATOR, RESPOND IN ACCORDANCE WITH ALARM RESPONSE PROCEDURE 47062-B(E).

0540110401A01 - GIVEN A P-485 INSTRUMENT FAILURE, RESPOND IN ACCORDANCE WITH A-TB-54.

51 . c

			RO Value	SRO Value	Ref. Section	Ref. Page	Ref. Revision
KA	061	K3.01	4.4	4.6			
Refs.	Flow Diagram - Service Water System			OP M202			CE
	Technical Specifications for Kewaunee Nuclear Plant				3.4.b.1.A & 3.4.b.5	TS 3.4-2, 3	Amend 123

Objs. 05B0000004K08 - IDENTIFY THE COMPONENTS THAT AFFECT THE AFW SYSTEM OPERABILITY PER TECHNICAL SPECIFICATIONS SECTION 3.4.

PRA SYSTEM Importance: Service Water – 3; MD AFW – 7

52 . a

			RO Value	SRO Value			
KA	061	K4.02	4.5	4.6	Ref. Section	Ref. Page	Ref. Revision
Refs.	System Description - Auxiliary Feedwater (AFW)			Number 5B	3.2.3	5B-10, 11	1
	Instrument Block Diagram - RX Protection & Control (SGWLC)			XK100-554			1V
	Integrated Logic - Aux. Feedwater						E-1602 AW

Objs. 05B0000001K01 - DISCUSS THE DESIGN CHARACTERISTICS OF THE AUXILIARY FEEDWATER SYSTEM FOR ANY MODE OF OPERATION PER SYSTEM DESCRIPTION 05B AND USAR SECTIONS 1.2.8.e, 6.6, AND 10A.3.5.

05B0000001K02 - EXPLAIN THE RESPONSE TO THE OPERATION OF ALL AUXILIARY FEEDWATER SYSTEM CONTROLS IN ACCORDANCE WITH LOGIC DIAGRAMS/SCHEMATICS: E1602, E2802, XK100-149, XK100-157.

PRA SYSTEM Importance: MD AFW – 7

53 . a

			RO Value	SRO Value			
KA	062	K4.10	3.1	3.5	Ref. Section	Ref. Page	Ref. Revision
Refs.	Circuit Diagram - DC Aux. & Emergency AC			E-233			AP
	System Description - DC & Emergency AC Electrical Dist (EDC)			Number 38	3.11	38-17	A

Objs. 0380000001K06 - DISCUSS THE DESIGN CHARACTERISTICS OF THE DC AND EMERGENCY AC SUPPLY SYSTEM FOR ANY MODE OF OPERATION PER SYSTEM DESCRIPTION 38 AND USAR SECTIONS 1.8 AND 8.0.

54 . c

			RO Value	SRO Value			
KA	063	K4.01	2.7	3.0*	Ref. Section	Ref. Page	Ref. Revision
Refs.	DC Supply and Distribution System			A-EDC-38	4.7.3	5	U
	Circuit Diagram - DC Aux. & Emergency AC			E-233			AP

Objs. 0380000001K02 - EXPLAIN THE RESPONSE TO THE OPERATION OF ALL DC AND EMERGENCY AC SUPPLY SYSTEM CONTROLS IN ACCORDANCE WITH LOGIC DIAGRAMS E233 AND E3626.

0380060401A01 - GIVEN A BRC-103/BRD-103 FEEDER BREAKER UNDERVOLTAGE CONDITION, RESPOND IN ACCORDANCE WITH A-EDC-38.

PRA SYSTEM Importance: Batteries – 4

55 . b

			RO Value	SRO Value			
KA	064	A1.08	3.1	3.4	Ref. Section	Ref. Page	Ref. Revision
Refs.	Diesel Generator A Manual Operation			N-DGM-10A	2.5, 4.3.1	1, 9	F
	Integrated Logic - Diesel Generator Electric			E-2022			N
	Integrated Logic - Diesel Generator Electric			E-1621			AK

Objs. 0420000001K02 - Explain the response to the operation of all Diesel Generator Electric system controls in accordance with Logic diagrams/schematics E1621, E1622, E1634 THROUGH E1639, E2000, E2001, E2002, E2022 AND E2900.

0420000001K06 - Discuss the design characteristics of the Diesel Generator Electric system for any mode of operation per System Description 42 and USAR sections 1.8, 8 and 14.1.

PRA SYSTEM Importance: Bus 5 or 6 – 2

56 . d

			RO Value	SRO Value			
KA	064	A4.10	3.3	3.4	Ref. Section	Ref. Page	Ref. Revision
Refs.	Loss of Reactor Or Secondary Coolant			E-1	Step 16 RNO a.2)	10	M
	PLANT SPECIFIC BACKGROUND INFORMATION FOR KNPP IPEOP E-1			E-1	4. Step 16	52	J
	System Description - Diesel Generator Electric (DGE)			Number 42	3.1	42-1, 2	Orig

Objs. E000010501K08 - Given a Reactor Trip, explain the basis for actions taken per E-0 Background Document.

0420000004K03 - Explain the basic principles of operation for the DGE System and the major components and equipment per System Description 42.

E010010501K04 - Given a Loss of Reactor or Secondary Coolant, explain the basis for actions taken per E-1 Background Document

57 . b

			RO Value	SRO Value			
KA	068	A4.03	3.9	3.8	Ref. Section	Ref. Page	Ref. Revision
Refs.	CVC Monitor Tanks and Pumps			N-CVC-35M	4.3	2-3	E
	Abnormal Radiation Monitoring System			A-RM-45	3.8	5	AC
	Integrated Logic - CVCS			E-2047			K

Objs. 32A0000001K02 - EXPLAIN THE RESPONSE TO THE OPERATION OF ALL LIQUID WASTE SYSTEM CONTROLS IN ACCORDANCE WITH LOGIC DIAGRAMS E2046 AND E2047.

0450160401K01 - LIST THE IMMEDIATE OPERATOR ACTIONS, GIVEN AN R-18 RADIATION INDICATION HIGH CONDITION IN ACCORDANCE WITH A-RM-45.

32A0000004K06 - DESCRIBE THE LIQUID WASTE PROCESSING SYSTEM'S LOCAL COMPONENT OPERATION USING THE FOLLOWING LOGIC DIAGRAMS: E2046, E2047, AND E2048.

58 . b

			RO Value	SRO Value			
KA	068	K6.10	2.5	2.9	Ref. Section	Ref. Page	Ref. Revision
Refs.	Radiation Monitoring System			N-RM-45	4.3.17	15-16	AJ
	Abnormal Radiation Monitoring System			A-RM-45	3.5	3-4	AC
	Steam Generator Blowdown Treatment System			N-BT-07A	1.1	1	V

Objs. 0070000004K04 - DESCRIBE THE SYSTEM ARRANGEMENT AND FLOW PATHS WHICH ENABLE THE STEAM GENERATOR BLOWDOWN AND BLOWDOWN TREATMENT SYSTEM TO FULFILL ITS FUNCTION M-203, M-368, AND M-436.

0450000001K02 - EXPLAIN THE RESPONSE TO THE OPERATION OF ALL RADIATION MONITORING SYSTEM CONTROLS IN ACCORDANCE WITH THE FOLLOWING LOGIC DIAGRAMS: E2013, E2018, E2019, E2021, E2951, E3745 and E3748.

0450130401K01 - LIST THE IMMEDIATE OPERATOR ACTIONS, GIVEN A RADIATION INDICATION HIGH CONDITION ON R-15 AND R-19, IN ACCORDANCE WITH A-RM-45.

59 . d

			RO Value	SRO Value			
KA	071	K1.04	2.7	2.8	Ref. Section	Ref. Page	Ref. Revision
Refs.	Gaseous Waste Processing and Discharge System			N-GWP-32B	8 CAUTION	4	X
	Abnormal Radiation Monitoring System			A-RM-45	3.4	3	AC

Objs. 32B0000004K04 - DESCRIBE THE SYSTEM ARRANGEMENT AND FLOW PATHS WHICH ENABLE THE WD-(G) SYSTEM TO FULFILL ITS FUNCTION (i.e., USING A ONE-LINE DIAGRAM) PER XK100-132

32B0050104A01 - WHEN DIRECTED, PERFORM A GAS DISCHARGE IN ACCORDANCE WITH N-GWP-32B.

60 . b

			RO Value	SRO Value			
KA	071	K4.04	2.9	3.4	Ref. Section	Ref. Page	Ref. Revision
Refs.	Gaseous Waste Processing and Discharge System			N-GWP-32B	4.2.2	2-3	X
	Integrated Logic - Waste Disposal			E-2049			B

Objs. 32B0000004K03 - Explain the basic principles of operation for the WD-(G) System and the major components and equipment per System Description 32B and the Nash Engineering Co. Technical Manual (100-9001).

32B0000004K06 - DESCRIBE THE WD-(G) SYSTEM'S LOCAL COMPONENT OPERATION USING THE FOLLOWING LOGIC DIAGRAMS: E2048-E2050.

61 . a

			RO Value	SRO Value			
KA	072	A2.02	2.8	2.9	Ref. Section	Ref. Page	Ref. Revision
Refs.	Abnormal Radiation Monitoring System			A-RM-45	3.1.1	1	AC

Objs. 0450010401K01 - LIST THE IMMEDIATE OPERATOR ACTIONS, GIVEN A RADIATION INDICATION HIGH CONDITION ON R-1 THROUGH R-10, IN ACCORDANCE WITH A-RM-45.

62 . c

			RO Value	SRO Value			
KA	078	K4.03	3.1*	3.3*	Ref. Section	Ref. Page	Ref. Revision
Refs.	Control Room Alarm Response Procedures			AS-01	47052-I	1	D

Objs. 0010000001K02 - Discuss the design characteristics of the Station and Instrument Air system, for any mode of operation, per System Description 01.

0010040404A03 - Given an alarm received on an air compressor, respond in accordance with A-AS-01.

63 . d

			RO Value	SRO Value			
KA	086	A2.01	2.9	3.1	Ref. Section	Ref. Page	Ref. Revision
Refs.	Emergency Operating Procedure - Fire			E-FP-08	3.1	1	AC
	Integrated Logic - Fire Protection			E-1619			W
	System Description - Fire Protection (FP)			Number 8	3.3	8-12, 13	A

Objs. 0080000001K02 - DISCUSS THE DESIGN CHARACTERISTICS OF THE FIRE PROTECTION SYSTEM, FOR ANY MODE OF OPERATION, PER SYSTEM DESCRIPTION 08 AND USAR SECTION 7.7-5.

0080000001K04 - EXPLAIN THE RESPONSE TO THE OPERATION OF ALL FIRE PROTECTION SYSTEM CONTROLS IN ACCORDANCE WITH LOGIC DIAGRAM E1619.

0080010501K01- LIST THE IMMEDIATE OPERATOR ACTIONS, GIVEN A FIRE, IN ACCORDANCE WITH E-FP-08.

64 . c

			RO Value	SRO Value			
KA	086	K6.04	2.6	2.9	Ref. Section	Ref. Page	Ref. Revision
Refs.	System Description - Fire Protection (FP)			Number 8	3.6, 3.9.2, 3.9.4, 3.13	8-16,21,25,31	A
	Integrated Logic - Fire Protection			E-1619			W

Objs. 0080000001K02 - DISCUSS THE DESIGN CHARACTERISTICS OF THE FIRE PROTECTION SYSTEM, FOR ANY MODE OF OPERATION, PER SYSTEM DESCRIPTION 08 AND USAR SECTION 7.7-5.

0080000001K04 - EXPLAIN THE RESPONSE TO THE OPERATION OF ALL FIRE PROTECTION SYSTEM CONTROLS IN ACCORDANCE WITH LOGIC DIAGRAM E1619.

65 . b			RO Value	SRO Value			
KA	005	2.2.34	2.8	3.2*	Ref. Section	Ref. Page	Ref. Revision
Refs.	KNPP Reactor Data Manual				2.2.8, 5.1.1.1, & 6.4	1	May 19, 2000
	Control Room Alarm Response Procedures			CRD-49	47033-11	1	A

Objs. 0490010401A01- GIVEN A ROD MISALIGNMENT, RESPOND IN ACCORDANCE WITH A-CRD-49B.

0490000001K02 - DISCUSS THE DESIGN CHARACTERISTICS OF THE ROD CONTROL SYSTEM, FOR ANY MODE OF OPERATION, PER SYSTEM DESCRIPTION 49 AND USAR SECTIONS 1.2, 1.3, 1.8, 3.1, 3.2, 7.2, 7.3, 14.1 AND 14.2.

66 . a			RO Value	SRO Value			
KA	007	EA1.04	3.6	3.7	Ref. Section	Ref. Page	Ref. Revision
Refs.	System Description - Reactor Coolant (RC)			Number 36	3.9.4	36-40	A
	Flow Diagram - RCS			OP XK100-10			BE
	PLANT SPECIFIC BACKGROUND INFORMATION FOR KNPP IPEOP E-1			E-1	2.1	3	J

Objs. 0360000004K03 - Explain the basic principles of operation for the RC System and the major components and equipment per System Description 36.

0470000001K02 - Explain the response to the operation of all Reactor Control & Protection system controls in accordance with Logic Diagrams/Schematics...

Contributor to Core Damage Sequence: Small LOCA – 7%

67 . b

			RO Value	SRO Value	Ref. Section	Ref. Page	Ref. Revision
KA	009	EK2.03	3.0	3.3*			
Refs.	PLANT SPECIFIC BACKGROUND INFORMATION FOR KNPP IPEOP E-1			E-1	2.1.4	13(-15)	J

Objs. 2050000001K04 - Name the four categories of LOCA, giving an approximate size range for each category, and describe RCS pressure response for each.

2050000001K11 - Explain why core uncovery will occur in the case of energy-balance controlled SBLOCAs in the cold leg and not for those in the hot leg.

Contributor to Core Damage Sequence: Medium LOCA – 13%

DELETED

68 . b

			RO Value	SRO Value	Ref. Section	Ref. Page	Ref. Revision
KA	011	EK3.10	3.7	3.9			
Refs.	Response to Pressurized Thermal Shock Condition			FR-P.1	Step 1 (RNO)	2	N
	PLANT SPECIFIC BACKGROUND INFORMATION FOR KNPP IPEOP FR-P.1			FR-P.1	4. Step 1	9B-9C	K

Objs. FRP0010501K04 - GIVEN AN IMMINENT PRESSURIZED THERMAL SHOCK CONDITION, EXPLAIN THE BASIS FOR ACTIONS TAKEN, PER FR-P.1 BACKGROUND DOCUMENT.

FRP0010501K05 - IDENTIFY the conditions which would require a procedural transition, while responding to an IMMINENT PRESSURIZED THERMAL SHOCK CONDITION, IN ACCORDANCE WITH FR-P.1.

Contributor to Core Damage Sequence: Large LOCA – 9%

69 . d

			RO Value	SRO Value			
KA	015	AA2.11	3.4*	3.8*	Ref. Section	Ref. Page	Ref. Revision
Refs.	Response to Inadequate Core Cooling			FR-C.1	Step 17 (& NOTE)	8	L
	PLANT SPECIFIC BACKGROUND INFORMATION FOR KNPP IPEOP FR-C.1			FR-C.1	4 - Step 17 NOTE & 17	35-37	H

Objs. FRC0010501K03 - LIST THE MAJOR STEPS IN THE HIGH LEVEL ACTION SUMMARY FOR FR-C.1, RESPONSE TO AN INADEQUATE CORE COOLING CONDITION, PER THE IPEOP BACKGROUND DOCUMENT.

Contributor to Core Damage Sequence: Medium LOCA – 13%

70 . b

			RO Value	SRO Value			
KA	017	AA2.10	3.7	3.7	Ref. Section	Ref. Page	Ref. Revision
Refs.	Abnormal Reactor Coolant Pump Operation			A-RC-36C	3.7.2.b.	6	M

Objs. 0360130401A01 - Given a loss of Component Cooling water to a RXCP, respond in accordance with A-RC-36C.

0360130401K02 - IDENTIFY THE CONDITIONS WHICH WOULD REQUIRE A REACTOR TRIP, GIVEN A LOSS OF COMPONENT COOLING WATER TO A RXCP IN ACCORDANCE WITH A-RC-36C

71 . c

			RO Value	SRO Value			
KA	017	AK1.04	2.9	3.1*	Ref. Section	Ref. Page	Ref. Revision
Refs.	System Description - Reactor Coolant (RC)			Number 36	3.5.4	36-22	A
	KNPP USAR				4.1.5	4.1-13 & 14	14

Objs. 0360000001K32 - DISCUSS THE DESIGN CHARACTERISTICS OF THE REACTOR COOLANT SYSTEM FOR ANY MODE OF OPERATION PER SYSTEM DESCRIPTION 36 AND USAR SECTIONS 4.0 AND 6.5

72 . a

			RO Value	SRO Value			
KA	022	AA2.04	2.9	3.8	Ref. Section	Ref. Page	Ref. Revision
Refs.	Control Room Alarm Response Procedures			RC-36	47042-F	1	B
	Reactor Coolant System Leak Rate Check			SP 36-082	Data Sheet 2, F	10	Y

Objs. 0350000001K06 - DISCUSS THE DESIGN CHARACTERISTICS OF THE CVC SYSTEM FOR ANY MODE OF OPERATION PER SYSTEM DESCRIPTION 35 AND USAR SECTIONS 9.2, 10A.5, AND 14.1.

0360000001K23 - EXPLAIN THE RESPONSE TO THE OPERATION OF ALL PRESSURIZER LEVEL CONTROL SYSTEM CONTROLS IN ACCORDANCE WITH LOGIC DIAGRAMS: E2023, E2025, E2039, E2041, XK100-154, K100-547, XK100-148.

73 . c

			RO Value	SRO Value			
KA	022	AK1.01	2.8	3.2*	Ref. Section	Ref. Page	Ref. Revision
Refs.	Abnormal Reactor Coolant Pump Operation			A-RC-36C	4.1.3	8-9	M
	PLANT SPECIFIC BACKGROUND INFORMATION FOR KNPP IPEOP ECA-0.1			ECA-0.1	3	3A-4	I

Objs. 0360000001K33 - DISCUSS THE DESIGN CHARACTERISTICS OF THE REACTOR COOLANT PUMP FOR ANY MODE OF OPERATION PER SYSTEM DESCRIPTION 36 AND USAR SECTIONS 4.1 AND 4.2.

0360180401A01 - RESPOND TO ABNORMAL RXCP OPERATION DUE TO A LOSS OF SEAL INJECTION, IN ACCORDANCE WITH A-RC-36C.

E000080501K04 - GIVEN A RECOVERY FROM A LOSS OF ALL AC POWER WITHOUT SI REQUIRED, EXPLAIN THE BASIS FOR ACTIONS TAKEN PER ECA-0.1 BACKGROUND DOCUMENT.

74 . c

			RO Value	SRO Value			
KA	024	AA1.26	3.3	3.3	Ref. Section	Ref. Page	Ref. Revision
Refs.	Emergency Boration			E-CVC-35	4.3.1	3	P
	KNPP Reactor Data Manual				RD 6.7, 6.6, 2.2.6	9-10 (2.2.6)	May 19, 2000 & 11/7/96
	Operator Aid 89-13				Boric acid Tank 1A Level Calibration		5-1-89

Objs. 0350010501A01 - PERFORM AN EMERGENCY BORATION WHILE RESPONDING TO AN UNEXPECTED POSITIVE REACTIVITY CONDITION IN ACCORDANCE WITH E-CVC-35.

75 . c		RO Value	SRO Value			
KA	024 AK3.02	4.2	4.4	Ref. Section	Ref. Page	Ref. Revision
Refs.	Response to Nuclear Power Generation/ATWS		FR-S.1	Step 6.g RNO	5	M
	PLANT SPECIFIC BACKGROUND INFORMATION FOR KNPP IPEOP ECA-0.0		FR-S.1	4. Step 6	40	J

Objs. FRS0020501K04 - GIVEN A NUCLEAR GENERATION/ATWS CONDITION, EXPLAIN THE BASIS FOR ACTIONS TAKEN, PER FR-S.1 BACKGROUND DOCUMENT.

76 . d		RO Value	SRO Value			
KA	025 AK2.02	3.2*	3.2	Ref. Section	Ref. Page	Ref. Revision
Refs.	RHR Operation at a Reduced Inventory Condition		N-RHR-34C	Step 4.8, 4.9, Reference Sheet	5, 14	1

Objs. 0340000001K02 - DISCUSS THE DESIGN CHARACTERISTICS OF THE RESIDUAL HEAT REMOVAL SYSTEM, FOR ANY MODE OF OPERATION, PER SYSTEM DESCRIPTION 34 AND USAR SECTION 6.1, 6.2, 6.5, AND 9.3.

0340010101A01 - GIVEN A REDUCED INVENTORY CONDITION, OPERATE THE RHR SYSTEM IN ACCORDANCE WITH N-RHR-34C.

77 . b

			RO Value	SRO Value			
KA	026	AA1.07	2.9	3.0	Ref. Section	Ref. Page	Ref. Revision
Refs.	LOSS OF ALL AC POWER RECOVERY WITHOUT SI REQUIRED			ECA-0.1	steps 1.b & 3.c	2, 4	L
	PLANT SPECIFIC BACKGROUND INFORMATION FOR KNPP IPEOP ECA-0.0			ECA-0.0 (& ECA-0.1)	4. - step 18 (4. - 59 (9) step 1)		N (I)

Objs. E000080501K04 - GIVEN A RECOVERY FROM A LOSS OF ALL AC POWER WITHOUT SI REQUIRED, EXPLAIN THE BASIS FOR ACTIONS TAKEN PER ECA-0.1 BACKGROUND DOCUMENT.

E000070501K05 - GIVEN A LOSS OF ALL AC POWER, EXPLAIN THE BASIS FOR ACTIONS TAKEN, PER ECA-0.0 BACKGROUND DOCUMENT.

Contributor to Core Damage Sequence: Loss of Offsite Power – 38%, Loss of AC Bus – 2%

78 . d

			RO Value	SRO Value			
KA	029	AA1.15	4.1	3.9	Ref. Section	Ref. Page	Ref. Revision
Refs.	KNPP USAR				14.11	14.1-35	15
	KNPP USAR				6.6.2	6.6-2	14
	Integrate Logic - S/G Trip Signals E2802						L

Objs. 05B0000001K01 - DISCUSS THE DESIGN CHARACTERISTICS OF THE AUXILIARY FEEDWATER SYSTEM FOR ANY MODE OF OPERATION PER SYSTEM DESCRIPTION 05B AND USAR SECTIONS 1.2.8.e, 6.6, AND 10A.3.5.

0470000001K05 - Discuss the design characteristics of the Reactor Control & Protection system for any mode of operation per System Description 47 and USAR sections 1.3, 1.8, 4.1, 4.2, 7.2, 7.5.3, 14.1 and 14.2.

0490000004K07 - LIST THE AUTOMATIC TRIPS FOR THE ROD DRIVE MG SETS PER A-CRD-49.

79 . a			RO Value	SRO Value			
KA	029	AK2.06	2.9*	3.1*	Ref. Section	Ref. Page	Ref. Revision
Refs.	System Description - Reactor Protection & RCS Temperature Instruments (RCP)			Number 47	3.13.4	47-39	Orig
	Logic diagram - Rector Trip Signals				XK100-144		5C

Objs. 0470000001K02 - Explain the response to the operation of all Reactor Control & Protection system controls in accordance with Logic Diagrams/Schematics...

0470000001K05 - Discuss the design characteristics of the Reactor Control & Protection system for any mode of operation per System Description 47 and USAR sections 1.3, 1.8, 4.1, 4.2, 7.2, 7.5.3, 14.1 and 14.2.

80 . c			RO Value	SRO Value			
KA	036	AK3.03	3.7	4.1	Ref. Section	Ref. Page	Ref. Revision
Refs.	Loss of Reactor Cavity Inventory During Fuel Movement			E-FH-53B	3.2.1, 4.2.1.c, 4.3, 4.4	1, 3	C

Objs. 0530020501A01 - GIVEN A LOSS OF REACTOR CAVITY INVENTORY DURING FUEL MOVEMENT, RESPOND IN ACCORDANCE WITH E-FH-53B.

0530020501K01 - LIST THE OPERATOR IMMEDIATE ACTIONS, GIVEN A LOSS OF REACTOR CAVITY INVENTORY DURING FUEL MOVEMENT IN ACCORDANCE WITH E-FH-53B.

81 . b

			RO Value	SRO Value			
KA	038	EK1.02	3.2	3.5	Ref. Section	Ref. Page	Ref. Revision
Refs.	Steam Generator Tube Rupture			E-3	Step 19	14	R
	PLANT SPECIFIC BACKGROUND INFORMATION FOR KNPP IPEOP E-3			E-3	3.1.3	15	N

Objs. 0360000001K32 - DISCUSS THE DESIGN CHARACTERISTICS OF THE REACTOR COOLANT SYSTEM FOR ANY MODE OF OPERATION PER SYSTEM DESCRIPTION 36 AND USAR SECTIONS 4.0 AND 6.5.

0360000004K03 - Explain the basic principles of operation for the RC System and the major components and equipment per System Description 36.

E030010501K04 - GIVEN A STEAM GENERATOR TUBE RUPTURE, EXPLAIN THE BASIS FOR ACTIONS TAKEN, PER E-3 BACKGROUND DOCUMENT

82 . b

			RO Value	SRO Value			
KA	040	AK1.06	3.7	3.8	Ref. Section	Ref. Page	Ref. Revision
Refs.	System Description - Feedwater (FW)			Number 5A	3.8	5A-12 & 13	B
	Instrument Block Diagram			XK-100-554	B-3		1V
	Integrated Logic - Feedwater System			E-2006	5		S

Objs. N030010101A01 - PERFORM A POWER INCREASE FROM 15% TO 100% DURING A PLANT STARTUP IN ACCORDANCE WITH N-0-03.

0060000001K08 - Discuss the design characteristics of the Steam Generator and Steam Generator Water Level Control System, for any mode of operation, per System Description 05A and USAR Sections 4.3.1, 7.2.3, and 7.3.3.

83 . b

			RO Value	SRO Value	Ref. Section	Ref. Page	Ref. Revision
KA	040	AK2.02	2.6*	2.6			
Refs.	Control Room Alarm Response Procedures			MS-06	47062-J		Orig
	Integrated Logic - Main Steam & Steam Dump SYS			E-1627			AF

Objs. 0060000001K02 - Discuss the design characteristics of the main steam and steam dump system, for any mode of operation, per System Description 06 and/or USAR sections...

0060000001K03 - EXPLAIN THE RESPONSE TO THE OPERATION OF ALL MAIN STEAM SYSTEM CONTROLS IN ACCORDANCE WITH LOGIC DIAGRAM/SCHEMATIC E1627

84 . c

			RO Value	SRO Value	Ref. Section	Ref. Page	Ref. Revision
KA	054	2.4.50	3.3	3.3			
Refs.	Abnormal Feedwater System Operation			A-FW-05A	4.4	4	O
	Control Room Alarm Response Procedures			FW-05A	47064-D		Orig

Objs. 05A0000001K04 - EXPLAIN THE RESPONSE TO THE OPERATION OF ALL MAIN FEEDWATER SYSTEM CONTROLS IN ACCORDANCE WITH LOGIC DIAGRAMS/SCHEMATICS: E1624, E1625, E2006.

05A0020401A01 - GIVEN A FEEDWATER PUMP TRIP, RESPOND IN ACCORDANCE WITH A-FW-05A.

05A0120401A01 - GIVEN A S/G LEVEL LOW LOW ANNUNCIATOR, RESPOND IN ACCORDANCE WITH ALARM RESPONSE PROCEDURE 47064-A(D).

85 . d

			RO Value	SRO Value			
KA	055	EA1.07	4.3	4.5	Ref. Section	Ref. Page	Ref. Revision
Refs.	Restoration of Off-site Power			A-SUB-59	4.2.2, 4.3	5-6	B
	4160V AC Supply and Distribution System Operation			N-EHV-39	2.7	1	N
	System Description - Substation Electrical (SUB)			Number 59	1.1, 4.1	59-1-3, 59-54	Orig

Objs. 0590010401A02 - GIVEN A LOSS OF OFF-SITE POWER, RESTORE POWER THROUGH F-84 AND Q-303 IN ACCORDANCE WITH A-SUB-59.

0590010401A03 - GIVEN A LOSS OF OFF-SITE POWER, RESTORE NORMAL PLANT ELECTRICAL LINEUPS IN ACCORDANCE WITH A-SUB-59.

0590000004K04 - Draw a one-line diagram of the Substation System showing transformers, breakers, disconnects, transmission line destination, and operating voltages per System Description 59.

86 . b

			RO Value	SRO Value			
KA	055	2.4.45	3.3	3.6	Ref. Section	Ref. Page	Ref. Revision
Refs.	Control Room Alarm Response Procedures			DGM-10	47091-B		Orig
	Local Alarm Response Procedures			DGM-10	D-1A-8		Orig

Objs. 0100000001K02 - DISCUSS THE DESIGN CHARACTERISTICS OF THE DIESEL GENERATOR MECHANICAL SYSTEM FOR ANY MODE OF OPERATION PER SYSTEM DESCRIPTION 10 AND USAR SECTIONS 1.2.8, 1.8, 8.1, AND 8.2.

0100030401A01 - Respond to a Diesel Gen A(B) MECH Lockout Annunciator

Contributor to Core Damage Sequence: Loss of Offsite Power – 38%; PRA SYSTEM Importance: Diesel – 6

87 . c		RO Value	SRO Value			
KA	056 AA1.01	4.0*	3.8*	Ref. Section	Ref. Page	Ref. Revision
Refs.	Fire In Alternate Fire Zone		E-0-06	18	17	M
	System Description - Main Steam Number 6 & Steam Dump (MS)			3.4	6-10, 11	B

Objs. 0060000001K02 - DISCUSS THE DESIGN CHARACTERISTICS OF THE MAIN STEAM AND STEAM DUMP SYSTEM, FOR ANY MODE OF OPERATION, PER SYSTEM DESCRIPTION 06 AND/OR USAR SECTIONS.

0060000004K03 - Explain the basic principles of operation for the MS System and the major components and equipment per System Description 06.

E060010501A01 - GIVEN A FIRE IN AN ALTERNATE FIRE ZONE, RESPOND IN ACCORDANCE WITH E-0-06.

88 . d		RO Value	SRO Value			
KA	062 AK3.04	3.5	3.7	Ref. Section	Ref. Page	Ref. Revision
Refs.	Flow Diagram - SW		OP M-202			CD
	Integrated Logic - SW System		E-1632			AF
	System Description - Service Water (SW)		Number 2	3.6.5	2-24, 25	1

Objs. 0020000004K03 - Explain the basic principles of operation for the SW System and the major components and equipment per System Description 02.

0020000004K04 - Describe the system arrangement and flow paths which enable the SW System to fulfill its function (i.e., draw a one-line diagram) per M-202.

PRA SYSTEM Importance: Service Water – 3

89 . a

			RO Value	SRO Value	Ref. Section	Ref. Page	Ref. Revision
KA	065	AA1.03	2.9	3.1			
Refs.	Loss of Instrument Air			E-AS-01	Steps 4.23.15 & 16	10	M
	Integrated Logic - ICS System			E-2012			K
	Flow Diagram - ICS System			OP M-207			AK

Objs. 0010010504A02 - Upon restoration of Instrument Air, align affected Systems/Components per E-AS-01.

0230000001K04 - EXPLAIN THE RESPONSE TO THE OPERATION OF ALL INTERNAL CONTAINMENT SPRAY SYSTEM CONTROLS IN ACCORDANCE WITH LOGIC DIAGRAMS E1604 AND E2012.

90 . a

			RO Value	SRO Value	Ref. Section	Ref. Page	Ref. Revision
KA	068	AK2.03	2.9	3.1			
Refs.	Fire in Alternate Fire Zone			E-0-06	Step 21	19	M
	Integrated Logic - CVCS			E-2025			AC
	System Description - Chemical and Volume Control (CVC)			Number 35	3.3.4	35-28, 29	Orig

Objs. 0350000001K02 - EXPLAIN THE RESPONSE TO THE OPERATION OF ALL CHEMICAL AND VOLUME CONTROL - CHARGING AND LETDOWN SYSTEM CONTROLS IN ACCORDANCE WITH LOGIC DIAGRAMS E2020, E2023 THROUGH E2028, E2030, E3000, AND E2039.

0350000001K05 - LOCATE ALL SYSTEM CONTROLS AND INDICATIONS WHILE OPERATING THE CHEMICAL AND VOLUME CONTROL - CHARGING AND LETDOWN SYSTEM FROM THE CONTROL ROOM AND THE DEDICATED SHUTDOWN PANEL.

91 . b

			RO Value	SRO Value	Ref. Section	Ref. Page	Ref. Revision
KA	074	EK3.05	4.2	4.5			
Refs.	Response to Inadequate Core Cooling			FR-C.1	Steps 1 & 2	2	L
	PLANT SPECIFIC BACKGROUND INFORMATION FOR KNPP IPEOP FR-C.1			FR-C.1	2.	2	H

Objs. FRC0010501K01 - EXPLAIN THE PURPOSE OF FR-C.1, RESPONSE TO AN INADEQUATE CORE COOLING CONDITION.

FRC0010501K03 - LIST THE MAJOR STEPS IN THE HIGH LEVEL ACTION SUMMARY FOR FR-C.1, RESPONSE TO AN INADEQUATE CORE COOLING CONDITION, PER THE IPEOP BACKGROUND DOCUMENT.

Contributor to Core Damage Sequence: Medium LOCA – 13%

92 . c

			RO Value	SRO Value	Ref. Section	Ref. Page	Ref. Revision
KA	076	AK2.01	2.6	3.0			
Refs.	System Description -Radiation Monitoring			Number 45	3.4	45-27	Orig
	Integrated Logic - Radiation Monitoring			E2021			T
	Flow Diagram CVCS			OP XK100-36			AS

Objs. 0450000001K03 - DISCUSS THE DESIGN CHARACTERISTICS OF THE RADIATION MONITORING SYSTEM FOR ANY MODE OF OPERATION PER SYSTEM DESCRIPTION 45 AND USAR SECTIONS 1.3.3, 1.6.6, 1.8, 6.5, AND 11.2.3.

93 . c		RO Value	SRO Value			
KA	E02 EK1.1	3.2	3.8	Ref. Section	Ref. Page	Ref. Revision
Refs.	SI Termination		ES-1.1	steps 5-8	3-5	M

Objs. E010020501A01 - GIVEN CONDITIONS TO TERMINATE SAFETY INJECTION, PERFORM THE ACTIONS OF ES-1.1.

E010020501K03 - LIST THE MAJOR STEPS IN THE HIGH LEVEL ACTION SUMMARY FOR ES-1.1, SI TERMINATION, PER THE IPEOP BACKGROUND DOCUMENT.

94 . d		RO Value	SRO Value			
KA	E03 EK2.1	3.6	4.0	Ref. Section	Ref. Page	Ref. Revision
Refs.	System Description - Safety Injection (SI)		Number 33	3.8	33-20	C
	Integrated Logic - SIS		E-2032			V

Objs. 0330000001K04 - EXPLAIN THE RESPONSE TO THE OPERATION OF ALL SAFETY INJECTION SYSTEM CONTROLS IN ACCORDANCE WITH LOGIC DIAGRAMS/SCHEMATICS E1635, E2032, E2033, E2034, E2035, XK100-148 AND XK100-150.

Contributor to Core Damage Sequence: Large LOCA – 9%

95 . c

			RO Value	SRO Value			
KA	E04	EK2.2	3.8	4.0	Ref. Section	Ref. Page	Ref. Revision
Refs.	Flow Diagram CVCS			OP XK100-35			AA
	Integrated Logic - CVCS			E2023	LD-4B		W

Objs. E010050501K03 - Explain the applicability and entry conditions of ECA-1.1, Loss of Emergency Coolant Recirculation.

0350000001K02 - EXPLAIN THE RESPONSE TO THE OPERATION OF ALL CHEMICAL AND VOLUME CONTROL - CHARGING AND LETDOWN SYSTEM CONTROLS IN ACCORDANCE WITH LOGIC DIAGRAMS E2020, E2023 THROUGH E2028, E2030, E3000, AND E2039.

0360000001K32 - DISCUSS THE DESIGN CHARACTERISTICS OF THE REACTOR COOLANT SYSTEM FOR ANY MODE OF OPERATION PER SYSTEM DESCRIPTION 36 AND USAR SECTIONS 4.0 AND 6.5.

96 . d

			RO Value	SRO Value			
KA	E04	EK3.2	3.4	4.0	Ref. Section	Ref. Page	Ref. Revision
Refs.	Loss of Emergency Coolant Recirculation			ECA-1.1	Step 21	10	K
	PLANT SPECIFIC BACKGROUND INFORMATION FOR KNPP IPEOP ECA-1.1			ECA-1.1	3.1.4	5	G

Objs. E010050501K02 - LIST THE MAJOR STEPS IN THE HIGH LEVEL ACTION SUMMARY OF ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION, PER THE IPEOP BACKGROUND DOCUMENT.

E010050501K04 - GIVEN A LOSS OF EMERGENCY COOLANT RECIRCULATION, EXPLAIN THE BASIS FOR ACTIONS TAKEN, PER ECA-1.1 BACKGROUND DOCUMENT.

Contributor to Core Damage Sequence: Medium LOCA – 13%

97 . b

			RO Value	SRO Value			
KA	E08	EK1.2	3.4	4.0	Ref. Section	Ref. Page	Ref. Revision
Refs.	Response To Imminent Pressurized Thermal Shock Condition			FR-P.1	Step 24	11	N
	PLANT SPECIFIC BACKGROUND INFORMATION FOR KNPP IPEOP FR-P.1			FR-P.1	4, Step 24	39-40	K

Objs. FRP0010501A01 - GIVEN AN IMMINENT PRESSURIZED THERMAL SHOCK CONDITION, RESPOND IN ACCORDANCE WITH FR-P.1.

FRP0010501K04 - GIVEN AN IMMINENT PRESSURIZED THERMAL SHOCK CONDITION, EXPLAIN THE BASIS FOR ACTIONS TAKEN, PER FR-P.1 BACKGROUND DOCUMENT.

98 . c

			RO Value	SRO Value			
KA	E11	EK3.2	3.5	4.0	Ref. Section	Ref. Page	Ref. Revision
Refs.	Loss of Emergency Coolant Recirculation			ECA-1.1	Figure ECA-1.1-1	17	K
	USAR				Figure 14.2.4-1 (SI Pumps Flow)		15

Objs. E010050501A01 - GIVEN A LOSS OF EMERGENCY COOLANT RECIRCULATION CONDITION, RESPOND IN ACCORDANCE WITH ECA-1.1.

99 . a

			RO Value	SRO Value			
KA	E16	EK1.3	3.0	3.3	Ref. Section	Ref. Page	Ref. Revision
Refs.	Abnormal Radiation Monitoring System			A-RM-45	3.3 & 3.4	2-3	AC
	System Description - Radiation Monitoring (RM)			Number 45	1.3 Table	45-7	Orig
	Integrated Logics - Process Radiation Monitors			E-3748	Table 1 (8)		B

Objs. 0450000001K02 - EXPLAIN THE RESPONSE TO THE OPERATION OF ALL RADIATION MONITORING SYSTEM CONTROLS IN ACCORDANCE WITH THE FOLLOWING LOGIC DIAGRAMS: E2013, E2018, E2019, E2021, E2951, E3745 and E3748.

0450110401K01 - LIST THE IMMEDIATE OPERATOR ACTIONS, GIVEN AN R-13 RADIATION INDICATION HIGH CONDITION IN ACCORDANCE WITH A-RM-45.

100 . b

			RO Value	SRO Value			
KA	E16	EK3.3	3.0	3.0	Ref. Section	Ref. Page	Ref. Revision
Refs.	Response to High Containment Radiation Level			FR-Z.3	Step 2	2	F
	PLANT SPECIFIC BACKGROUND INFORMATION FOR KNPP IPEOP FR-Z.3			FR-Z.3	4, Step 3	6A	C

Objs. FRZ0030501K01- GIVEN A CONTAINMENT HIGH RADIATION LEVEL CONDITION, EXPLAIN THE BASIS FOR ACTIONS TAKEN PER FR-Z.3 BACKGROUND DOCUMENT.

U.S. Nuclear Regulatory Commission
Site-Specific
Written Examination

Applicant Information

Name: MASTER EXAMINATION	Region: III
Date: 12/11/00	Facility/Unit: Kewaunee
License Level: SRO	Reactor Type: W
Start Time:	Finish Time:

Instructions

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. The passing grade requires a final grade of at least 80.00 percent. Examination papers will be collected four hours after the examination starts.

Applicant Certification

All work done on this examination is my own. I have neither given nor received aid.

_____
Applicant's Signature

Results

Examination Value	_____ Points
Applicant's Score	_____ Points
Applicant's Grade	_____ Percent

Kewaunee NRC SRO Examination Answer Key

1 .b	26 .d	51 .c	76 .b
2 .c	27 .c	52 .b	77 .b
3 .d	28 .c	53 .d	78 .b
4 .d	29 .c	54 .b	79 .b
5 .d	30 .c	55 .a	80 .c
6 .b	31 .b	56 .c	81 .d
7 .c	~~32 .d~~ DELETED DH	57 .d	82 .c
8 .b	33 .a	58 .a	83 .c
9 .a	34 .c	59 .a	84 .c
10 .d	35 .c	60 .b	85 .a
11 .b	36 .b	~~61 .b~~ DELETED DH	86 .b
12 .d	37 .d	62 .b	87 .c
13 .d	38 .c	63 .c	88 .c
14 .d	39 .c	64 .a	89 .a
15 .a	40 .a	65 .c	90 .d
16 .a	41 .a	66 .c	91 .c
17 .b	42 .c	67 .c	92 .d
18 .b	43 .c	68 .d	93 .b
19 .a	44 .b	69 .b	94 .b
20 .d	45 .d	70 .a	95 .c
21 .b	46 .b	71 .a	96 .a
22 .a	47 .c	72 .d	97 .c
23 .d	48 .a	73 .a	98 .b
24 .d	49 .a	74 .a	99 .a
25 .d	50 .a	75 .c	100 .b

Senior Reactor Operator Examination

1 . Given the following conditions:

- The plant is at HOT SHUTDOWN
- A cooldown to COLD SHUTDOWN has been initiated.

What is the difference in staffing requirements for the minimum on-duty shift complement per NAD 3.17, "Shift Operation and Turnover," when COLD SHUTDOWN is achieved?

a. Only ONE Nuclear Control Operator is required and the STA is NOT required.

b. The Control Room Supervisor and the STA are NOT required.

c. The fire response team can be reduced to FOUR persons.

d. Only ONE Nuclear Auxiliary Operator is required.

2 . Given the following conditions:

- The plant has been shut down for 10 days
- RHR Pump A has tripped
- RHR Pump B is running but motor amps and indicated flow are fluctuating
- RCS Tavg is at 140°F
- Refueling level is 10.5%
- S/G levels are 20% narrow range

Which of the following actions would the Control Room Supervisor direct at this time?

a. Open S/G PORVs fully.

b. Reset and start RHR Pump A.

c. Evacuate containment and initiate containment closure.

d. Establish RHR Train A cooling from the Dedicated Shutdown Panel.

3 . Which of the following identifies those CVCS-related components that can be operated from the Dedicated Shutdown Panel?

a. CVC-11/CV-31229 Charging Line Isolation.
CV-301/MV-32056 RWST Supply to Charging Pumps.
CVC-203B/CV-31689 Seal Injection Filter Bypass Valve.

b. CVC-440/MV-32217 Emergency Boration to Charging Pumps.
CVC-207A(B)/CV-31237(31238) RXCP A(B) #1 Seal Leakoff Isolation.
LD-27/CV31096 VCT/Holdup Tank Divert Valve.

c. LD-4A(B,C)/CV-31231(31232, 31233) Letdown Orifice A (B,C) Isolation.
LD-60/MV-32099 RHR to CVCS Letdown Line.
CVC-7/CV-31103 Charging Control Chg Line.

d. CVC-15/CV-31230 PRZR Auxiliary Spray Valve.
CVC-215B/CV-31682 Seal Water Filter Bypass Valve.
CC-302/CV-31100 Letdown Cont Outl Temp Controller.

4 . Given the following conditions:

- The plant is operating at 100% power
- Troubleshooting is being performed on Intermediate Range NIS channel N35
- I&C has energized the P6 bistable for this channel by installing jumpers in the NIS drawer bypassing the channel input to the bistable
- The work is expected to extend over two shifts

What are the minimum requirements for documenting the defeat of this interlock?

a. The equipment status logged in the Shift Supervisor's Log, and a Controlled Jumper entry recorded in the Operation's Controlled Jumper Log.

b. The equipment status logged in the Shift Supervisor's Log, and Hold Cards placed on the N35 drawer and the Train A and Train B SR RESET pushbuttons.

c. A Controlled Jumper entry recorded in the I&C Controlled Jumper Log, and Danger Cards placed on the N35 drawer and the Train A and Train B SR RESET pushbuttons.

d. The equipment status logged in the Shift Supervisor's Log, a Controlled Jumper entry recorded in the I&C Controlled Jumper Log, and Danger Cards placed on the N35 drawer and the Train A and Train B SR RESET pushbuttons.

5 . When Refueling Operations are in progress, what is the requirement for performing the Shutdown Safety Assessment (SSA) checklist?

a. The onshift Refueling SRO conducts ONCE daily.

b. The onshift Refueling SRO conducts TWICE daily.

c. The onshift STA conducts ONCE daily.

d. The onshift STA conducts TWICE daily.

6 . Given the following conditions:

- The plant is in COLD SHUTDOWN
- Maintenance had requested a tagout for SI Pump B
- The tagout included taking the handswitch to PULLOUT and racking out its supply breaker only
- A visual inspection showed that NO work is required
- NO disassembly work was performed on the equipment

What is the minimum requirement for SI pump operability?

Following the tagout removal, SI Pump B can be considered restored to service when...

a. the supply breaker is racked in.

b. the supply breaker is racked in and, the breaker has been tested.

c. the supply breaker is racked in and its handswitch is placed in AUTO.

d. the supply breaker is racked in and "SI Pump B Test" of SP 33-098, "SI PUMP and Valve Test", is completed.

7 . Given the following timeline for plant operation:

- 4/22/00 0900, Entered HOT SHUTDOWN (reactor was tripped from 2% power during shutdown)
- 4/22/00 1100, Entered INTERMEDIATE SHUTDOWN
- 4/23/00 0600, Entered COLD SHUTDOWN
- 4/23/00 2300, Entered REFUELING

Which of the following times would be the earliest time that spent fuel movement in the reactor vessel is allowed?

a. 4/25/00, 1100.

b. 4/26/00, 0600.

c. 4/26/00, 1300.

d. 4/27/00, 0300.

8 . Who has the authority to direct the use of the OVERLOAD BYPASS for the Manipulator Crane when lowering a fuel assembly into the reactor vessel?

a. The Shift Supervisor only.

b. The Refueling SRO in Containment only.

c. The Shift Supervisor or Reactor Engineering.

d. The Refueling SRO in Containment and Reactor Engineering.

9 . Given the following conditions:

- The plant is in REFUELING
- Fuel shuffle is complete
- Preparations are completed to initiate draining of the Reactor Cavity

What is the responsibility of the NCO concerning the Source Range instrumentation?

As a MINIMUM, the operator must verify...

a. at least ONE channel is operating and it is monitored during the draining operation.

b. at least ONE channel is operating and it shall be monitored both in the Control Room and in Containment.

c. TWO channels are operating and the audio count can be monitored in Containment.

d. TWO channels are operating and each shall be monitored during the draining operation.

10 . A point source in containment is reading 500 mRem/hr at a distance of two (2) feet. Two options are available to complete a mandatory work assignment near this radiation source:

Option 1 - ONE operator can perform the assignment in forty (40) minutes working at a distance of three (3) feet from the source

Option 2 - TWO operators, trained in the use of special extension tooling, can perform the assignment in sixty-five (65) minutes at a distance of six (6) feet from the source

Which is the preferred option when considering the total exposure based on the ALARA plan?

a. Option 1, which results in total exposure of 0.222 MAN-REM.

b. Option 1, which results in total exposure of 0.148 MAN-REM.

c. Option 2, which results in total exposure of 0.361 MAN-REM.

d. Option 2, which results in total exposure of 0.120 MAN-REM.

11 . Given the following conditions:

- A LOCA outside containment has occurred 15 minutes ago at 0130
- The Shift Supervisor has declared a SITE EMERGENCY
- The faulted line was manually isolated locally, however the NAO performing the task was injured and CANNOT leave the area on his own
- Initial dose estimates for the area are 90 R/hr primarily due to gamma radiation
- The recovery time using one individual is estimated to take 10 minutes with a maximum time of 15 minutes.

Which of the following describes the conditions concerning a rescue attempt?

a. NO attempted rescue may be made since the exposure will exceed the allowed dose guidelines.

b. A qualified individual selected by the Shift Supervisor may attempt the rescue with the approval of the Shift Supervisor and concurrence of the on-shift HP.

c. Only a volunteer, after being made aware of all risks, can attempt the rescue when authorized by the Shift Supervisor and with the concurrence of the Radiological Protection Director.

d. A qualified individual selected by the Shift Supervisor may attempt the rescue once the authorization of the Vice President - Nuclear is obtained and concurrence given by the Radiological Protection Director.

12 . Given the following conditions:

- Discharge is in progress from Waste Condensate Tanks.
- R-18, Waste Discharge Liquid radiation monitor, fails off-scale high.

Which of the following actions is NOT required prior to reinitiating the release?

Technically qualified members of the Facility Staff must...

a. complete TWO independent verifications of the discharge line valving.

b. perform TWO independent verifications of the release rate calculations.

c. analyze TWO independent samples from the tanks for gamma and tritium.

d. establish TWO independent locations for taking grab samples during the release.

13 . Given the following conditions:

- Steam Generator A level rose to 70% following failure of FW-7A/CV-31027, S/G A Main FW Valve
- The reactor tripped following a turbine trip
- Offsite power was lost to the plant when the generator output breakers opened
- All equipment operated normally

Which of the following describes the response of the NCO following the trip?

The operator should...

a. perform ECA-0.0 "Loss Of All AC Power", immediate actions.

b. immediately reduce Auxiliary Feedwater flow to 200 gpm total.

c. close both MS-1A and MS-1B, S/G A and B Main Steam Isolation Valves.

d. verify plant indications and response for the E-0, "Reactor Trip Or Safety Injection", immediate actions.

14 . Which Safety Function would be addressed FIRST if they were all discovered at the same time?

a. Subcriticality Orange Path.

b. Containment Red Path.

c. Integrity Orange Path.

d. Heat Sink Red Path.

15 . Given the following conditions:

- A LOCA has occurred
- Containment hydrogen is 0.5% and has a rising trend
- Actions are underway to establish Train B Hydrogen Dilution using Instrument Air
- A "new" NAO has been assigned to assist by aligning the valves to establish Instrument Air flow to containment

Where would you direct the NAO to go?

a. Main Steam Penetration Area A.

b. Auxiliary Building Loading Dock.

c. Boric Acid Tank Room.

d. Decon Storage Room.

16 . Given the following conditions:

- A Site Emergency has been declared due to S/G tube rupture with offsite release of radiation
- The TSC, OSF and RAF are manned and activated, but the EOF is NOT activated
- The Shift Supervisor is the acting Emergency Director

Which of the following responsibilities of the Shift Supervisor CANNOT be delegated to another individual?

a. Determining protective action recommendations to give to the governmental agencies.

b. Implementing actions recommended by the Severe Accident Management Team.

c. Directing the search and rescue operations.

d. Deploying environmental monitoring teams.

17 . Given the following conditions:

- A loss of normal feedwater flow has occurred
- The actions of FR-S.1 "Response to Nuclear Power Generation/ATWS" are being performed due to a failure of the plant to trip

Prior to transitioning from FR-S.1 to E-0 "Reactor Trip or Safety Injection", why does the operator have to ensure the turbine has been tripped?

a. To provide a signal to initiate automatic voltage restoration for the electrical buses.

b. To maintain the S/G adequate inventory to serve as a heat sink for cooling the reactor core.

c. To prevent Safety Injection actuation which would result in a longer time to reach the required RCS boron concentration.

d. To ensure adequate shutdown margin by adding negative reactivity via the moderator temperature coefficient feedback.

18 . Given the following conditions:

- The plant is at 35% power
- Circ Water Pump B is out of service for maintenance
- Service Transformer 1-33 is out of service for maintenance
- Bus 61 lockout was received 5 minutes ago and is being investigated
- Battery Charger A is out of service for testing

A loss of power to which of the following will result in an IMMEDIATE reactor trip?

a. Bus 3.

b. Bus 4.

c. Bus 5.

d. Bus 6.

19 . Given the following conditions:

- Reactor power is at 75% and increasing slowly
- PRZR pressure is slowly increasing
- PRZR level is increasing
- Tavg is increasing
- Containment parameters are normal

What event is occurring?

a. Control Bank D continuous withdrawal.

b. A loop Tavg circuit is drifting high.

c. A PORV is leaking to the PRT.

d. Turbine runback in progress.

20 . Given the following conditions:

- The plant is at 60% power steady state
- PRZR Spray Valves PS-1A & PS-1B are currently closed in AUTO
- PRZR Pressure Control Channel Switch is selected to the 2-3 position
- PRZR pressure PT-429 reads 2230 psig and rising
- PRZR pressure PT-430 reads 2235 psig and rising
- PRZR pressure PT-431 has just failed to 2185 psig
- PRZR pressure PT-449 reads 2235 psig and rising

Assuming NO operator action is taken, what is the Pressurizer Pressure Control System response to these conditions?

a. PRZR spray Valves PS-1A/B will fully open and continue to depressurize the PRZR until saturation conditions exist in the RCS.

b. PRZR pressure will oscillate between 2210 psig and 2250 psig by the cycling of the proportional and backup heaters.

c. PRZR pressure will stabilize below 2310 psig by the operation of the PRZR spray valves PS-1A/1B.

d. PRZR pressure will oscillate between 2315 and 2335 psig by the cycling of PORV PR-2B.

21 . The plant was manually tripped from rated load because of a slow uncontrollable decrease in RCS pressure. A Safety Injection was automatically initiated a short while ago.

Current indications are as follows:

- RCS pressure is 1700 psig slowly decreasing
- Highest CET temperature reads 575°F
- All PRZR heaters are energized
- PRZR level is 30% and decreasing on LT-426, 427, and 428
- PRZR level LT-433 is 20% and decreasing
- Containment pressure is 1.5 psig and increasing
- Annunciator 47031-Q CONTAINMENT SUMP A LEVEL HIGH is in alarm
- PRT level is 74% and stable
- PRT temperature is 100°F and stable

Which failure has caused the above events and indications?

a. A PRZR PORV has failed open.

b. A PRZR heater well has ruptured.

c. A PRZR reference leg has ruptured.

d. A PRZR Spray Valve has failed open.

22 . Given the following conditions:

- The plant is in HOT SHUTDOWN at 547°F
- RXCP B has just been stopped
- The NCO has placed the controller for PS-1B/CV-31111 PRZR Spray Control Loop B in MANUAL and shut the valve

Why did the operator close PS-1B?

a. Prevent spray flow from bypassing the PRZR.

b. Prevent spray bypass flow from affecting RCS pressure.

c. Prevent rapid depressurization during RXCP B coastdown.

d. Minimize differential temperature between the PRZR and the spray line.

23 . Given the following conditions:

- The plant is in HOT SHUTDOWN following a trip from 100% power
- 5 minutes following the trip, both RXCPs tripped

What is the response of core delta-T?

Core delta-T will...

a. remain constant, then drop as natural circulation flow is developed.

b. drop, then return to the no-load delta-T as natural circulation flow is developed.

c. rise, then return to the no-load delta-T as natural circulation flow is developed.

d. rise, then stabilize at the higher delta-T as natural circulation flow is developed.

24 . Given the following conditions:

- RCS boron concentration is currently 900 ppm
- RCS leakage is less than 0.1 gpm
- Both Reactor Makeup Pumps are in AUTO with Reactor Makeup Pump A running
- Boric Acid Transfer Pump A is in AUTO and FAST
- Boric Acid Transfer Pump B is in PULLOUT
- Auto makeup to the VCT has actuated due to low VCT level

What will occur if the Boric Acid Transfer Pump trips when VCT level reaches 20% and NO operator action is taken?

VCT level will...

a. continue to rise to 28% and VCT boron concentration will remain the same.

b. continue to rise to 28% and VCT boron concentration will decrease between 250 and 260 ppm.

c. stabilize at approximately 21% and VCT boron concentration will remain the same.

d. stabilize at approximately 21% and VCT boron concentration will decrease between 20 and 30 ppm.

25 . Given the following conditions:

- The RCS is solid
- RCS temperature is 185°F
- RHR Train A is operating to maintain RCS temperature
- RHR letdown is aligned through the RHR/CVCS spectacle flange
- LD-10/CV-31099 Letdown Cont Pressure is in AUTO set to 325 psig

What will occur if the controller output for LD-10 fails low (0%)?

LD-10 moves fully...

a. closed to raise the upstream letdown pressure as indicated on PI-155, and RCS pressure decreases.

b. open to raise the upstream letdown pressure as indicated on PI-155, and RCS pressure increases.

c. closed to lower the upstream letdown pressure as indicated on PI-155, and RCS pressure increases.

d. open to lower the upstream letdown pressure as indicated on PI-155, and RCS pressure decreases.

26 . Given the following conditions:

- The plant is in REFUELING Mode
- Refueling Level is at 65%

Why is only ONE train of RHR required to be operable?

a. It minimizes the effect of an inadvertent dilution accident.

b. It allows for ease in handling fuel assembly removal and placement in core locations.

c. LTOPS using the RHR system is NO longer required since an adequate vent path exists.

d. The available heat sink allows sufficient time to initiate alternate core cooling if RHR is lost.

27 . Given the following conditions:

- The plant is in INTERMEDIATE SHUTDOWN
- RCS temperature is 275°F
- RHR Pump A is operating with BOTH RHR Heat Exchangers in service for cooldown
- RHR Pump B is out of service for maintenance
- Annunciator 47024-H CC SURGE TANK LEVEL HIGH/LOW is in alarm
- CC Surge Tank level is verified to be increasing
- R-17 Comp Cooling Liquid Monitor indicates increasing radiation levels

Identify the leak location and the correct operator response for these conditions.

The leak is located in the...

a. RHR Pump A Seal Cooler. Isolate flow to the cooler and continue the cooldown to 201°F using RHR Pump A.

b. Seal Water Heat Exchanger. Isolate flow to the heat exchanger and continue the cooldown to COLD SHUTDOWN using both RHR Heat Exchangers.

c. RHR Heat Exchanger A. Isolate flow to the heat exchanger and continue the cooldown to 201°F using RHR Heat Exchanger B only.

d. RHR Heat Exchanger B. Isolate flow to the heat exchanger and continue the cooldown to COLD SHUTDOWN after aligning the Spent Fuel Pool Cooling Heat Exchanger to the RHR system.

28 . Given the following conditions:

- A cooldown to COLD SHUTDOWN is in progress
- RCS Tavg is 450°F
- RCS pressure is 990 psig
- Both SI Accumulator Discharge Isolation power supply breakers are closed
- SI-20A/MV-32091 Accumulator A Isolation valve is closed with its switch in AUTO
- SI-20B/MV-32096 Accumulator B Isolation valve is open with its switch in AUTO

What is the response of the SI Accumulators if a double-ended pipe break occurs on Loop B Cold Leg at the reactor vessel nozzle (Design Basis LOCA)?

a. Neither SI Accumulator will inject to the reactor vessel.

b. Only SI Accumulator B will inject to the reactor vessel.

c. Only SI Accumulator A will inject to the reactor vessel.

d. Both SI Accumulators will inject to the reactor vessel.

29 . Given the following conditions:

- The plant is at 100% power
- At 1200 on 12/01/00 annunciator 47023-I RHR PUMPS CC FLOW LOW alarmed
- Investigation showed CCW flow to RHR Pump A is lost and CCW flow to RHR Pump B is normal
- At 1600, the plant experienced a trip due to a spurious reactor trip signal
- At 1700 all procedural actions required for a plant restart have been completed

Which of the following describes the status for plant startup?

The plant may...

a. NOT be taken critical until normal CCW flow is reestablished to RHR Pump A.

b. be taken critical and power operations continued without restrictions.

c. be taken critical and power operations continued with RHR Pump A to be restored to service by 1200 on 12/04/00.

d. be taken critical and power operations continued with RHR Pump A to be restored to service by 1700 on 12/04/00.

30 . The following PRT parameters are noted:

- Temperature is 125°F
- Level is 72%
- Pressure is 5 psig
- Hydrogen concentration is 1.3%

What action should be taken regarding these conditions?

a. Decrease level to less than 67%.

b. Increase pressure to greater than 8 psig.

c. Decrease temperature to less than 120°F.

d. Increase hydrogen content to greater than 2%.

31 . Given the following conditions:

- Initial PRZR pressure was 2235 psig
- PR-2A/CV-31110 PRZR PORV has popped open
- The operator has just closed PR-1A/MV-32089 PRZR PORV Block Valve
- Current PRZR pressure is 2190 psig
- PRT parameters:
  - Level is 75%
  - Pressure is 6.5 psig
  - Temperature is 123°F

What is the expected temperature indication for TI-438, PORV Outlet temperature, immediately following closure of PR-1A?

a. 650°F.

b. 230°F.

c. 170°F.

d. 125°F.

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32 . Given the following conditions:

- The Plant is at 100% power
- Reactor trip breaker testing is being performed with Reactor Trip Bypass Breaker B (52/BYB) racked in and closed
- Both Reactor Trip Breakers (52/RTA and 52/RTB) are closed
- Reactor Trip Bypass Breaker A (52/BYB) is open and racked out

What is the effect on systems operation if Reactor Trip Bypass Breaker B (52/BYB) failed to open on a reactor trip?

a. Only the turbine control valves and intercept valves will close.

b. The Atmospheric Steam Dump valves receive an open signal but do NOT arm.

c. If RCS Low Tavg occurs, FW-7B/CV-31027 S/G B Main FW Valve does NOT close.

d. Following any SI actuation and reset, automatic actuation of SI Train B CANNOT be blocked.

33 . Given the following conditions:

- The reactor has just tripped
- Prior to the trip, reactor power was at 30% with all systems in their normal lineup
- PRZR pressure channel 4 (PT-449) previously failed low and was removed from service in accordance with A-MI-87 "Bistable Tripping for Failed Reactor Protection or Safeguards Inst."
- Investigation showed a Reactor Protection System bistable failure (actuation) precipitated the reactor trip

Which of the following bistable failures would have caused the reactor trip?

a. Channel 1 Overtemperature Delta-T 405C OVER TEMP TRIP.

b. Channel 2 Turbine Impulse Pressure 486A TURBINE PRESS P13.

c. Channel 3 Overpower Delta-T 407A OVER POWER TRIP.

d. Channel 4 Nuclear Power Range Instrument Drawer N44A OVERPOWER TRIP HIGH RANGE.

34 . Given the following conditions:

- A large break LOCA has occurred
- Containment pressure is observed to be 25 psig
- Containment Spray has NOT initiated
- Manual actuation of Containment Spray has been unsuccessful
- All other ESF actuations and components have functioned normally

What is the proper sequence of actions to be taken to manually initiate Containment Spray for Train A?

Manually start the ICS Pump A...

a. Check ICS-5A/MV-32066 and ICS-6A/MV-32067, Cntmt Spray Pump A Discharge Isolation valves automatically open.
Check ICS-201/CV31272 and ICS-202/CV-31273, ICS Recirculation RWST are closed.
Check CI-1001A/CV-31393 and CI-1001B/CV-31394, Caustic Additive to Cntmt Spray valves automatically open.

b. Manually open ICS-5A/MV-32066 and ICS-6A/MV-32067, Cntmt Spray Pump A Discharge Isolation valves.
Check ICS-201/CV31272 and ICS-202/CV-31273, ICS Recirculation to RWST are closed.
Check CI-1001A/CV-31393 and CI-1001B/CV-31394, Caustic Additive to Cntmt Spray valves automatically open.

c. Manually open ICS-5A/MV-32066 and ICS-6A/MV-32067, Cntmt Spray Pump A Discharge Isolation valves.
Check ICS-201/CV31272 and ICS-202/CV-31273, ICS Recirculation to RWST are closed.
Manually open CI-1001A/CV-31393 and CI-1001B/CV-31394, Caustic Additive to Cntmt Spray valves.

d. Manually open ICS-5A/MV-32066 and ICS-6A/MV-320.67, Cntmt Spray Pump A Discharge Isolation valves.
Manually close ICS-201/CV31272 and ICS-202/CV-31273, ICS Recirculation to RWST.
Manually open CI-1001A/CV-31393 and CI-1001B/CV-31394, Caustic Additive to Cntmt Spray valves.

35 . Given the following conditions:

- The plant is at 100% power
- Control Bank D rods are at 208 steps withdrawn
- An electrical failure deenergizes Instrument Bus BRB-114

If the NCO attempts to withdraw rods, which of the following prevents rod motion?

a. Overpower Delta-T Rod Stop.

b. Overtemperature Delta-T Rod Stop.

c. Power Range Overpower Rod Stop.

d. Intermediate Range High Flux Rod Stop.

36 . If a turbine runback to 60% power occurs from 100% steady-state power, what is the expected response for Power Range Nuclear Instruments (NIs) to Xenon?
(Consider only Xenon effect)

The NIs will drop to 60%,...

a. rise above 60% over the next 8 hours, and then slowly drop to 60%.

b. drop below 60% over the next 8 hours, and then slowly rise to 60%.

c. drop below 60% and stabilize over the next 72 hours at less than 60%.

d. rise above 60% and stabilize over the next 72 hours at greater than 60%.

37 . Which statement correctly defines when a minimum of FOUR core exit thermocouples per quadrant shall be available for readout per Technical Specification 3.11?

a. During all power operation greater than 50% power.

b. Quadrant Power Tilt Ratio is greater than or equal to 1.02.

c. Whenever the movable detector system has less than TWO thimbles per quadrant operable.

d. Reactor power is greater than 85% with ONE excore nuclear power channel out of service.

38 . Given the following conditions:

- A LOCA has occurred
- Containment Spray has actuated
- RWST level currently reads 40%
- Caustic Additive Standpipe level currently reads 100%

What would be the effect of these conditions?

a. Containment radiation levels are higher due to the increased radioactive noble gas production.

b. Containment pressure peaks at higher value due to the reduced heat removal capacity of the ICS spray.

c. Corrosion of components in containment increases due to lower pH value of the containment sump fluid.

d. Removal of hydrogen in the containment atmosphere is lower due to reduce volume of injected sodium hydroxide.

39 . Given the following conditions:

- The plant is in INTERMEDIATE SHUTDOWN 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 CANNOT be restored but heatup to HOT SHUTDOWN can continue.

b. Bus 51 and Bus 61 can be cross-connected only if RCS temperature is reduced to and maintained less than 350°F.

c. Bus 51 and Bus 61 can be cross-connected, heatup can continue to HOT SHUTDOWN, but the reactor CANNOT be taken critical.

d. Bus 51 and Bus 61 can be cross-connected, heatup can continue to HOT SHUTDOWN, and normal power operations commenced.

40 . Given the following conditions:

- A LOCA has occurred
- Containment hydrogen is at 3.4% and increasing

Which of the following describes the appropriate method for reducing the hydrogen concentration per N-RBV-18C "POST LOCA Hydrogen Control"?

a. Use Shield Building Vent System

b. Use the Auxiliary Building Vent System.

c. Use only Containment Purge and Vent System.

d. Use only the Containment 2-inch Vent System.

41 . Given the following conditions:

- A LOCA has occurred
- Venting and filtration of containment atmosphere to control the hydrogen concentration is about to commence

What is the limiting factor in setting the containment atmosphere release flow rate?

The capability of the...

a. Shield Building Ventilation system to maintain a negative pressure in the Shield Building Annulus with respect to the Auxiliary Building.

b. Shield Building Ventilation system to maintain a negative pressure in the Shield Building Annulus with respect to the outside atmosphere.

c. Aux Building Special Ventilation system to maintain a negative pressure in the Auxiliary Building with respect to the outside atmosphere.

d. Aux Building Special Ventilation system to maintain a negative pressure in the Auxiliary Building with respect to the Shield Building Annulus.

42 . Given the following conditions:

- The plant is at 100% power
- Annunciator 47055-N SPENT FUEL POOL ABNORMAL is in alarm
- Spent Fuel Pool level is reported to be decreasing very rapidly

Which of the following would be the appropriate source to use for SFP makeup using the highest possible flow capacity?

a. RWST.

b. Blender.

c. Service Water.

d. Reactor Makeup Water.

43 . Given the following conditions:

- The plant is at 75% power
- Failure of automatic control results in FW-7A, S/G A Main Feedwater valve going closed
- The operator takes manual control and rapidly opens the valve to near its previous position

Which of the following is an immediate result of re-opening the valve?

a. Pressurizer level increases due to increase in RCS Tavg.

b. Rods step out due to the rapid increase in reactor power.

c. S/G level shrinks due to the rapid addition of colder feedwater.

d. S/G level increases rapidly with higher moisture carryover in steam to the turbine.

44 . Given the following conditions:

- The plant is at 8% power
- Main turbine rollup completed at 1800 rpm
- Main Feedwater Pump A is operating and the Main Feed Control Bypass valves are in AUTO
- Steam Dump Control is in Steam Pressure mode with AUTO setpoint at 1005 psig

If main steam line pressure transmitter PT-484 fails high, which of the following will occur?

a. All Steam Dump Valves remain closed.

b. All Condenser Steam Dumps open but reclose when RCS temperature falls to 540°F.

c. All Condenser and all Atmospheric Steam Dump valves open but reclose when the SI signal occurs.

d. All Atmospheric Steam Dump valves open and pressure continues to fall until automatic MSIV closure occurs.

45 . Given the following conditions:

- The plant was originally at 100% power
- Rod Control is selected to MANUAL
- BS-100A/CV-31167, Heater Bleed Steam Supply to FW Heater 15A and 15B valve fails closed

Which of the following describes the initial affect on the given plant parameters, and the actions the operator would take?

	Plant Efficiency	Action
a.	Increases,	NO action required.
b.	Decreases,	Increase power to 100% per plant procedures.
c.	Increases,	Decrease power to 100% power or less per plant procedures.
d.	Decreases,	Decrease power to 100% power or less per plant procedures.

46 . Given the following conditions:

- A plant startup is in progress at 12% reactor power
- Turbine power is at 63 MW (11%)
- Feedwater Control transfer to Main Feedwater Flow Control Valves has been completed and FW-7A and FW-7B controllers are in AUTO maintaining program level
- S/G A and B Bypass Flow Control Valve controllers are set at 35% level setpoint

What is the IMMEDIATE response of the feedwater control system if PT-485, Turbine Impulse Pressure instrument, fails low?

FW-7A/7B throttle to control S/G level at...

a. 33%, and NO annunciators alarm.

b. 33%, and only annunciators 47062-A[47062-D], S/G A[B] PROGRAM LEVEL DEVIATION alarm.

c. 44%, and only annunciators 47062-B[47062-E] S/G A[B] BYPASS CV LEVEL DEVIATION alarm.

d. 44%, and annunciators 47062-B[47062-E] S/G A[B] BYPASS CV LEVEL DEVIATION and 47062-A[47062-D], S/G A[B] PROGRAM LEVEL DEVIATION alarm.

47 . Given the following conditions:

- The plant is at HOT SHUTDOWN 48 hours after a plant trip
- ALL of the below conditions were identified immediately following the Trip and have NOT been corrected

Which of these conditions, if it continued, would require that the plant be cooled to below 350°F due to entry into a Limiting Condition for Operation?

a. AFW-10A, AFW Train A Crossover Valve, is closed.

b. ONE Turbine Overspeed Protection System is inoperable.

c. SW-601A, SW to Aux. Feedwater Pump 1A, is inoperable.

d. 39,000 gallons of water is available in the Condensate Storage Tanks.

48 . Given the following conditions:

- The plant is at 55% power
- S/G level channel LT-473 is removed from service per A-MI-87

If S/G level channel LT-471 fails high, what would be the status of feed for the S/Gs?

a. Both S/Gs are being fed from the motor-driven AFW Pumps.

b. Both S/Gs are being fed from the turbine-driven AFW Pump only.

c. Feed to both S/Gs increases as FW-7A/B, their respective S/G Main Feed valves, throttle open.

d. Feed to S/G B lowers due to throttling close of FW-7B, S/G B Main Feed valve. Feed to S/G A remains normal.

49 . Given the following conditions:

- The plant is at 100% power.
- A major electrical transient has resulted in the following equipment being declared inoperable:

  - Diesel Generator A
  - Safety Injection Pump A
  - Residual Heat Removal Pump B

What is the maximum time the plant has to reach HOT SHUTDOWN as allowed in Technical Specifications?

a. 12 hours.

b. 30 hours.

c. 84 hours.

d. 168 hours (7days).

50 . Which of the following describes the electrical power sources to the Instrument Bus Inverters in order of priority?

	Most Preferred	Next Preferred	Least Preferred
a.	480 VAC,	125 VDC,	120 VAC.
b.	480 VAC,	120 VAC,	125 VDC.
c.	120 VAC,	480 VAC,	125 VDC.
d.	120 VAC,	125 VDC,	480 VAC.

51 . Given the following conditions:

- The plant is in INTERMEDIATE SHUTDOWN with the spare Charger off-site for repair.
- Annunciators 47105A BATTERY A ABNORMAL and 47104-A BATTERY A CHARGER TROUBLE are in alarm
- Both trains of RHR are operable with RHR in service controlling RCS temperature at 300°F
- S/G B is operable
- Battery A Charger is found to have tripped and must be replaced.

What action is required to provide charging capability to both batteries?

a. Install the charger from Battery C or D.

b. Align 250 VDC battery charger BRE-108 to Bus BRA-104.

c. If grounds do NOT exist on either train, cross-connect the 125 VDC Buses BRA-102 and BRB-102.

d. Take the plant to REFUELING shutdown conditions, and if grounds do NOT exist on either train, then cross-connect the 125 VDC Buses BRA-102 and BRB-102.

52 . Given the following conditions:

- Diesel Generator A (DG A) was running paralleled to its associated 4160 VAC Bus per SP-42-312A "Diesel Generator A Available Test"
- The Control Room operator was adjusting load and voltage when the Speed Control switch sticks in the LOWER position

Which of the following describes the FIRST action the operator should take?

a. Take the Control Room Diesel Engine A Control Switch to STOP/PULLOUT.

b. Take the Control Room DG A to Bus 5 supply breaker 1-509 to TRIP/PULLOUT.

c. Direct the local operator to take the DG Excitation and Control Panel Governor switch to RAISE.

d. Direct the local operator to take the DG Excitation and Control Panel (Voltage Control) Mode Selector switch to OFF.

53 . Given the following conditions:

- A LOCA has occurred, followed by a loss of offsite power
- Diesel Generator (DG) A is operating at 60.8 Hz
- The load on DG A is 3025 KW
- Bus 5 voltage is 4250 VAC

Which of the following actions would reduce the KW loading on D/G A by the largest amount?

a. Raising Bus 5 voltage and lowering DG speed.

b. Raising DG speed and stopping Containment Fan Coil Unit C.

c. Lowering Bus 5 voltage and raising DG speed.

d. Lowering Bus 5 voltage and stopping Containment Spray Pump A.

54 . During a radioactive waste release of the CVCS monitor tank to the circulating water system, R-18 Waste Discharge Liquid monitor alarms. The operator notes that the release is terminated.

How was the release terminated?

a. Liquid waste discharge valves WD-18 and WD-19 both auto closed.
b. Liquid waste discharge valve WD-19 only auto closed.
c. CVCS MT discharge valve CVC-918 auto closed.
d. The running CVCS Monitor Tank pump tripped.

55 . Which of the following describes an immediate action of A-RM-45, "Abnormal Radiation Monitoring System Operation", for the failure of R-1, Control Room Area radiation monitor, that results in a high alarm?

a. Direct unnecessary personnel to exit the Control Room.
b. Implement the actions of E-0-06 "Fire In Alternate Fire Zone".
c. Verify closed ACC-4, Control Room A/C Normal Recirc Damper.
d. Verify Control Room Ventilation shifts to the Post Accident Recirc mode of operation.

56 . Why is Air Compressor 1A stopped if cooling water flow to the compressor is lost?

The air compressor will trip...

a. due to overheating of the compressor motor resulting in overload.
b. due to seal leakage resulting in low air discharge pressure.
c. when the limit for air outlet temperature is exceeded.
d. when the limit for oil temperature is exceeded.

57 . Given the following conditions:

- 0100:00 - FP-331/CV-31377 Turbine Lube Oil Storage Tank Deluge valve failed opened (NO fire present)
- 0100:10 - Fire header pressure read 105 psig
- 0100:12 - Fire header pressure read 98 psig
- 0102:00 - FP-330 Manual Isolation for FP-331 was closed
- 0102:05 - Fire header pressure restored to normal

Which of the following describes the steps that must be taken to restore the Fire Pumps to their normal status once FP-331 is reset?

a. Fire Pump A and Fire Pump B control switches must be momentarily taken to STOP/SI RESET.

b. Fire Pump A and Fire Pump B control switches must be momentarily taken to STOP/SI RESET, and Fire Pump A local STOP pushbutton must be depressed.

c. Only Fire Pump A local STOP pushbuttons must be depressed.

d. Both Fire Pump A and Fire Pump B local STOP pushbuttons must be depressed.

58 . Given the following conditions:

- The plant is operating at 18% power
- The high pressure tap to RCS flow instrument FT-411 on loop A fails

What is the resulting plant condition, if NO operator action is taken?

a. All loop A flow indicators will read low, and the reactor trip is generated on RCS loop low flow.

b. All loop A flow indicators will read low, but the reactor trip is generated on low PRZR pressure.

c. Only FI-411 RCS flow indication will read low, and the reactor trip is generated on RCS loop low flow.

d. Only FI-411 RCS flow indication will read low, but the reactor trip is generated on low PRZR pressure.

59 . Given the following conditions:

- Power has stabilized at 60% power following a turbine runback
- During the runback, PORV PR2B/CV-31110 opened
- The associated computer point is in alarm
- Annunciator 47042-B PRESSURIZER PORV DISCHARGE TEMPERATURE HIGH remains actuated
- With the PORV closed, leakage past the PORV has been determined to be 8 gpm
- PR-1B/MV-32089, PRZR PORV Block Valve is currently open
- The previous performance of SP 36-082 "Reactor Coolant Leak Rate Check" (from the previous shift) recorded an identified leak to the RCDT of 2.2 gpm and S/G tube leakage of 0.015 gpm.

Which of the following describes the required actions for this condition?

a. The PORV Block valve must be shut within ONE hour and plant operation may continue.

b. The PORV Block valve must be shut and the plant must be in HOT SHUTDOWN within 12 hours.

c. The PORV Block Valve must be shut with in ONE hour, have its power removed within 72 hours or the plant must be in HOT STANDBY within 6 hours.

d. If annunciator 47042-B is NOT cleared within ONE hour, BOTH PORV Block Valves must be shut and the plant must be in HOT STANDBY within 6 hours.

60 . During a small break LOCA on a cold leg, a phase is reached where the vessel level continues to decrease below the hot leg penetrations and boiling in the core is the means of transporting the core heat to the bubble above the core. A fixed pressure differential exists between the core and the break and is maintained by the loop seal (in the intermediate leg). Natural circulation flow as a heat removal mechanism for the RCS has been lost.

Which of the following describes the main heat removal mechanism for the RCS?

a. Slug flow occurs via the cold legs through the loop seal and flashing occurs across the cold leg break.

b. Steam from the bubble condenses on the hot leg side of the S/G U-tubes which then drains back to the core via the hot legs.

c. Partial natural circulation flow, characterized by liquid pulses, flows from the cold leg over the S/G U-tubes and into the hot legs.

d. Steam from the bubble condenses in the reactor vessel head, which is cooled by fans in the containment, and drains back to the core.

DELETED RXP

61 . Given the following conditions:

- A LOCA has occurred
- RCS pressure is 125 psig
- RCS Core Exit TCs read 380°F
- SI Pump A is running providing 325 gpm flow
- RHR Pump A is running providing 1150 gpm flow

What is the appropriate action taken in response to the above conditions?

Entry into FR-P.1 "Response to Pressurized Thermal Shock Condition" is...

a. NOT required since RCS pressure is below 350 psig.

b. made but NO actions are implemented before returning to procedure in effect.

c. made and a RCS temperature soak for a ONE hour period will be completed.

d. made and cooldown will continue within a limit of 50°F in any 60 minute period.

62 . Given the following conditions:

- The plant is at 100% power
- The following annunciator alarms have actuated:
  - 47013-I RXCP A SEAL LEAKOFF FLOW HIGH/LOW
  - 47015-L RXCP A STANDPIPE HIGH/LOW
- Seal Leakoff Flow Recorder pens indicates one reading low
- RXCP A #1 Seal Outlet and Bearing Water temperatures are stable

If these conditions have existed for 5 minutes without any operator action, what would be the required operator response and reason for this action?

a. The reactor must be tripped and RXCP A stopped since the No. 1 seal has failed.

b. The plant must be shut down and RXCP A stopped as soon as plant conditions permit since the No. 2 seal has failed.

c. MU-1013A/CV31240, RXCP A Standpipe Make-up Isolation, must be manually closed since it has inadvertently opened.

d. The RCDT must be drained to Containment Sump "A" since both RCDT Pumps have tripped and the RCDT has pressurized.

63 . Given the following conditions:

- The reactor is at 9% power during a startup
- Turbine load is 7% (37 MW)
- Steam Dumps are in TAVG Mode
- RXCP A trips
- The reactor is NOT tripped

Which of the following sets of conditions describes the expected condition for the parameters listed below?

	Actual Reactor Power	Steam Flow for S/G A	Steam Flow for S/G B
a.	DECREASE,	DECREASE,	DECREASE.
b.	CONSTANT,	INCREASE,	INCREASE.
c.	CONSTANT,	DECREASE,	INCREASE.
d.	DECREASE,	DECREASE,	CONSTANT.

64 . Given the following conditions:

- The plant is in HOT SHUTDOWN following a reactor trip from 100% power
- All primary and secondary parameters are being maintained at their no-load values
- Charging Pump relief valve lifts causing a loss of charging and seal injection flows

What occurs within the given time frame if the operator fails to take action to stop letdown flow?

a. LD-2/CV-31104 and LD-3/CV-31108, Letdown Isolation valves, will go closed in approximately 3 minutes.

b. Annunciator alarm 47043-C PRESSURIZER LEVEL DEVIATION will actuate in approximately 10 minutes.

c. The operator will be required to place all PRZR heaters in OFF position in approximately 11 minutes.

d. The operator will be required to manually initiate Safety Injection in approximately 16 minutes.

65 . Given the following plant conditions:

- The plant is at 50% power
- Seal injection flow has been lost to the RXCPs
- CVC-207A, RXCP A #1 Seal Leakoff Isolation has been closed due to a high RXCP seal leakoff flow alarm
- CVC-204A RXCP Seal Supply Line Throttle as been closed
- After 15 minutes, the line blockage was cleared and seal injection is ready to be restored to RXCP A
- The operator then reopens CVC-204A to its original position.

What is the result of his actions?

The RXCP...

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

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

c. seal package may be damaged as it undergoes a great than 1°F/minute cool down.

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

66 . Given the following conditions:

- The plant is at HOT SHUTDOWN, 12000 MWD/MTU
- RCS Boron is 940 ppm
- 47042G, AUCTIONEERED TAVG-TREF DEVIATION, has alarmed
- 47043H, RCS LOOP 1A TAVG LO-LO, alarms
- 47043G, RCS LOOP 1B TAVG LO-LO, alarms
- All RCS temperature indications are decreasing due to an uncontrolled cooldown
- Boric Acid Tank A level indication is 56% as indicated on LI-172

Following completion of the emergency boration, what would the tank level read?

a. 53%.

b. 50%.

c. 42%.

d. 28%.

67 . When performing an emergency boration in accordance with FR-S.1, "Response to Nuclear Power Generation/ATWS", the operator is directed to manually start SI Pumps if charging flow is inadequate.

What is the PRIMARY reason Safety Injection is NOT manually initiated?

a. Containment process radiation monitors would be isolated.

b. Instrument Air to containment would be isolated.

c. The Main Feedwater Pumps would be tripped.

d. RXCP seal return flow would be interrupted.

68 . Given the following conditions:

- RCS temperature is 118°F.
- The reactor vessel head is removed.
- Reactor vessel upper internals are installed in the reactor vessel.
- Refueling level is 10.25%
- RCS draining is proceeding at 10 gpm
- RHR Pump A is running with indicated flow of 2000 gpm
- RHR Pump A begins to exhibit indications of cavitation

Which of the following explains why cavitation of the RHR Pump occurred in this situation?

a. Steam binding of the RHR pump occurred due to low recirculation flow.

b. Draining of water from the S/G tubes has increased the RHR suction temperature.

c. Draining with the upper internals in place has reduced the RHR discharge pressure.

d. Air entrainment at the RHR suction inlet has occurred due to the high flow conditions.

69 . Given the following conditions:

- A loss of all AC power has occurred
- After 10 minutes, power was restored to Busses 5 and 6
- The actions of ECA-0.1 are being performed to start a CC Pump

Why are CC-613A and B, RXCP CC Return Manual Isolation Valves verified closed prior to restarting the CC Pump?

a. Reduce CC heat loads to the minimum based on SW loads.

b. Protect CC availability by precluding steam formation in the CC piping.

c. Prevent damage to the RXCP bearings due to excessive cooldown rate.

d. Maximize flow to the CVCS components for reestablishing charging, letdown and seal return.

70 . Given the following conditions:

- The plant is at 100% power
- Rod Control is in MANUAL
- TI-401 Reactor Coolant Loop B Channel 1 Tavg fails high due to failure of TE-15072, B Hot Leg RTD

What is the effect on operations if NO operator action is taken?

a. Operations will continue with PRZR level rising to 50%.

b. Operations will continue with NO change in PRZR level.

c. The reactor will trip when PRZR level rises past the PRZR High Level Reactor Trip setpoint.

d. Technical Specification 3.0.c, Standard Shutdown Sequence, will be entered due to inoperable heaters after PRZR level falls to the cut-out setpoint.

71 . Given the following conditions:

- The plant is at 100% power
- PRZR level transmitter LT-428 has failed to ZERO
- All actions of A-MI-87, "Bistable Tripping for Failed Reactor Protection or Safeguards Instruments", associated with removing LT-428 from service are complete.

- About 5 hours later the following alarms are received:
  - 47042-F PRESSURIZER LVL LETDOWN ISOL & HEATERS OFF
  - 40743-E PRESSURIZER LEVEL DEVIATION
  - 47043-F PRESSURIZER LEVEL LOW
- PRZR level transmitter LT-427 reads ZERO
- PRZR level transmitter LT-426 reads 36% and is rising

What actions are required to be performed by the Operator?

a. Initiate plant shutdown to HOT SHUTDOWN as directed by Technical Specifications.

b. Manually trip the reactor and perform actions of E-0 "Reactor Trip Or Safety Injection".

c. Bypass channel LT-428 and then remove PRZR level channel LT-427 from service per A-MI-87.

d. Position PRZR Level Control Channel Selector to 1-3, restore letdown, and then remove PRZR level channel LT-427 from service per A-MI-87.

72 . Given the following conditions:

- The plant is at 25% power
- The only available Feedwater Pump tripped
- The reactor and turbine failed to trip manually
- S/G narrow range levels are now offscale low
- Control rods are being inserted manually
- Unloading rate is 20%/minute

Which of the following correctly describes of the expected response of the ATWS Mitigation System Actuation Circuitry (AMSAC)?

AMSAC...

a. will NOT trip the turbine nor actuate diverse circuits to trip the reactor trip breakers because reactor power will be below P-10, but will start only the motor-driven AFW Pumps.

b. will NOT trip the turbine nor actuate the circuits to remove excitation from the Rod Drive MG sets because power will be below P-7, but will start all AFW Pumps.

c. will actuate diverse circuits to trip the reactor trip breakers, trip the turbine and start only the motor-driven AFW Pumps.

d. will trip the turbine, actuate circuits to remove excitation from the Rod Drive MG sets, and start all AFW Pumps.

73 . Given the following conditions:

- Reactor power is 100%
- Reactor trip breaker testing is being performed with Reactor Trip Bypass Breaker A (52/BYA) racked in and closed
- Both Reactor Trip Breakers (52/RTA and 52/RTB) are closed
- The NAO racks in and closes Reactor Trip Bypass Breaker B (52/BYB)
- Breakers 52/RTB and 52/BYA open

Which of the following describes the response to this condition?

The reactor is...

a. NOT tripped, and the NCO should manually trip the reactor.

b. NOT tripped as this is the expected response when 52/BYB was closed.

c. tripped and the NCO should direct the NAO to locally open both 52/RTA and 52/BYB.

d. tripped and the NCO should manually trip the reactor as directed by E-0 "Reactor Trip Or Safety Injection".

74 . Given the following conditions:

- Plant startup is in progress
- The reactor is critical with power at 4E-4%
- Source Range channel N31 has a failed discriminator card
- Intermediate Range channel N35 is in TEST
- Power Range channel N44 has a failed high voltage power

Which of the following additional conditions if it were to occur would allow continued plant startup?

a. Source Range channel N32 high voltage power supply fails.

b. Power Range channel N42 summing amplifier fails low.

c. Intermediate Range channel N36 output fails high.

d. Power is lost to Instrument Bus 2 (BRB-113).

75 . The following conditions exist on Unit 3:

- Reactor power is 100%
- A known tube leak of 3.8 GPD exists in S/G B
- The time this value was determined last was 1200
- At 1300 the leak increases to 25 GPD
- At 1400 the leak increases to 45 GPD
- At 1500 the leak increases to 90 GPD
- Secondary radiation monitor response confirms all increases

What action, if any, is required at the current time (1501) as a result of the increasing S/G tube leak as directed by A-RC-36D, "Reactor Coolant Leak", or Technical Specifications?

a. Operation of the plant can continue as normal with increase radiation monitor surveillance per RCC-088 Action Level 1.

b. A normal shutdown to HOT SHUTDOWN must be completed by 1500 the following day per RCC-088 Action Level 2.

c. Load must be reduced below 50% power by 1600 and the plant placed in HOT SHUTDOWN by 1800 per RCC-088 Action Level 3.

d. Cooldown to COLD SHUTDOWN must be completed by within 36 hours since primary to secondary leakage exceeds the Technical Specification allowed limit.

76 . Given the following conditions:

- A S/G tube rupture has occurred in S/G A
- SI is actuated
- The required RCS cooldown has been completed per E-3 "Steam Generator Tube Rupture"
- RCS subcooling is 75°F
- PRZR pressure is 1600 psig
- PRZR level is stable at 20%
- PRZR backup heaters are in PULLOUT
- Charging Pump A is running in MAN
- Charging flow is 52 gpm
- S/G A pressure is 1020 psig
- S/G B pressure is 675 psig

What action should be performed at these conditions to minimize leakage flow from the RCS to the Steam Generator?

a. Terminate SI and stop SI Pumps.

b. Manually throttle open PRZR sprays.

c. Increase feedwater flow to the S/G A.

d. Lower the Steam Dump steam release pressure setpoint

77 . Given the following conditions:

- A plant startup is in progress
- S/G levels are at 33%
- Reactor power is at 8%
- One Main Feedwater Pump is running
- Feedwater Bypass Valves are in AUTO maintaining S/G levels

If a steamline leak results in a 50% step increase in steam flow, what is the FIRST indication the NCO will observe?

a. S/G pressure indications will rise.

b. Feedwater flow indications will lower.

c. S/G narrow range level indications will drop.

d. Feedwater temperature indications will increase rapidly.

78 . Given the following conditions:

- An SI has occurred due to multiple steam leaks
- Train B SI failed to initiate either automatically or manually
- The crew has started the necessary Train B equipment as directed by E-0 "Reactor Trip or Safety Injection"
- Selected plant parameters read:

Instrument Channel -	I	II	III	IV
- RCS Tavg:	541°F	538°F	541°F	539°F
- PRZR pressure:	1834 psig	1830 psig	1832 psig	1828 psig
- Cntmt pressure:	5.2 psig	5.3 psig		5.5 psig
		5.2 psig	60.0 psig	5.3 psig
- S/G A pressure:	800 psig	795 psig	800 psig	
- S/G A steam flow:	0.725×10^6 lbs/hr	0.730×10^6 lbs/hr		
- S/G B pressure:	785 psig		780 psig	770 psig
- S/G B steam flow:			0.751×10^6 lbs/hr	0.745×10^6 lbs/hr

Assuming NO operator action has been taken concerning the MSIVs, which of the following describes their expected positions under these conditions?

	MSIV A	MSIV B
a.	Open	Open.
b.	Open	Closed.
c.	Closed	Open.
d.	Closed	Closed.

79 . Given the following conditions:

- During a plant startup, turbine power has just been raised to 77 MWe
- Reactor power is at 11%
- Circulating Water Pump A was running, but has just tripped on overcurrent
- Circulating Water Pump B is out of service
- Main condenser vacuum indicates 3.3 inches Hg absolute and the backpressure is increasing slowly.

What action should be implemented for these conditions?

a. Load the turbine per N-TB-54, "Turbine Generator Operation".

b. Trip the reactor and enter E-0 "Reactor Trip Or Safety Injection".

c. Place the hogging jet in service per E-AR-09, "Loss of Condenser Vacuum".

d. Restart the tripped Circ Water Pump A per E-CW-04, "Loss of Circulating Water".

80 . Given the following conditions:

- Plant has been operating at 75% power
- The following annunciators go into alarm
  - 47061-D FEEDWATER PUMP B TRIP
  - 47062-A S/G A PROGRAM LEVEL DEVIATION
  - 47062-C S/G A LEVEL LOW
  - 47062-D S/G B PROGRAM LEVEL DEVIATION
  - 47062-F S/G B LEVEL LOW
  - 40763-F FEEDWATER PUMP B ABNORMAL
  - 47064-D S/G B LEVEL LOW LOW
- S/G A levels read 29%, 31% & 31% across the board
- S/G B levels read 0%, 29% & 29% across the board
- Turbine load has stabilized at 60%

Which of the following identifies the action the NCO is required to take?

a. Depress CV LOWER pushbutton until plant load is reduced to less than 50%.

b. Manually trip the reactor and enter E-0 "Reactor Trip Or Safety Injection".

c. Control FW-7B Main Feedwater Control Valve in MAN.

d. Verify BOTH motor-driven AFW pumps have started.

81 . Given the following conditions:

- A loss of offsite power has occurred
- The plant is in HOT SHUTDOWN
- PRZR pressure is at 2000 psig
- Bus 5 and Bus 6 are deenergized due to failures of their respective Diesel Generators
- System Operating has cleared all faulted lines and is ready to restore power to Kewaunee switchyard
- Power will be restored through Q-303 from Point Beach
- NO electrical faults exist on Kewaunee plant equipment

What is the sequence of reenergizing the transformers and repowering the 4160 KV buses?

a. The Tertiary Auxiliary Transformer (TAT) is energized.
Bus 5 is aligned to the TAT.
Bus 6 is aligned to the TAT.
The Main Auxiliary Transformer (MAT) is energized.
Buses 1, 2, 3 and 4 are aligned to the MAT.

b. The Tertiary Auxiliary Transformer (TAT) is energized.
The Main Auxiliary Transformer (MAT) is energized.
Bus 5 is aligned to the TAT.
Bus 6 is aligned to the TAT.
Buses 1, 2, 3 and 4 are aligned to the MAT.

c. The Tertiary Auxiliary Transformer (TAT) is energized.
Bus 5 is aligned to the TAT.
Bus 6 is aligned to the TAT.
The Reserve Auxiliary Transformer (RAT) is energized.
Buses 1, 2, 3 and 4 are aligned to the RAT.

d. The Tertiary Auxiliary Transformer (TAT) is energized.
The Reserve Auxiliary Transformer (RAT) is energized.
Bus 5 is aligned to the TAT.
Bus 6 is aligned to the RAT.
Buses 1, 2, 3 and 4 are aligned to the RAT.

82 . Given the following conditions:

- A fire has occurred in the Control Room
- The actions of E-0-06 "Fire In Alternate Fire Zone" are being performed:
  - The CRS has just started Diesel Generator 1A
  - NCO A has transferred controls to the Dedicated Shutdown Panel (DSP)
  - SD3A/CV-31170 controller at the DSP is set to 0% demand
  - All other local actions are complete
  - A loss of offsite power then occurs

What would NCO A expect to see concerning RCS temperature control following the loss of offsite power?
(Assume the fire has NOT caused any spurious actuations)

a. Both S/G PORVs open to maintain RCS temperature about 549°F.

b. Only S/G PORV SD-3B/CV-31174 opens to maintain RCS temperature about 549°F.

c. Only S/G PORV SD-3A/CV-31170 opens to maintain RCS temperature about 552°F.

d. The first set of Main Steam Safety Valves opens to maintain RCS temperature about 556°F.

83 . Given the following conditions:

- Plant is operating at 100% power
- Electrical systems lineup is normal for at-power operation
- 47102-D INSTUMENT BUS INVERTER TROUBLE has actuated
- SER Point 500 - BRA-111 Instrument Bus Inverter Trouble is indicated
- The NAO dispatched to the inverter reports that the Inverter AC Input Breaker is tripped

Which of the following describes Instrument Bus BRA-113 status?

a. It is deenergized and the Inverter AC source must be restored within 72 hours.

b. It is deenergized and may be manually repowered from 480V AC MCC 52-C via 120/208V AC Panel BRA-105 with plant operations continuing for 7 days.

c. It is powered from 125V DC cabinet BRA-104 via BRA-111 and plant operations may continue indefinitely.

d. It is powered from 480V AC MCC 52-C via 120/208V AC Panel BRA-105 and plant operations may continue indefinitely.

84 . Given the following conditions:

- Plant is operating at 100% power
- Internal Containment Spray (ICS) Pump B was taken out of service 6 hours ago
- The operator has determined a SW header leak exists in containment
- The actions have just been completed for A-SW-02 "Abnormal Service Water System Operation" that have isolated the supply and return headers for Containment Fan Coil Unit C

For the above conditions, which of the following describes the maximum allowed out of service time if power operations is to continue?

a. BOTH ICS Pump B and Fan Coil Unit C must be restored to operable within 66 hours.

b. ICS Pump B may remain out of service indefinitely, but Fan Coil Unit C must be restored to operable within 7 days.

c. ICS Pump B must be restored to operable within 66 hours. Fan Coil Unit C must be restored to operable within 7 days.

d. ICS Pump B must be restored to operable within 66 hours. NO time limit exist for the Fan Coil Unit C as long as Fan Coil Unit D remains operable.

85 . Given the following conditions:

- Plant has been tripped due to a loss of Instrument Air pressure.
- Air pressure has been restored.
- Equipment restoration is in progress.

Following the realignment of Internal Containment Spray (ICS) suction piping, what additional action should be taken?

a. Flush the ICS suction piping.

b. Drain ICS pump discharge piping.

c. Verify flow through recirculation lines.

d. Stroke test CI-1001A and B, Caustic Line Control Valves.

86 . Given the following conditions:

- A LOCA is in progress
- The reactor is tripped
- Core exit thermocouples read 1205°F
- RCS pressure is 900 psig

Assuming each of the following is available, what is the preferred method that provides the most effective means of cooling the core?

a. Starting a Reactor Coolant Pump.

b. Establishing high pressure safety injection flow.

c. Injecting the SI Accumulators by opening the PRZR PORVs to reduce RCS pressure.

d. Injecting the SI Accumulators by dumping steam from the S/Gs to reduce RCS pressure.

87 . What is the detector type and sensing location for R-9, RCS Letdown Radiation monitor?

a. A scintillation detector located near the letdown line upstream of the reactor coolant filter.

b. A scintillation detector located near the letdown line downstream of the reactor coolant filter.

c. A Geiger Mueller detector located near the letdown line upstream of the reactor coolant filter.

d. A Geiger Mueller detector located near the letdown line downstream of the reactor coolant filter.

88 . Given the following conditions:

- A turbine trip resulted in premature opening of one PRZR safety valve and a steamline break inside the reactor containment.
- The operators have performed all the actions of the following
  - E-0 "Reactor Trip Or Safety Injection"
  - E-2 "Faulted Steam Generator Isolation"
  - ES-1.3 "Transfer to Containment Sump Recirculation"
- The faulted S/G is at ZERO psig
- The primary safety reclosed after a period of time
- RCS pressure is 1950 psig and stable
- PRZR level is at 32%
- RCS subcooling is 125°F
- Intact S/G level is 22% with AFW flow controlled by the operator to maintain this level
- Containment pressure rose to 24 psig and is now at 8 psig

When the crew transitions from E-1 "Loss of Reactor Or Secondary Coolant" to ES-1.1 "SI Termination", what actions will be taken in securing the ECCS Pumps?

a. The SI and RHR pumps will be stopped first, charging flow established and then the ICS stopped.

b. The SI and RHR pumps will be stopped, charging flow established but the ICS pumps will continue to run.

c. The SI pumps will be stopped, charging will be established, the ICS pumps and the RHR pumps will continue to run.

d. The SI pumps will be stopped, charging will be established, the ICS pumps will be stopped and then the RHR pumps will be stopped.

89 . Given the following conditions:

- A steamline break has occurred
- Coincident with the steamline break, a RCS leak developed
- RCS pressure is at 375 psig
- RCS wide range temperature is 390°F
- PRZR level is 10%
- ONE Charging Pumps and ONE SI Pump are running
- RHR Pump B is running in SI Injection mode
- Currently the crew is at Step 24 of ES-1.2 "Post LOCA Cooldown And Depressurization"
- The NCO is completing the alignment of Train A RHR for cooldown in accordance with A-RHR-34B "Residual Heat Removal Split-Train Mode"
- When the NCO starts RHR Pump A, PRZR level indication drops offscale and RCS pressure begins to fall

What is the appropriate operation action?

a. Manually start the idle SI Pump as directed by E-1 QRF.

b. Manually initiate Safety Injection and return to Step 1 of E-0.

c. Transition to and perform actions of A-RC-36D "Reactor Coolant Leak".

d. Transition to and perform actions of A-RHR-34 "Abnormal Residual Heat Removal System Operation".

90 . Why is SI-5A/MV-32107, SI Pump A Suction Isolation valve, closed prior to opening RHR-299A/MV-32134, Residual Heat Exchanger Outlet to Safety Injection Pump 1A valve, during the recirculation phase of a LOCA?

a. Reduces system pressure to the SI Pump suction header resulting increased SI flow due to decreased RHR Pump NPSH requirement.

b. Prevents tripping of the SI pumps on high discharge flow rate resulting from increased SI Pump recirculation flow.

c. Protects the SI Pump suction header from overpressurization if the RHR pump was aligned to the RCS hot leg.

d. Prevents the RHR Pump from recirculating contaminated Sump water directly to the RWST.

91 . Given the following conditions:

- An inadvertent Safety Injection signal tripped the reactor
- All equipment responded as required with the following exception: LD-4B/CV-31232, Letdown Orifice B Isolation, failed to close.
- PRZR level fell to 20%; is now at 23% and rising.

If the CVCS letdown line were to break just outside of the containment penetration, which of the following conditions could be expected if the operators failed to respond?

a. Rapid core uncovery with fuel damage would occur.

b. SI injection flow to RCS Loop B Cold Leg would be lost.

c. Recirculation capability from the Containment Sump B would be lost.

d. The charging line penetration to the RCS would undergo thermal shock.

92 . Given the following conditions:

- A LOCA outside containment has occurred
- SI was manually actuated.
- The crew has completed ECA-1.2, "LOCA Outside Containment," and transitioned ECA-1.1, "Loss of Emergency Coolant Recirculation"

Why is subcooling minimized once cooldown has been started?

a. It allows the operator to stop all ECCS pumps.

b. It allows RHR to be placed in service in cooldown mode earlier.

c. This conserves the inventory of the RWST during the injection phase.

d. This lowers the RCS pressure reducing the amount of RCS inventory loss.

93 . Given the following conditions:

- A LOCA has occurred
- At step 17.a of E-1 "Loss Of Reactor Or Secondary Coolant", "Verify recirculation capability", an Orange path for Integrity is noted
- Transition was made to FR-P.1 "Response To Imminent Pressurized Thermal Shock Condition."
- While performing step 2 of FR-P.1, "Check RCS Cold Leg Temperatures - stable or increasing", the following annunciator actuates
  - 47023-B RWST LEVEL LOW

What is the appropriate action?

a. Complete the actions of FR-P.1 and then transition to ES-1.3 "Transfer to Containment Sump Recirculation".

b. Transition directly to ES-1.3 "Transfer to Containment Sump Recirculation", and return to FR-P.1 when actions completed.

c. Immediately return to Step 17 of E-1 and transition to ES-1.3 "Transfer to Containment Sump Recirculation" when directed at step 19.

d. Complete the actions of FR-P.1, return to Step 17 of E-1, and transition to ES-1.3 "Transfer to Containment Sump Recirculation" when directed at step 19.

94 . Given the following conditions:

- A main steam line break occurred inside containment
- The actions of FR-P.1, "Response To Imminent Pressurized Thermal Shock Condition" are being performed
- A RCS temperature soak is required and has been initiated
- NO RXCPs are running

Which evolution would be permitted during the soak period?

a. Energize all PRZR heaters.

b. Place PRZR auxiliary spray in service.

c. Raise the non-faulted S/G level from 18% to 40%.

d. Lower the S/G PORV controller pressure setpoint 25 psi.

95 . Given the following conditions:

- Loss of off-site power has occurred
- Reactor/Turbine Trip has occurred
- Natural circulation conditions have been established
- BOTH CRDM Cooling Fans are inoperable

Which of the following RCS pressure and cold leg wide range temperature relationships are acceptable for a Natural Circulation cooldown?

a. 425°F and 1650 psig.

b. 400°F and 650 psig.

c. 375°F and 360 psig.

d. 180°F and 450 psig.

96 . Given the following conditions:

- A cooldown and depressurization of the RCS is in progress as directed by ES-0.3, "Natural Circulation Cooldown with Steam Void in Vessel"
- A Yellow path is noted for Inventory that directs the crew to FR-I.3, "Response to Voids in Reactor Vessel"
- The CRS continues to direct the actions of ES-0.3

Why was the transition to FR-I.3 NOT made?

a. Upper head steam voiding is expected in these conditions and accounted for in the current procedure.

b. FR-I.3 addresses voids resulting from non-condensable gas evolution, NOT from steam void formation.

c. FR-I.3 would only be entered prior to performing a cooldown and depressurization.

d. All Status Trees are monitored for "information only" in these conditions.

97 . Given the following conditions:

- At 0630 hours a reactor trip with SI occurred
- At 0700 hours the crew enter ECA-1.1, "Loss of Emergency Coolant Recirculation"
- At 0730 hours the following conditions exist:
  - RCS subcooling is 52°F
  - RCS pressure is 2100 psig
  - RCS is on natural circulation
  - SI Pump A is running

How would operators establish the required SI flow to the RCS to meet the requirements of Figure ECA-1.1-1? (For USAR Figure 14.2.4-1, assume the SI pumps provide equal flow at the given pressure and assume 8.3 lbs/gallon.)

a. Throttle the running SI Pump discharge valve.

b. Start the second SI Pump and operate at full flow.

c. Start the second SI Pump and throttle its discharge valve.

d. With the ONE SI Pump running, NO further action is required.

98 . Given the following conditions:

- At 0100 a steamline break occurred with the Unit in HOT SHUTDOWN at normal operating pressure and temperature
- The MSIVs failed to close resulting in depressurization of both S/Gs
- SI automatically actuated
- The SS has declared an Unusual Event and notifications were completed
- At 0110 the MSIVs were closed and the following conditions were noted:
  - RCS wide range temperatures - 440°F
  - RCS pressure - 1850 psig
  - S/G pressure - 200 psig (A); 275 psig (B)
  - Secondary radiation - negligible
  - All ESF equipment functioning as expected

In addition to the notification report based on declaration of emergency classification, identify ALL reports required to be made to the NRC.

a. A 1-hour report per 10 CFR 50.72(b)(1)(iv) for ECCS discharge into the RCS as result of a valid signal.
A 30-day report per 10 CFR 50.73(a)(2)(iv) for ESF actuation.
A 30-day report per 10 CFR 50.73(a)(2)(I)(B) for operation with condition prohibited by plant Technical Specifications.

b. A 1-hour report per 10 CFR 50.72(b)(1)(iv) for .ECCS discharge into the RCS as result of a valid signal.
A 4-hour report per 10 CFR 50.72(b)(2)(ii) for automatic actuation of ESF.
A 30-day report per 10 CFR 50.73(a)(2)(iv) for ESF actuation.
A 30-day report per 10 CFR 50.73(a)(2)(I)(B) for operation with condition prohibited by plant Technical Specifications.

c. A 4-hour report per 10 CFR 50.72(b)(1)(iv) for ECCS discharge into the RCS as result of a valid signal.
A 4-hour report per 10 CFR 50.73(a)(2)(I)(B) for operation with condition prohibited by plant Technical Specifications.
A 4-hour report per 10 CFR 50.72(b)(2)(ii) for automatic actuation of ESF.
A 30-day report per 10 CFR 50.73(a)(2)(iv) for ESF actuation.

d. A 4-hour report per 10 CFR 50.73(a)(2)(I)(B) for operation with condition prohibited by plant Technical Specifications.
A 30-day report per 10 CFR 50.73(a)(2)(iv) for ESF actuation.

99 . Given the following conditions:

- The plant has tripped from 100% power
- Following the trip, 47011B, RADIATION INDICATION HIGH annunciator actuated
- The SER indicates R-13 Aux Building Vent Radiation High

Which of the following describes the plant response?

a. All running Aux Building Supply and Exhaust Fans stop.
SFP Charcoal Filters outlet damper opens and bypass dampers close.
R11/12 sample discharge is routed to containment.
Train A Zone SV Exhaust Fan and Train A Safeguards Fan Coils Units start

b. All running Aux Building Supply and Exhaust Fa.ns stop.
SFP Charcoal Filters outlet damper opens and bypass dampers close.
Train B Zone SV Exhaust fan and Train B safeguards fan coils units start.

c. Aux Building Supply Fan A and Aux Building Exhaust Fan A only stop.
SFP Exhaust Fan A starts.
R11/12 sample discharge is routed to containment.
Train A Zone SV Exhaust Fan and Train A Safeguards Fan Coils Units start.

d. All running Aux Building Supply and Exhaust Fans stop.
Both SFP Exhaust Fan and the SFP Supply Fan start.
Both Trains of Zone SV Exhaust Fans and Safeguards Fan Coils Units start.

100 . Given the following conditions:

- A LOCA has occurred
- The crew is performing cooldown as directed by ES-1.2 " Post LOCA Cooldown And Depressurization"
- ECCS Pumps are still operating in injection phase
- The ICS system has been stopped
- TWO Containment Cooling Fan Coil Units are running
- Containment pressure is stable at 2.2 psig
- The CRS transitions to FR-Z.3 "Response to High Containment Radiation Level" in response to a YELLOW path condition

In addition to running ICS, what action does the CRS direct in FR-Z.3 in order to help reduce containment radiation levels?

a. Initiate Containment Post Accident Vent.

b. The idle Containment Cooling Fan Coil Units are started.

c. A RHR Pump is started and aligned to supply the associated ICS header.

d. ONE train of venting and filtering containment atmosphere through Shield Building Vent is initiated.

Senior Reactor Operator Examination

1 . b		RO Value	SRO Value			
KA	GENERIC 2.1.4	2.3	3.4	Ref. Section	Ref. Page	Ref. Revision
Refs.	Shift Operation and Turnover		NAD 3.17	5.1	2-3	C

Objs. 1190060302K01 - Explain the shift organization during all plant conditions in accordance with NAD 3.17.

2 . c		RO Value	SRO Value			
KA	GENERIC 2.1.14	2.5	3.3	Ref. Section	Ref. Page	Ref. Revision
Refs.	Loss of RHR Cooling While Operating In A Reduced Inventory Condition		A-RHR-34C	3, 4	3	G
	Loss of RHR Cooling While Operating In A Reduced Inventory Condition		A-RHR-34C	Attachment A	14	G

Objs. 0340020401A01 - GIVEN A REDUCED INVENTORY CONDITION, RESPOND TO A LOSS OF RHR IN ACCORDANCE WITH A-RHR-34C.

0340020401K01 - GIVEN A LOSS OF RHR WHILE OPERATING IN A REDUCED INVENTORY CONDITION, EXPLAIN ATTACHMENT A, TIME TO REACH SATURATION VS. TIME SHUTDOWN, PER A-RHR-34C.

3 . d		RO Value	SRO Value			
KA	GENERIC 2.1.30	3.9	3.4	Ref. Section	Ref. Page	Ref. Revision
Refs.	Fire In Alternate Fire Zone		E-0-06	Steps 21, 28, 29 & 42	19-20, 26, 27-28, 34	M
	Charging and Volume Control Prestartup Checklist		N-CVC-35B-CL	Section 4.4	8-10	AG

Objs. 0350000001K05 - LOCATE ALL SYSTEM CONTROLS AND INDICATIONS WHILE OPERATING THE CHEMICAL AND VOLUME CONTROL - CHARGING AND LETDOWN SYSTEM FROM THE CONTROL ROOM AND THE DEDICATED SHUTDOWN PANEL.

0350000001K06 - DISCUSS THE DESIGN CHARACTERISTICS OF THE CVC SYSTEM FOR ANY MODE OF OPERATION PER SYSTEM DESCRIPTION 35 AND USAR SECTIONS 9.2, 10A.5, AND 14.1.

4 . d		RO Value	SRO Value			
KA	GENERIC 2.2.11	2.5	3.4*	Ref. Section	Ref. Page	Ref. Revision
Refs.	Jumper Control Log		NAD 8.14	5.3	2	A
	Shift Operations and Turnover		NAD 3.17	5.3.8.4	8	C
	Tagout Control		NAD 3.3	3.3	2	A

Objs. 1190020302K01 - Explain the use of Controlled Jumpers in accordance with NAD 8.14.

1190120302A01 - Process a tagout in accordance with GNP 3.3.1.

5 . d		RO Value	SRO Value			
KA	GENERIC 2.2.18	2.3	3.6	Ref. Section	Ref. Page	Ref. Revision
Refs.	Safety Shutdown Assessment		GNP 8.4.1	2.7	2	C

Objs. 1190310302K01 - Explain the responsibilities of the refueling SRO per NAD 2.7.

6 . b		RO Value	SRO Value			
KA	GENERIC 2.2.21	2.3	3.5	Ref. Section	Ref. Page	Ref. Revision
Refs.	Operations Department Instructions			Operating Instructions & Duties - TESTING OF SAFETY SYSTEM COMPONENTS		March 14, 1989

Objs. 0330000004K08 - IDENTIFY THE COMPONENTS THAT AFFECT THE SI SYSTEM OPERABILITY PER TECHNICAL SPECIFICATIONS SECTION 3.3.

7 . c		RO Value	SRO Value			
KA	GENERIC 2.2.26	2.5	3.7	Ref. Section	Ref. Page	Ref. Revision
Refs.	KNPP Technical Specifications			3.8.a.3	TS 3.8-1	Amend 132
	Reactor Shutdown		N-CRD-49C	4.4.3 & 4	1, 2	K

Objs. 1190310302A01 - Supervise Refueling operations in accordance with NAD 2.7.

8 . b		RO Value	SRO Value			
KA	GENERIC 2.2.28	2.6	3.5	Ref. Section	Ref. Page	Ref. Revision
Refs.	Refueling Equipment Operating Instructions		RF-03.02	5.1.1.4	2	A
	Fuel Handling (FH)		Number 53	5.1	53-38	A
	Kewaunee Refueling Operations		NAD-02.07	4.2.1.7	2	B

Objs. 1190310302K01 - Explain the responsibilities of the refueling SRO per NAD 2.7

9 . a		RO Value	SRO Value			
KA	GENERIC 2.2.30	3.5	3.3	Ref. Section	Ref. Page	Ref. Revision
Refs.	Reactor Cavity Draining with Fuel or Upper Internals Installed		N-FH-53E	2.6 & 3.8	1-2	H
	KNPP Technical Specifications			3.8.3	TS 3.8-1	Amend 132

Objs. 0530000001K05 - LOCATE ALL SYSTEM CONTROLS AND INDICATIONS WHILE OPERATING IN SUPPORT OF THE REFUELING FROM THE CONTROL ROOM.

0530080101A01 - GIVEN FUEL OR UPPER INTERNALS INSTALLED, DRAIN THE REACTOR CAVITY IN ACCORDANCE WITH N-FH-53E.

10 . d		RO Value	SRO Value			
KA	GENERIC 2.3.2	2.5	2.9	Ref. Section	Ref. Page	Ref. Revision
Refs.	ALARA Program		NAD-01.23	5.5 & 5.7	2	B
	Radiation Worker Training		T-GET-LP RWT	ALARA - III.	31-35	B

Objs. T-GET-LP RWT: Enabling Objective 6. - Apply basic methods to minimize radiation exposure to a given scenario

T-GET-LP RWT: Learning Objectives for ALARA: Calculate stay time given a dose rate, current exposure, and an exposure limit.

11 . b		RO Value	SRO Value			
KA	GENERIC 2.3.4	2.5	3.1	Ref. Section	Ref. Page	Ref. Revision
Refs.	Emergency Radiation Controls		EP-AD-11	4.1.4, 4.2 & 4.3	2-3	P

Objs. T-GET-LP RWT: Enabling Objective 5. - Identify the federal limits and plant administrative guidelines on radiation dose

12 . d		RO Value	SRO Value			
KA	GENERIC 2.3.11	2.7	3.2	Ref. Section	Ref. Page	Ref. Revision
Refs.	Offsite Dose Calculation Manual			Specification 3.1 & Table 3.1	3-2, 3-13	8
	Radiological Liquid Discharges (Batch Mode)	SP 32A-136		3.4	2	U

Objs. 32A0110104A01 - WHEN DIRECTED, DISCHARGE THE WASTE CONDENSATE TANKS TO THE AUXILIARY BUILDING STANDPIPE IN ACCORDANCE WITH N-LWP-32A-3.

13 . d		RO Value	SRO Value			
KA	GENERIC 2.4.12	3.4	3.9	Ref. Section	Ref. Page	Ref. Revision
Refs.	User's Guide for Integrated Plant Emergency Operating Procedures	UG-0		k.5, n	3, 5	C

Objs. E000020501K03 - LIST THE MAJOR STEPS IN THE HIGH LEVEL ACTION SUMMARY FOR E-0, "REACTOR TRIP OR SAFETY INJECTION," PER THE IPEOP BACKGROUND DOCUMENT.

E000020501K04 - LIST THE IMMEDIATE OPERATOR ACTIONS, GIVEN A REACTOR TRIP, IN ACCORDANCE WITH E-0.

14 . d

		RO Value	SRO Value			
KA	GENERIC 2.4.21	3.7	4.3	Ref. Section	Ref. Page	Ref. Revision
Refs.	User's Guide for Integrated Plant Emergency Operating Procedures		UG-0	I.1, 2, 5 & 9	4	C
	Heat Sink		F-0.3		1	D

Objs. 46A0000001K02 - Discuss the design characteristics of the Honeywell Plant Computer, for any mode of operation, per System Description 46A.

15 . a

		RO Value	SRO Value			
KA	GENERIC 2.4.35	3.3	3.5	Ref. Section	Ref. Page	Ref. Revision
Refs.	POST-LOCA Hydrogen Control		N-RBV-18C	4.1.1.b	2	K

Objs. 0180000004K05 - LOCATE ALL LOCAL CONTROLS AND INDICATIONS FOR THE RBV SYSTEM PER M-547 AND M-602.

0180010104A01 - WHEN DIRECTED, STARTUP THE POST-LOCA HYDROGEN CONTROL SYSTEM FOR HYDROGEN DILUTION OF CONTAINMENT WITH INSTRUMENT AIR AVAILABLE IN ACCORDANCE WITH N-RBV-18C.

16 . a

		RO Value	SRO Value			
KA	GENERIC 2.4.38	2.2	4.0	Ref. Section	Ref. Page	Ref. Revision
Refs.	KNPP Emergency Plan			Appendix A	A-6	24
	KNPP Response to Alert or Higher		EP-AD-4	3.3	1-2	AB

Objs. 1190040502K01 - Identify the responsibilities of the Emergency Director that shall not be delegated while responding to an emergency classification of Alert or higher in accordance with EP-AD-4.

17 . b		RO Value	SRO Value			
KA	GENERIC 2.4.49	4.0	4.0	Ref. Section	Ref. Page	Ref. Revision
Refs.	Response to Nuclear Power Generation		FR-S.1	Step 5	4	M

Objs. FRS0020501K06 - LIST THE IMMEDIATE OPERATOR ACTIONS, GIVEN A NUCLEAR GENERATION/ATWS CONDITION, IN ACCORDANCE WITH FR-S.1.

FRS0020501K05 - IDENTIFY THE CONDITIONS WHICH WOULD REQUIRE A PROCEDURAL TRANSITION, WHILE RESPONDING TO A NUCLEAR GENERATION/ATWS, IN ACCORDANCE WITH FR-S.1.

18 . b			RO Value	SRO Value			
KA	001	K2.05	3.1*	3.5	Ref. Section	Ref. Page	Ref. Revision
Refs.	System Description - Rod Control Number 49 & Rod Position Indication (CRD)				3.2	4	A
	Circuit Diagram - 4160 & 480 V Power Sources			E-240			AQ
	Circuit Diagram - DC Aux. And Emergency AC			E-233			AP

Objs. 0490000001K02 - Discuss the design characteristics of the Rod Control System, for any mode of operation, per System Description 49 and USAR sections 1.2, 1.3, 1.8, 3.1, 3.2, 7.2, 7.3, 14.1 AND 14.2.

0490000004K06 - List the power supplies for the following components per N-CRD-49-CL: Rod Drive MG Sets

Contributor to Core Damage Sequence: Loss of AC Bus – 2%

19 . a

			RO Value	SRO Value	Ref. Section	Ref. Page	Ref. Revision
KA	001	K3.02	3.4*	3.5			
Refs.	Continuous Rod Withdrawal			E-CRD-49B	2.0, 3.2	1, 2	H
	KNPP USAR				14.1.2	14.1-5 - 7	15

Objs. 0490000001K02 - DISCUSS THE DESIGN CHARACTERISTICS OF THE ROD CONTROL SYSTEM, FOR ANY MODE OF OPERATION, PER SYSTEM DESCRIPTION 49 AND USAR SECTIONS 1.2, 1.3, 1.8, 3.1, 3.2, 7.2, 7.3, 14.1 AND 14.2.

20 . d

			RO Value	SRO Value	Ref. Section	Ref. Page	Ref. Revision
KA	002	A3.03	4.4	4.6			
Refs.	Integrated Logic - RCS			E-2038			Z
	Logic Diagram - PRZR Pressure & Level Control			XK100-154			4

Objs. 0360000001K17 - EXPLAIN THE RESPONSE TO THE OPERATION OF ALL PRESSURIZER PRESSURE CONTROL SYSTEM CONTROLS IN ACCORDANCE WITH LOGIC DIAGRAMS/SCHEMATICS E2037, E2038, E2039, E2042, XK100-148, XK100-150, XK100-154, XK100-155, AND XK100-546.

21 . b

			RO Value	SRO Value	Ref. Section	Ref. Page	Ref. Revision
KA	002	K3.03	4.2	4.6			
Refs.	Flow Diagram - RCS			OP XK100-10			BE
	Reactor Coolant Leak			A-RC-36D	4.3	5	AB

Objs. 0360030401A01 - GIVEN AN UNIDENTIFIED RCS LEAK, RESPOND IN ACCORDANCE WITH A-RC-36D.

0360000001K32 - DISCUSS THE DESIGN CHARACTERISTICS OF THE REACTOR COOLANT SYSTEM FOR ANY MODE OF OPERATION PER SYSTEM DESCRIPTION 36 AND USAR SECTIONS 4.0 AND 6.5.

Contributor to Core Damage Sequence: Small LOCA – 7%

22 . a

			RO Value	SRO Value			
KA	003	A1.06	2.9	3.1	Ref. Section	Ref. Page	Ref. Revision
Refs.	Reactor Coolant Pump Operation N-RC-36A				4.3.2	4	Y
	System Description - Reactor Coolant (RC)			Number 36	3.6.3	36-27	A

Objs. 0360000001K32 - DISCUSS THE DESIGN CHARACTERISTICS OF THE REACTOR COOLANT SYSTEM FOR ANY MODE OF OPERATION PER SYSTEM DESCRIPTION 36 AND USAR SECTIONS 4.0 AND 6.5.

0360000001K35 - DISCUSS THE DESIGN CHARACTERISTICS OF THE PRESSURIZER PRESSURE CONTROL SYSTEM FOR ANY MODE OF OPERATION PER SYSTEM DESCRIPTION 36 AND USAR SECTIONS 7.2 AND 7.3.

23 . d

			RO Value	SRO Value			
KA	003	K1.10	3.0	3.2	Ref. Section	Ref. Page	Ref. Revision
Refs.	Accident & Transient Analysis				VI.5	VI-18 - 20	1987
	PLANT SPECIFIC BACKGROUND INFORMATION FOR KNPP IPEOP ES-0.1			ES-0.1	4. Step 9	22-23	I

Objs. 2030000001K06 - Describe the initial trends of various primary and secondary parameters following a DRCF event.

E000040501K06 - DISCUSS THE CONDITIONS REQUIRED TO SUPPORT OR INDICATE NATURAL CIRCULATION, GIVEN A REACTOR TRIP RECOVERY, IN ACCORDANCE WITH ES-0.1.

24 . d

			RO Value	SRO Value	Ref. Section	Ref. Page	Ref. Revision
KA	004	A3.01	3.5	3.7			
Refs.	Control Room Alarm Response Procedures			CVC-35	47044L		A
	Integrated Logics - CVCS			E-2024			
	Integrated Logics - CVCS			E-2023			X

Objs. 0350260101A01 - PERFORM AN AUTOMATIC MAKEUP, WHILE CONTROLLING RCS BORON CONCENTRATION IN ACCORDANCE WITH N-CVC-35A.

0350000001K02 - EXPLAIN THE RESPONSE TO THE OPERATION OF ALL CHEMICAL AND VOLUME CONTROL - CHARGING AND LETDOWN SYSTEM CONTROLS IN ACCORDANCE WITH LOGIC DIAGRAMS E2020, E2023 THROUGH E2028, E2030, E3000, AND E2039.

25 . d

			RO Value	SRO Value	Ref. Section	Ref. Page	Ref. Revision
KA	004	K6.24	2.5	2.6			
Refs.	System Description - Chemical and Volume Control (CVC)			Number 35	3.2.8	35-16, 17	Orig
	Integrated Logic - CVCS			E-2023			X
	Flow Diagram - CVCS			OP XK100-36			AS

Objs. 0350000001K06 - DISCUSS THE DESIGN CHARACTERISTICS OF THE CVC SYSTEM FOR ANY MODE OF OPERATION PER SYSTEM DESCRIPTION 35 AND USAR SECTIONS 9.2, 10A.5, AND 14.1.

0350000001K02 - EXPLAIN THE RESPONSE TO THE OPERATION OF ALL CHEMICAL AND VOLUME CONTROL - CHARGING AND LETDOWN SYSTEM CONTROLS IN ACCORDANCE WITH LOGIC DIAGRAMS E2020, E2023 THROUGH E2028, E2030, E3000, AND E2039.

26 . d

			RO Value	SRO Value			
KA	005	2.2.25	2.5	3.7	Ref. Section	Ref. Page	Ref. Revision
Refs.	Technical Specifications for Kewaunee Nuclear Plant				3.1.a.2.B.	TS 3.1-2	Amend 100
	Technical Specifications for Kewaunee Nuclear Plant				BASES (3.1.a.2.)	TS B3.1-2	Amend 108

Objs. 0340000004K08 - IDENTIFY THE COMPONENTS THAT AFFECT THE RHR SYSTEM OPERABILITY PER TECHNICAL SPECIFICATIONS SECTIONS 3.1 AND 3.3.

0340000001K02 - DISCUSS THE DESIGN CHARACTERISTICS OF THE RESIDUAL HEAT REMOVAL SYSTEM, FOR ANY MODE OF OPERATION, PER SYSTEM DESCRIPTION 34 AND USAR SECTION 6.1, 6.2, 6.5, AND 9.3.

PRA SYSTEM Importance: RHR – 8

27 . c

			RO Value	SRO Value			
KA	005	K6.03	2.5	2.6	Ref. Section	Ref. Page	Ref. Revision
Refs.	Residual Heat Removal System Operation			N-RHR-34	2.3, 2.12, 4.2	1, 3, 8	AM
	Leakage into Component Cooling System			A-CC-31B	4.8.3	3	I
	Flow Diagram Aux. Coolant			OP XK100-18			AK

Objs. 0340000001K02 - DISCUSS THE DESIGN CHARACTERISTICS OF THE RESIDUAL HEAT REMOVAL SYSTEM, FOR ANY MODE OF OPERATION, PER SYSTEM DESCRIPTION 34 AND USAR SECTION 6.1, 6.2, 6.5, AND 9.3.

0340000004K08 - IDENTIFY THE COMPONENTS THAT AFFECT THE RHR SYSTEM OPERABILITY PER TECHNICAL SPECIFICATIONS SECTIONS 3.1 AND 3.3.

0310010401A01 - Given leakage into the Component Cooling System, identify the source in accordance with A-CC-31B.

28 . C

			RO Value	SRO Value			
KA	006	A3.01	4.0*	3.9	Ref. Section	Ref. Page	Ref. Revision
Refs.	KNPP USAR				6.2.2 Accumulators	6.2-10	15
	KNPP USAR				14.3.2	14.3-8	15
	Integrated Logic - SI			E-2034			R

Objs. 0330000001K02 - DISCUSS THE DESIGN CHARACTERISTICS OF THE SAFETY INJECTION SYSTEM, FOR ANY MODE OF OPERATION, PER SYSTEM DESCRIPTION 33 AND USAR SECTIONS 1.3, 1.5, 1.8, 6.2, 7.2, 7.5, 14.2, 14.3.

0330000001K04 - EXPLAIN THE RESPONSE TO THE OPERATION OF ALL SAFETY INJECTION SYSTEM CONTROLS IN ACCORDANCE WITH LOGIC DIAGRAMS/SCHEMATICS E1635, E2032, E2033, E2034, E2035, XK100-148 AND XK100-150.

Contributor to Core Damage Sequence: Large LOCA – 9%; PRA SYSTEM Importance: SI Accumulator – 5

29 . C

			RO Value	SRO Value			
KA	006	2.4.10	3.0	3.1	Ref. Section	Ref. Page	Ref. Revision
Refs.	Control Room Alarm Response Procedures			RHR-34	47023-I	1	A
	Technical Specifications for Kewaunee Nuclear Plant				3.3.b.2	3.3-2	Amend 116

Objs. 0330000004K08 - IDENTIFY THE COMPONENTS THAT AFFECT THE SI SYSTEM OPERABILITY PER TECHNICAL SPECIFICATIONS SECTION 3.3.

0340000004K08 - IDENTIFY THE COMPONENTS THAT AFFECT THE RHR SYSTEM OPERABILITY PER TECHNICAL SPECIFICATIONS SECTIONS 3.1 AND 3.3.

PRA SYSTEM Importance: RHR – 8

30 . c

			RO Value	SRO Value			
KA	007	A1.03	2.6	2.7	Ref. Section	Ref. Page	Ref. Revision
Refs.	Pressurizer Relief Tank Operation			N-RC-36B	2.0	1, 2	N

Objs. 0360000001K34 - DISCUSS THE DESIGN CHARACTERISTICS OF THE PRESSURIZER AND PRT SYSTEM FOR ANY MODE OF OPERATION PER SYSTEM DESCRIPTION 36 AND USAR SECTION 4.2.

0360230101A01 - WHEN DIRECTED, ESTABLISH NORMAL OPERATING CONDITIONS IN THE PRT IN ACCORDANCE WITH N-RC-36B.

31 . b

			RO Value	SRO Value			
KA	010	K5.02	2.6	3.0*	Ref. Section	Ref. Page	Ref. Revision
Refs.	Steam Tables						

Objs. 0360000001K34 - DISCUSS THE DESIGN CHARACTERISTICS OF THE PRESSURIZER AND PRT SYSTEM FOR ANY MODE OF OPERATION PER SYSTEM DESCRIPTION 36 AND USAR SECTION 4.2.

DELETED DRB

32 . d

			RO Value	SRO Value			
KA	012	K1.05	3.8*	3.9	Ref. Section	Ref. Page	Ref. Revision
Refs.	Logic Diagram - Reactor Trip Signals			Xk100-144			5C
	Integrated Logic - DG Electric			E-1635			O

Objs. 0470000001K02 - Explain the response to the operation of all Reactor Control & Protection system controls in accordance with Logic Diagrams/Schematics E2042, E2043, E2044, E2051-1, E2051-2, XK100-144...

0490000004K03 - Explain the basic principles of operation for the following major components of the CRD System per System Description 49:1) Rod Drive MG Sets2) Reactor Trip and Bypass Breakers

33 . a

			RO Value	SRO Value			
KA	012	K6.01	2.8	3.3	Ref. Section	Ref. Page	Ref. Revision
Refs.	Bistable Tripping for Failed Reactor Protection or Safeguards Inst.			A-MI-87	Attachment I PT-449	50	M
	Instrument Block Diagram - PRZR Press						XK100-546 2R
	Instrument Block Diagram - dT-Tavg			XK100-551			1N

Objs. 0470000001K06 - Discuss the purposes/functions of the Reactor Control & Protection system and it's components for any mode of operation per System Description 47 and USAR sections 1.3, 7.2.

0470000001K02 - Explain the response to the operation of all Reactor Control & Protection system controls in accordance with Logic Diagrams/Schematics E2042, E2043, E2044, E2051-1, E2051-2, XK100-144,..., XK100-546 through XK100-559

34 . c

			RO Value	SRO Value			
KA	013	A4.01	4.5	4.8	Ref. Section	Ref. Page	Ref. Revision
Refs.	Response to High Containment Pressure			FR-Z.1	Step 3	3-4	
	Integrated Logic Diagram ICS Sys			E-1604			U
	Integrated Logic Diagram ICS System			E-2012			K

Objs. 0230000004K06 - DESCRIBE THE ICS SYSTEM'S LOCAL COMPONENT OPERATION USING THE FOLLOWING LOGIC DIAGRAMS: E1604 AND E2012.

0550000001K03 - EXPLAIN THE RESPONSE TO THE OPERATION OF ALL ENGINEERED SAFETY FEATURES SYSTEM CONTROLS IN ACCORDANCE WITH LOGIC DIAGRAMS/SCHEMATICS

FRZ0010501K04 - GIVEN A HIGH CONTAINMENT PRESSURE CONDITION, EXPLAIN THE BASIS FOR ACTIONS TAKEN, PER FR-Z.1 BACKGROUND DOCUMENT.

35 . c

			RO Value	SRO Value			
KA	015	K2.01	3.3	3.7	Ref. Section	Ref. Page	Ref. Revision
Refs.	Control Room Alarm Response Procedures			NI-48	47031-L		A
	Circuit Diagram - DC Aux & Emergency AC			E233			AP

Objs. 0480000001K02 - DISCUSS THE DESIGN CHARACTERISTICS OF THE NUCLEAR INSTRUMENTATION SYSTEM, FOR ANY MODE OF OPERATION, PER SYSTEM DESCRIPTION 048 AND USAR SECTIONS 7.2 AND 7.4.

0480110401A01 - GIVEN A POWER RANGE OVERPOWER ROD STOP ANNUNCIATOR, RESPOND IN ACCORDANCE WITH ALARM RESPONSE PROCEDURE 47031-L.

36 . b

			RO Value	SRO Value			
KA	015	K5.15	3.3	3.7	Ref. Section	Ref. Page	Ref. Revision
Refs.	Kewaunee Core Control Theory				Chapter 3 E & Fig 3-10	3-15	

Objs. 05A0020401A01 - GIVEN A FEEDWATER PUMP TRIP, RESPOND IN ACCORDANCE WITH A-FW-05A.

37 . d

			RO Value	SRO Value			
KA	017	2.1.12	2.9	4.0	Ref. Section	Ref. Page	Ref. Revision
Refs.	Technical Specifications for Kewaunee Nuclear Plant				3.11.c.	TS 3.11-1	

Objs. 0500000001K02 - DISCUSS THE DESIGN CHARACTERISTICS OF THE INCORE INSTRUMENTATION SYSTEM AND ICCMS FOR ANY MODE OF OPERATION PER SYSTEM DESCRIPTION 050 AND USAR SECTION 7.6.

0500010101A02 - OPERATE THE ICCM SYSTEM DURING STEADY STATE CONDITIONS IN ACCORDANCE WITH N-II-50.

38 . c

			RO Value	SRO Value			
KA	026	A1.05	3.1	3.4	Ref. Section	Ref. Page	Ref. Revision
Refs.	Technical Specifications for Kewaunee Nuclear Plant				Basis (3.3)	TS B3.3-3	Amend 143
	System Description - Internal Containment Spray (ICS)			Number 23	3.7	23-10	B

Objs. 0230000004K02 - IDENTIFY AND EXPLAIN THE PURPOSE OF THE MAJOR COMPONENTS OF THE ICS SYSTEM PER USAR SECTIONS 6.3 AND 6.4 AND SYSTEM DESCRIPTION 23.

Contributor to Core Damage Sequence: Large LOCA – 9%

39 . c

			RO Value	SRO Value			
KA	026	2.4.11	3.4	3.6	Ref. Section	Ref. Page	Ref. Revision
Refs.	480V AC Supply Distribution System			A-ELV-40	4.5.4 NOTES	5	N
	Technical Specifications for Kewaunee Nuclear Plant				3.3.c.1.A.1	TS 3.3-4	Amend 116
	Technical Specifications for Kewaunee Nuclear Plant				3.7.c	TS 3.7-2	Amend 122

Objs. 0230000004K08 - IDENTIFY THE COMPONENTS THAT AFFECT THE ICS SYSTEM OPERABILITY PER TECHNICAL SPECIFICATIONS SECTION 3.3.

0400050401A01 - GIVEN A SAFEGUARDS 480V BUS UNDERVOLTAGE, RESPOND IN ACCORDANCE WITH A-ELV-40.

0400000004K07 - IDENTIFY THE COMPONENTS THAT AFFECT THE ELV SYSTEM OPERABILITY PER TECHNICAL SPECIFICATIONS SECTION 3.7.

40 . a			RO Value	SRO Value			
KA	028	A2.02	3.5	3.9	Ref. Section	Ref. Page	Ref. Revision
Refs.	POST-LOCA Hydrogen Control			N-RBV-18C	2.4	1	K

Objs. 0180000001K09 - DISCUSS THE DESIGN CHARACTERISTICS OF THE POST LOCA HYDROGEN CONTROL SYSTEM FOR ANY MODE OF OPERATION PER SYSTEM DESCRIPTION 18 AND USAR SECTION 14.3.9.

0180040104A01 - WHEN DIRECTED, STARTUP THE POST-LOCA HYDROGEN CONTROL SYSTEM TO VENT AND FILTER THE CONTAINMENT ATMOSPHERE THROUGH SHIELD BUILDING VENT IN ACCORDANCE WITH N-RBV-18C.

Contributor to Core Damage Sequence: Medium LOCA – 13%

41 . a			RO Value	SRO Value			
KA	028	K1.01	2.5*	2.5	Ref. Section	Ref. Page	Ref. Revision
Refs.	POST-LOCA Hydrogen Control			N-RBV-18C	CAUTION step 4.1.6.4	8	K
	KNPP USAR				5.5.1	5.5.2	9

Objs. 0240000001K01- Discuss the design characteristics of the Shield Building Ventilation System, for any mode of operation, per System Description 24 and USAR Section 5.5.

0180040104A01 - WHEN DIRECTED, STARTUP THE POST-LOCA HYDROGEN CONTROL SYSTEM TO VENT AND FILTER THE CONTAINMENT ATMOSPHERE THROUGH SHIELD BUILDING VENT IN ACCORDANCE WITH N-RBV-18C.

Contributor to Core Damage Sequence: Medium LOCA – 13%

42 . C

			RO Value	SRO Value			
KA	033	A1.01	2.7	3.3	Ref. Section	Ref. Page	Ref. Revision
Refs.	Abnormal Spent Fuel Pool Cooling and Cleanup System Operation			A-SFP-21	4.2.2	3, 4	N
	Emergency Spent Fuel Pool Cooling and Cleanup System (SFP)			E-SFP-21	2.0	1	N

Objs. 0210000001K02 - Discuss the design characteristics of the Spent Fuel Pool Cooling and Clean Up System, for any mode of operation, per System Description 21 and USAR Section 9.3.

0210020401A02 - GIVEN A SPENT FUEL POOL LEVEL LOW CONDITION, RESPOND IN ACCORDANCE WITH A-SFP-21

43 . C

			RO Value	SRO Value			
KA	035	K5.03	2.8	3.1	Ref. Section	Ref. Page	Ref. Revision
Refs.	Kewaunee Thermodynamics Theory				Chapter 12	12-30, 31	

Objs. 05A0090401A01 - GIVEN A S/G PROGRAM LEVEL DEVIATION ANNUNCIATOR, RESPOND IN ACCORDANCE WITH ALARM RESPONSE PROCEDURE 47062-A(D).

44 . b

			RO Value	SRO Value			
KA	039	K3.06	2.8*	3.1	Ref. Section	Ref. Page	Ref. Revision
Refs.	System Description - Main Steam & Steam Dump (MS)			Number 6	3.11	6-22, 23	B
	Integrated Logic - Main Steam & Steam Dump			E-1626			U

Objs. 0060000001K15 - EXPLAIN THE RESPONSE TO THE OPERATION OF ALL STEAM DUMP SYSTEM CONTROLS IN ACCORDANCE WITH LOGIC DIAGRAM/SCHEMATIC E1626 AND XK100-153.

45 . d

			RO Value	SRO Value			
KA	059	A2.06	2.7*	2.9*	Ref. Section	Ref. Page	Ref. Revision
Refs.	Abnormal Heater & MS Drain System & Bleed Steam System			A-HD-11	4.2	4	H
	Kewaunee Thermodynamics Theory				Increasing Thermodynamic Efficiency...	8-51, 52	
	Shift Instrument Channel Checks - Operating				SP 87-125	2.1	1 AZ

Objs. 2060000001K01 - List the various transients and accidents leading to either positive or negative reactivity insertion and their possible causes.

46 . b

			RO Value	SRO Value			
KA	059	A3.02	2.9	3.1	Ref. Section	Ref. Page	Ref. Revision
Refs.	Instrument Block Diagram - RX Protection & Control (SGWLC)			XK100-554			1V
	Control Room Alarm Response Procedures			FW05	47062-A/D		A
	Control Room Alarm Response Procedures			FW05	47062-B/E		Orig

Objs. 05A0000001K04 - EXPLAIN THE RESPONSE TO THE OPERATION OF ALL MAIN FEEDWATER SYSTEM CONTROLS IN ACCORDANCE WITH LOGIC DIAGRAMS/SCHEMATICS: E1624, E1625, E2006.

05A0080401A01 - GIVEN A S/G BYPASS CV LEVEL DEVIATION ANNUNCIATOR, RESPOND IN ACCORDANCE WITH ALARM RESPONSE PROCEDURE 47062-B(E).

0540110401A01 - GIVEN A P-485 INSTRUMENT FAILURE, RESPOND IN ACCORDANCE WITH A-TB-54.

47 . c

			RO Value	SRO Value			
KA	061	K3.01	4.4	4.6	Ref. Section	Ref. Page	Ref. Revision
Refs.	Flow Diagram - Service Water System			OP M202			CE
	Technical Specifications for Kewaunee Nuclear Plant				3.4.b.1.A & 3.4.b.5	TS 3.4-2, 3	Amend 123

Objs. 05B0000004K08 - IDENTIFY THE COMPONENTS THAT AFFECT THE AFW SYSTEM OPERABILITY PER TECHNICAL SPECIFICATIONS SECTION 3.4.

PRA SYSTEM Importance: Service Water – 3; MD AFW – 7

48 . a

			RO Value	SRO Value			
KA	061	K4.02	4.5	4.6	Ref. Section	Ref. Page	Ref. Revision
Refs.	System Description - Auxiliary Feedwater (AFW)			Number 5B	3.2.3	5B-10, 11	1
	Instrument Block Diagram - RX Protection & Control (SGWLC)			XK100-554			1V
	Integrated Logic - Aux. Feedwater						E-1602 AW

Objs. 05B0000001K01 - DISCUSS THE DESIGN CHARACTERISTICS OF THE AUXILIARY FEEDWATER SYSTEM FOR ANY MODE OF OPERATION PER SYSTEM DESCRIPTION 05B AND USAR SECTIONS 1.2.8.e, 6.6, AND 10A.3.5.

05B0000001K02 - EXPLAIN THE RESPONSE TO THE OPERATION OF ALL AUXILIARY FEEDWATER SYSTEM CONTROLS IN ACCORDANCE WITH LOGIC DIAGRAMS/SCHEMATICS: E1602, E2802, XK100-149, XK100-157.

PRA SYSTEM Importance: MD AFW – 7

49 . a

			RO Value	SRO Value			
KA	062	2.1.32	3.4	3.8	Ref. Section	Ref. Page	Ref. Revision
Refs.	4160V AC Supply and Distribution System Operation			N-EHV-39	2.11	2	N
	Technical Specifications for Kewaunee Nuclear Plant				3.3.b.1.a	TS 3.3-2	Amend 116
	Technical Specifications for Kewaunee Nuclear Plant				3.0.c	TS 3.0-1	Amend 119

Objs. 0390000004K08 - IDENTIFY THE COMPONENTS THAT AFFECT THE EHV SYSTEM OPERABILITY PER TECHNICAL SPECIFICATIONS SECTION 3.7.

PRA SYSTEM Importance: Diesel – 6; RHR – 8

50 . a

			RO Value	SRO Value			
KA	062	K4.10	3.1	3.5	Ref. Section	Ref. Page	Ref. Revision
Refs.	Circuit Diagram - DC Aux. & Emergency AC			E-233			AP
	System Description - DC & Emergency AC Electrical Dist (EDC)			Number 38	3.11	38-17	A

Objs. 0380000001K06 - DISCUSS THE DESIGN CHARACTERISTICS OF THE DC AND EMERGENCY AC SUPPLY SYSTEM FOR ANY MODE OF OPERATION PER SYSTEM DESCRIPTION 38 AND USAR SECTIONS 1.8 AND 8.0.

51 . c

			RO Value	SRO Value			
KA	063	K4.01	2.7	3.0*	Ref. Section	Ref. Page	Ref. Revision
Refs.	DC Supply and Distribution System			A-EDC-38	4.7.3	5	U
	Circuit Diagram - DC Aux. & Emergency AC			E-233			AP

Objs. 0380000001K02 - EXPLAIN THE RESPONSE TO THE OPERATION OF ALL DC AND EMERGENCY AC SUPPLY SYSTEM CONTROLS IN ACCORDANCE WITH LOGIC DIAGRAMS E233 AND E3626.

0380060401A01 - GIVEN A BRC-103/BRD-103 FEEDER BREAKER UNDERVOLTAGE CONDITION, RESPOND IN ACCORDANCE WITH A-EDC-38.

PRA SYSTEM Importance: Batteries – 4

52 . b

			RO Value	SRO Value			
KA	064	A1.08	3.1	3.4	Ref. Section	Ref. Page	Ref. Revision
Refs.	Diesel Generator A Manual Operation			N-DGM-10A	2.5, 4.3.1	1, 9	F
	Integrated Logic - Diesel Generator Electric			E-2022			N
	Integrated Logic - Diesel Generator Electric			E-1621			AK

Objs. 0420000001K02 - Explain the response to the operation of all Diesel Generator Electric system controls in accordance with Logic diagrams/schematics E1621, E1622, E1634 THROUGH E1639, E2000, E2001, E2002, E2022 AND E2900.

0420000001K06 - Discuss the design characteristics of the Diesel Generator Electric system for any mode of operation per System Description 42 and USAR sections 1.8, 8 and 14.1.

PRA SYSTEM Importance: Bus 5 or 6 – 2

53 . d

			RO Value	SRO Value			
KA	064	A4.10	3.3	3.4	Ref. Section	Ref. Page	Ref. Revision
Refs.	Loss of Reactor Or Secondary Coolant			E-1	Step 16 RNO a.2)	10	M
	PLANT SPECIFIC BACKGROUND INFORMATION FOR KNPP IPEOP E-1			E-1	4. Step 16	52	J
	System Description - Diesel Generator Electric (DGE)			Number 42	3.1	42-1, 2	Orig

Objs. E000010501K08 - Given a Reactor Trip, explain the basis for actions taken per E-0 Background Document.

0420000004K03 - Explain the basic principles of operation for the DGE System and the major components and equipment per System Description 42.

E010010501K04 - Given a Loss of Reactor or Secondary Coolant, explain the basis for actions taken per E-1 Background Document

54 . b

			RO Value	SRO Value			
KA	068	A4.03	3.9	3.8	Ref. Section	Ref. Page	Ref. Revision
Refs.	CVC Monitor Tanks and Pumps			N-CVC-35M	4.3	2-3	E
	Abnormal Radiation Monitoring System			A-RM-45	3.8	5	AC
	Integrated Logic - CVCS			E-2047			K

Objs. 32A0000001K02 - EXPLAIN THE RESPONSE TO THE OPERATION OF ALL LIQUID WASTE SYSTEM CONTROLS IN ACCORDANCE WITH LOGIC DIAGRAMS E2046 AND E2047.

0450160401K01 - LIST THE IMMEDIATE OPERATOR ACTIONS, GIVEN AN R-18 RADIATION INDICATION HIGH CONDITION IN ACCORDANCE WITH A-RM-45.

32A0000004K06 - DESCRIBE THE LIQUID WASTE PROCESSING SYSTEM'S LOCAL COMPONENT OPERATION USING THE FOLLOWING LOGIC DIAGRAMS: E2046, E2047, AND E2048.

55 . a

			RO Value	SRO Value			
KA	072	A2.02	2.8	2.9	Ref. Section	Ref. Page	Ref. Revision
Refs.	Abnormal Radiation Monitoring System			A-RM-45	3.1.1	1	AC

Objs. 0450010401K01 - LIST THE IMMEDIATE OPERATOR ACTIONS, GIVEN A RADIATION INDICATION HIGH CONDITION ON R-1 THROUGH R-10, IN ACCORDANCE WITH A-RM-45.

56 . c

			RO Value	SRO Value			
KA	078	K4.03	3.1*	3.3*	Ref. Section	Ref. Page	Ref. Revision
Refs.	Control Room Alarm Response Procedures			AS-01	47052-I	1	D

Objs. 0010000001K02 - Discuss the design characteristics of the Station and Instrument Air system, for any mode of operation, per System Description 01.

0010040404A03 - Given an alarm received on an air compressor, respond in accordance with A-AS-01.

57 . d

			RO Value	SRO Value			
KA	086	A2.01	2.9	3.1	Ref. Section	Ref. Page	Ref. Revision
Refs.	Emergency Operating Procedure - Fire			E-FP-08	3.1	1	AC
	Integrated Logic - Fire Protection			E-1619			W
	System Description - Fire Protection (FP)			Number 8	3.3	8-12, 13	A

Objs. 0080000001K02 - DISCUSS THE DESIGN CHARACTERISTICS OF THE FIRE PROTECTION SYSTEM, FOR ANY MODE OF OPERATION, PER SYSTEM DESCRIPTION 08 AND USAR SECTION 7.7-5.

0080000001K04 - EXPLAIN THE RESPONSE TO THE OPERATION OF ALL FIRE PROTECTION SYSTEM CONTROLS IN ACCORDANCE WITH LOGIC DIAGRAM E1619.

0080010501K01- LIST THE IMMEDIATE OPERATOR ACTIONS, GIVEN A FIRE, IN ACCORDANCE WITH E-FP-08.

58 . a

			RO Value	SRO Value	Ref. Section	Ref. Page	Ref. Revision
KA	007	EA1.04	3.6	3.7			
Refs.	System Description - Reactor Coolant (RC)			Number 36	3.9.4	36-40	A
	Flow Diagram - RCS			OP XK100-10			BE
	PLANT SPECIFIC BACKGROUND INFORMATION FOR KNPP IPEOP E-1			E-1	2.1	3	J

Objs. 0360000004K03 - Explain the basic principles of operation for the RC System and the major components and equipment per System Description 36.

0470000001K02 - Explain the response to the operation of all Reactor Control & Protection system controls in accordance with Logic Diagrams/Schematics...

Contributor to Core Damage Sequence: Small LOCA – 7%

59 . a

			RO Value	SRO Value	Ref. Section	Ref. Page	Ref. Revision
KA	008	AA2.04	3.2	3.4			
Refs.	Control Room Alarm Response Procedures			RC-36	47042B		A
	Technical Specifications for Kewaunee Nuclear Plant				3.1.a.5.A.1	3.1-3	Amend 108
	Technical Specifications for Kewaunee Nuclear Plant				3.1.d.1 & 3	3.1-9	Amend 118

Objs. 0360230401A01 - GIVEN A PRESSURIZER PORV DISCHARGE TEMP HIGH ANNUNCIATOR, RESPOND IN ACCORDANCE WITH ALARM RESPONSE PROCEDURE 47042-B.

0360000004K08 - IDENTIFY THE COMPONENTS THAT AFFECT THE RC SYSTEM OPERABILITY PER TECHNICAL SPECIFICATIONS SECTION 3.1.a.

60 . b

			RO Value	SRO Value			
KA	009	EK2.03	3.0	3.3*	Ref. Section	Ref. Page	Ref. Revision
Refs.	PLANT SPECIFIC BACKGROUND INFORMATION FOR KNPP IPEOP E-1			E-1	2.1.4	13(-15)	J

Objs. 2050000001K04 - Name the four categories of LOCA, giving an approximate size range for each category, and describe RCS pressure response for each.

2050000001K11 - Explain why core uncovery will occur in the case of energy-balance controlled SBLOCAs in the cold leg and not for those in the hot leg.

Contributor to Core Damage Sequence: Medium LOCA – 13%

DELETED DJP

61 . b

			RO Value	SRO Value			
KA	011	EK3.10	3.7	3.9	Ref. Section	Ref. Page	Ref. Revision
Refs.	Response to Pressurized Thermal Shock Condition			FR-P.1	Step 1 (RNO)	2	N
	PLANT SPECIFIC BACKGROUND INFORMATION FOR KNPP IPEOP FR-P.1			FR-P.1	4. Step 1	9B-9C	K

Objs. FRP0010501K04 - GIVEN AN IMMINENT PRESSURIZED THERMAL SHOCK CONDITION, EXPLAIN THE BASIS FOR ACTIONS TAKEN, PER FR-P.1 BACKGROUND DOCUMENT.

FRP0010501K05 - IDENTIFY the conditions which would require a procedural transition, while responding to an IMMINENT PRESSURIZED THERMAL SHOCK CONDITION, IN ACCORDANCE WITH FR-P.1.

Contributor to Core Damage Sequence: Large LOCA – 9%

62 . b

			RO Value	SRO Value	Ref. Section	Ref. Page	Ref. Revision
KA	017	AA2.01	3.0	3.5*			
Refs.	Abnormal Reactor Coolant Pump Operation			A-RC-36C	2.5, 4.5.1.a	2, 12-13	M
	Flow Diagram CVCS			OPXK100-35			AA

Objs. 0360190401A01 - RESPOND TO A #2 SEAL MALFUNCTION, GIVEN A DAMAGED #2 SEAL, IN ACCORDANCE WITH A-RC-36C.

0360340401A01- GIVEN A RXCP STANDPIPE HIGH/LOW ANNUNCIATOR, RESPOND IN ACCORDANCE WITH ALARM RESPONSE PROCEDURE 47015-I(L).

NRC NOTE: Although question is only on SRO exam (S), the NRC deems this question is NOT appropriate as SRO-only level question since procedure directs specific action. (Appropriate to a BOTH/RO level)

63 . c

			RO Value	SRO Value	Ref. Section	Ref. Page	Ref. Revision
KA	017	AK1.04	2.9	3.1*			
Refs.	System Description - Reactor Coolant (RC)			Number 36	3.5.4	36-22	A
	KNPP USAR				4.1.5	4.1-13 & 14	14

Objs. 0360000001K32 - DISCUSS THE DESIGN CHARACTERISTICS OF THE REACTOR COOLANT SYSTEM FOR ANY MODE OF OPERATION PER SYSTEM DESCRIPTION 36 AND USAR SECTIONS 4.0 AND 6.5

64 . a

			RO Value	SRO Value			
KA	022	AA2.04	2.9	3.8	Ref. Section	Ref. Page	Ref. Revision
Refs.	Control Room Alarm Response Procedures			RC-36	47042-F	1	B
	Reactor Coolant System Leak Rate Check			SP 36-082	Data Sheet 2, F	10	Y

Objs. 0350000001K06 - DISCUSS THE DESIGN CHARACTERISTICS OF THE CVC SYSTEM FOR ANY MODE OF OPERATION PER SYSTEM DESCRIPTION 35 AND USAR SECTIONS 9.2, 10A.5, AND 14.1.

0360000001K23 - EXPLAIN THE RESPONSE TO THE OPERATION OF ALL PRESSURIZER LEVEL CONTROL SYSTEM CONTROLS IN ACCORDANCE WITH LOGIC DIAGRAMS: E2023, E2025, E2039, E2041, XK100-154, K100-547, XK100-148.

65 . c

			RO Value	SRO Value			
KA	022	AK1.01	2.8	3.2*	Ref. Section	Ref. Page	Ref. Revision
Refs.	Abnormal Reactor Coolant Pump Operation			A-RC-36C	4.1.3	8-9	M
	PLANT SPECIFIC BACKGROUND INFORMATION FOR KNPP IPEOP ECA-0.1			ECA-0.1	3	3A-4	I

Objs. 0360000001K33 - DISCUSS THE DESIGN CHARACTERISTICS OF THE REACTOR COOLANT PUMP FOR ANY MODE OF OPERATION PER SYSTEM DESCRIPTION 36 AND USAR SECTIONS 4.1 AND 4.2.

0360180401A01 - RESPOND TO ABNORMAL RXCP OPERATION DUE TO A LOSS OF SEAL INJECTION, IN ACCORDANCE WITH A-RC-36C.

E000080501K04 - GIVEN A RECOVERY FROM A LOSS OF ALL AC POWER WITHOUT SI REQUIRED, EXPLAIN THE BASIS FOR ACTIONS TAKEN PER ECA-0.1 BACKGROUND DOCUMENT.

66 . C

			RO Value	SRO Value			
KA	024	AA1.26	3.3	3.3	Ref. Section	Ref. Page	Ref. Revision
Refs.	Emergency Boration			E-CVC-35	4.3.1	3	P
	KNPP Reactor Data Manual				RD 6.7, 6.6, 2.2.6	9-10 (2.2.6)	May 19, 2000 & 11/7/96
	Operator Aid 89-13				Boric acid Tank 1A Level Calibration		5-1-89

Objs. 0350010501A01 - PERFORM AN EMERGENCY BORATION WHILE RESPONDING TO AN UNEXPECTED POSITIVE REACTIVITY CONDITION IN ACCORDANCE WITH E-CVC-35.

67 . C

			RO Value	SRO Value			
KA	024	AK3.02	4.2	4.4	Ref. Section	Ref. Page	Ref. Revision
Refs.	Response to Nuclear Power Generation/ATWS			FR-S.1	Step 6.g RNO	5	M
	PLANT SPECIFIC BACKGROUND INFORMATION FOR KNPP IPEOP ECA-0.0			FR-S.1	4. Step 6	40	J

Objs. FRS0020501K04 - GIVEN A NUCLEAR GENERATION/ATWS CONDITION, EXPLAIN THE BASIS FOR ACTIONS TAKEN, PER FR-S.1 BACKGROUND DOCUMENT.

68 . d		RO Value	SRO Value			
KA	025 AK2.02	3.2*	3.2	Ref. Section	Ref. Page	Ref. Revision
Refs.	RHR Operation at a Reduced Inventory Condition		N-RHR-34C	Step 4.8, 4.9, Reference Sheet	5, 14	1

Objs. 0340000001K02 - DISCUSS THE DESIGN CHARACTERISTICS OF THE RESIDUAL HEAT REMOVAL SYSTEM, FOR ANY MODE OF OPERATION, PER SYSTEM DESCRIPTION 34 AND USAR SECTION 6.1, 6.2, 6.5, AND 9.3.

0340010101A01 - GIVEN A REDUCED INVENTORY CONDITION, OPERATE THE RHR SYSTEM IN ACCORDANCE WITH N-RHR-34C.

69 . b		RO Value	SRO Value			
KA	026 AA1.07	2.9	3.0	Ref. Section	Ref. Page	Ref. Revision
Refs.	LOSS OF ALL AC POWER RECOVERY WITHOUT SI REQUIRED		ECA-0.1	steps 1.b & 3.c	2, 4	L
	PLANT SPECIFIC BACKGROUND INFORMATION FOR KNPP IPEOP ECA-0.0		ECA-0.0 (& ECA-0.1)	4. - step 18 (4. - step 1)	59 (9)	N (I)

Objs. E000080501K04 - GIVEN A RECOVERY FROM A LOSS OF ALL AC POWER WITHOUT SI REQUIRED, EXPLAIN THE BASIS FOR ACTIONS TAKEN PER ECA-0.1 BACKGROUND DOCUMENT.

E000070501K05 - GIVEN A LOSS OF ALL AC POWER, EXPLAIN THE BASIS FOR ACTIONS TAKEN, PER ECA-0.0 BACKGROUND DOCUMENT.

Contributor to Core Damage Sequence: Loss of Offsite Power – 38%, Loss of AC Bus – 2%

70 . a

			RO Value	SRO Value			
KA	028	AA2.08	3.1	3.5	Ref. Section	Ref. Page	Ref. Revision
Refs.	System Description - Reactor Coolant (RC)			Number 36	3.8.3	36-37	A
	Logic Diagram - Rod Control & Rod Blocks			XK100-151			4
	Logic Diagram - PRZR Pressure & Level Control			XK100-154			4

Objs. 0360000001K23 - EXPLAIN THE RESPONSE TO THE OPERATION OF ALL PRESSURIZER LEVEL CONTROL SYSTEM CONTROLS IN ACCORDANCE WITH LOGIC DIAGRAMS: E2023, E2025, E2039, E2041, XK100-154, K100-547, XK100-148.

0360000001K36 - DISCUSS THE DESIGN CHARACTERISTICS OF THE PRESSURIZER LEVEL CONTROL SYSTEM FOR ANY MODE OF OPERATION PER SYSTEM DESCRIPTION 36 AND USAR SECTIONS 7.2 AND 7.3.

NRC NOTE: Although question is only on SRO exam (S), the NRC deems this question is NOT appropriate as SRO-only level question since procedure directs specific action. (Appropriate to a BOTH/RO level)

71 . a

			RO Value	SRO Value			
KA	028	2.4.10	3.0	3.1	Ref. Section	Ref. Page	Ref. Revision
Refs.	Control Room Alarm Response Procedures			RC-36	47042-F		B
	Technical Specifications for Kewaunee Nuclear Plant				3.5, Table TS 3.5.2	TS 3.5-1, 2	Amend 105, 94
	Integrate Logic - RCS			E2039			AA

Objs. 0360000001K23 - EXPLAIN THE RESPONSE TO THE OPERATION OF ALL PRESSURIZER LEVEL CONTROL SYSTEM CONTROLS IN ACCORDANCE WITH LOGIC DIAGRAMS: E2023, E2025, E2039, E2041, XK100-154, K100-547, XK100-148.

0360220401A01 - GIVEN A PRESSURIZER LVL LETDOWN ISOL & HEATERS OFF ANNUNCIATOR, RESPOND IN ACCORDANCE WITH ALARM RESPONSE PROCEDURE 47042-F.

72 . d

			RO Value	SRO Value			
KA	029	AA1.15	4.1	3.9	Ref. Section	Ref. Page	Ref. Revision
Refs.	KNPP USAR				14.11	14.1-35	15
	KNPP USAR				6.6.2	6.6-2	14
	Integrate Logic - S/G Trip Signals E2802						L

Objs. 05B0000001K01 - DISCUSS THE DESIGN CHARACTERISTICS OF THE AUXILIARY FEEDWATER SYSTEM FOR ANY MODE OF OPERATION PER SYSTEM DESCRIPTION 05B AND USAR SECTIONS 1.2.8.e, 6.6, AND 10A.3.5.

0470000001K05 - Discuss the design characteristics of the Reactor Control & Protection system for any mode of operation per System Description 47 and USAR sections 1.3, 1.8, 4.1, 4.2, 7.2, 7.5.3, 14.1 and 14.2.

0490000004K07 - LIST THE AUTOMATIC TRIPS FOR THE ROD DRIVE MG SETS PER A-CRD-49.

73 . a

			RO Value	SRO Value			
KA	029	AK2.06	2.9*	3.1*	Ref. Section	Ref. Page	Ref. Revision
Refs.	System Description - Reactor Protection & RCS Temperature Instruments (RCP)			Number 47	3.13.4	47-39	Orig
	Logic diagram - Rector Trip Signals				XK100-144		5C

Objs. 0470000001K02 - Explain the response to the operation of all Reactor Control & Protection system controls in accordance with Logic Diagrams/Schematics...

0470000001K05 - Discuss the design characteristics of the Reactor Control & Protection system for any mode of operation per System Description 47 and USAR sections 1.3, 1.8, 4.1, 4.2, 7.2, 7.5.3, 14.1 and 14.2.

74 . a

			RO Value	SRO Value			
KA	032	2.1.12	2.9	4.0	Ref. Section	Ref. Page	Ref. Revision
Refs.	Technical Specifications for Kewaunee Nuclear Plant				3.5.c, Table TS 3.5.2 #4	TS 3.5-1, 1 & 4	Amend. 137
	Abnormal Nuclear Instrumentation				A-NI-48	4.1.2.b	3 R
	NI Logics			E-2051-1 & -2			M, N

Objs. 0480000001K02 - DISCUSS THE DESIGN CHARACTERISTICS OF THE NUCLEAR INSTRUMENTATION SYSTEM, FOR ANY MODE OF OPERATION, PER SYSTEM DESCRIPTION 048 AND USAR SECTIONS 7.2 AND 7.4.

0480000001K05 - EXPLAIN THE RESPONSE TO THE OPERATION OF ALL NUCLEAR INSTRUMENTATION SYSTEM CONTROLS IN ACCORDANCE WITH THE FOLLOWING LOGIC DIAGRAMS/SCHEMATICS: E2051-1, E2051-2, XK100-692, XK100-693, XK100-694, XK100-695.

0480050401A01- RESPOND TO THE FAILURE OF SOURCE RANGE INSTRUMENTATION, GIVEN A SINGLE CHANNEL FAILURE, IN ACCORDANCE WITH A-NI-48.

75 . c

			RO Value	SRO Value			
KA	037	2.1.33	3.4	4.0	Ref. Section	Ref. Page	Ref. Revision
Refs.	Reactor Coolant Leak			A-RC-36D	4.8.2.e	9	AB
	Primary-To-Secondary Leak Rate Data			RCC-088	4.7	6	L

Objs. 0360040401A01 - GIVEN A PRIMARY TO SECONDARY LEAK, RESPOND IN ACCORDANCE WITH A-RC-36D.

1190110301K03 - Discuss RCS leakage criteria, including setpoints and bases, in accordance with T.S. 3.1.d.

76 . b

			RO Value	SRO Value			
KA	038	EK1.02	3.2	3.5	Ref. Section	Ref. Page	Ref. Revision
Refs.	Steam Generator Tube Rupture			E-3	Step 19	14	R
	PLANT SPECIFIC BACKGROUND INFORMATION FOR KNPP IPEOP E-3			E-3	3.1.3	15	N

Objs. 0360000001K32 - DISCUSS THE DESIGN CHARACTERISTICS OF THE REACTOR COOLANT SYSTEM FOR ANY MODE OF OPERATION PER SYSTEM DESCRIPTION 36 AND USAR SECTIONS 4.0 AND 6.5.

0360000004K03 - Explain the basic principles of operation for the RC System and the major components and equipment per System Description 36.

E030010501K04 - GIVEN A STEAM GENERATOR TUBE RUPTURE, EXPLAIN THE BASIS FOR ACTIONS TAKEN, PER E-3 BACKGROUND DOCUMENT

77 . b

			RO Value	SRO Value			
KA	040	AK1.06	3.7	3.8	Ref. Section	Ref. Page	Ref. Revision
Refs.	System Description - Feedwater (FW)			Number 5A	3.8	5A-12 & 13	B
	Instrument Block Diagram			XK-100-554	B-3		1V
	Integrated Logic - Feedwater System			E-2006	5		S

Objs. N030010101A01 - PERFORM A POWER INCREASE FROM 15% TO 100% DURING A PLANT STARTUP IN ACCORDANCE WITH N-0-03.

0060000001K08 - Discuss the design characteristics of the Steam Generator and Steam Generator Water Level Control System, for any mode of operation, per System Description 05A and USAR Sections 4.3.1, 7.2.3, and 7.3.3.

78 . b		RO Value	SRO Value			
KA	040 AK2.02	2.6*	2.6	Ref. Section	Ref. Page	Ref. Revision
Refs.	Control Room Alarm Response Procedures		MS-06	47062-J		Orig
	Integrated Logic - Main Steam & Steam Dump SYS		E-1627			AF

Objs. 0060000001K02 - Discuss the design characteristics of the main steam and steam dump system, for any mode of operation, per System Description 06 and/or USAR sections...

0060000001K03 - EXPLAIN THE RESPONSE TO THE OPERATION OF ALL MAIN STEAM SYSTEM CONTROLS IN ACCORDANCE WITH LOGIC DIAGRAM/SCHEMATIC E1627

79 . b		RO Value	SRO Value			
KA	051 AA2.02	3.9	4.1	Ref. Section	Ref. Page	Ref. Revision
Refs.	Loss of Condenser Vacuum		E-AR-09	Step 4.2, Fig. 1	2, 5	L
						A

Objs. 0090010501A04 - GIVEN A LOSS OF CONDENSER VACUUM, RESPOND IN ACCORDANCE WITH E-AR-09.

80 . c

			RO Value	SRO Value			
KA	054	2.4.50	3.3	3.3	Ref. Section	Ref. Page	Ref. Revision
Refs.	Abnormal Feedwater System Operation			A-FW-05A	4.4	4	O
	Control Room Alarm Response Procedures			FW-05A	47064-D		Orig

Objs. 05A0000001K04 - EXPLAIN THE RESPONSE TO THE OPERATION OF ALL MAIN FEEDWATER SYSTEM CONTROLS IN ACCORDANCE WITH LOGIC DIAGRAMS/SCHEMATICS: E1624, E1625, E2006.

05A0020401A01 - GIVEN A FEEDWATER PUMP TRIP, RESPOND IN ACCORDANCE WITH A-FW-05A.

05A0120401A01 - GIVEN A S/G LEVEL LOW LOW ANNUNCIATOR, RESPOND IN ACCORDANCE WITH ALARM RESPONSE PROCEDURE 47064-A(D).

81 . d

			RO Value	SRO Value			
KA	055	EA1.07	4.3	4.5	Ref. Section	Ref. Page	Ref. Revision
Refs.	Restoration of Off-site Power			A-SUB-59	4.2.2, 4.3	5-6	B
	4160V AC Supply and Distribution System Operation			N-EHV-39	2.7	1	N
	System Description - Substation Electrical (SUB)			Number 59	1.1, 4.1	59-1-3, 59-54	Orig

Objs. 0590010401A02 - GIVEN A LOSS OF OFF-SITE POWER, RESTORE POWER THROUGH F-84 AND Q-303 IN ACCORDANCE WITH A-SUB-59.

0590010401A03 - GIVEN A LOSS OF OFF-SITE POWER, RESTORE NORMAL PLANT ELECTRICAL LINEUPS IN ACCORDANCE WITH A-SUB-59.

0590000004K04 - Draw a one-line diagram of the Substation System showing transformers, breakers, disconnects, transmission line destination, and operating voltages per System Description 59.

82 . C

			RO Value	SRO Value			
KA	056	AA1.01	4.0*	3.8*	Ref. Section	Ref. Page	Ref. Revision
Refs.	Fire In Alternate Fire Zone			E-0-06	18	17	M
	System Description - Main Steam Number 6 & Steam Dump (MS)				3.4	6-10, 11	B

Objs. 0060000001K02 - DISCUSS THE DESIGN CHARACTERISTICS OF THE MAIN STEAM AND STEAM DUMP SYSTEM, FOR ANY MODE OF OPERATION, PER SYSTEM DESCRIPTION 06 AND/OR USAR SECTIONS.

0060000004K03 - Explain the basic principles of operation for the MS System and the major components and equipment per System Description 06.

E060010501A01 - GIVEN A FIRE IN AN ALTERNATE FIRE ZONE, RESPOND IN ACCORDANCE WITH E-0-06.

83 . C

			RO Value	SRO Value			
KA	057	2.4.46	3.5	3.6	Ref. Section	Ref. Page	Ref. Revision
Refs.	Control Room Alarm Response Procedures			EDC-38	47102-D		Orig
	DC Supply and Distribution System			A-EDC-38	4.4	3-4	U
	Circuit Diagram - DC Aux. & Emergency AC			E233			AP

Objs. 0380000001K02 - EXPLAIN THE RESPONSE TO THE OPERATION OF ALL DC AND EMERGENCY AC SUPPLY SYSTEM CONTROLS IN ACCORDANCE WITH LOGIC DIAGRAMS E233 AND E3626.

0380000001K06 - DISCUSS THE DESIGN CHARACTERISTICS OF THE DC AND EMERGENCY AC SUPPLY SYSTEM FOR ANY MODE OF OPERATION PER SYSTEM DESCRIPTION 38 AND USAR SECTIONS 1.8 AND 8.0.

0380040401A01 - GIVEN AN INSTRUMENT BUS INVERTER TROUBLE CONDITION, RESPOND IN ACCORDANCE WITH A-EDC-38.

84 . c

			RO Value	SRO Value	Ref. Section	Ref. Page	Ref. Revision
KA	062	AA2.03	2.6	2.9			
Refs.	Flow Diagram - SW			OP M-547			R
	Technical Specifications for Kewaunee Nuclear Plant				3.3.c.3	TS 3.3-4	Amend 116

Objs. 0020060401A04 - Respond to CNTMT Fan Coil SW Header Leakage in accordance with A-SW-02.

0020000004K08 - Identify the components that affect the SW System operability per Technical Specifications section 3.3.

PRA SYSTEM Importance: Service Water – 3

85 . a

			RO Value	SRO Value	Ref. Section	Ref. Page	Ref. Revision
KA	065	AA1.03	2.9	3.1			
Refs.	Loss of Instrument Air			E-AS-01	Steps 4.23.15 & 16	10	M
	Integrated Logic - ICS System			E-2012			K
	Flow Diagram - ICS System			OP M-207			AK

Objs. 0010010504A02 - Upon restoration of Instrument Air, align affected Systems/Components per E-AS-01.

0230000001K04 - EXPLAIN THE RESPONSE TO THE OPERATION OF ALL INTERNAL CONTAINMENT SPRAY SYSTEM CONTROLS IN ACCORDANCE WITH LOGIC DIAGRAMS E1604 AND E2012.

86 . b

			RO Value	SRO Value	Ref. Section	Ref. Page	Ref. Revision
KA	074	EK3.05	4.2	4.5			
Refs.	Response to Inadequate Core Cooling			FR-C.1	Steps 1 & 2	2	L
	PLANT SPECIFIC BACKGROUND INFORMATION FOR KNPP IPEOP FR-C.1			FR-C.1	2.	2	H

Objs. FRC0010501K01 - EXPLAIN THE PURPOSE OF FR-C.1, RESPONSE TO AN INADEQUATE CORE COOLING CONDITION.

FRC0010501K03 - LIST THE MAJOR STEPS IN THE HIGH LEVEL ACTION SUMMARY FOR FR-C.1, RESPONSE TO AN INADEQUATE CORE COOLING CONDITION, PER THE IPEOP BACKGROUND DOCUMENT.

Contributor to Core Damage Sequence: Medium LOCA – 13%

87 . c

			RO Value	SRO Value	Ref. Section	Ref. Page	Ref. Revision
KA	076	AK2.01	2.6	3.0			
Refs.	System Description -Radiation Monitoring			Number 45	3.4	45-27	Orig
	Integrated Logic - Radiation Monitoring			E2021			T
	Flow Diagram CVCS			OP XK100-36			AS

Objs. 0450000001K03 - DISCUSS THE DESIGN CHARACTERISTICS OF THE RADIATION MONITORING SYSTEM FOR ANY MODE OF OPERATION PER SYSTEM DESCRIPTION 45 AND USAR SECTIONS 1.3.3, 1.6.6, 1.8, 6.5, AND 11.2.3.

88 . c

			RO Value	SRO Value			
KA	E02	EK1.1	3.2	3.8	Ref. Section	Ref. Page	Ref. Revision
Refs.	SI Termination			ES-1.1	steps 5-8	3-5	M

Objs. E010020501A01 - GIVEN CONDITIONS TO TERMINATE SAFETY INJECTION, PERFORM THE ACTIONS OF ES-1.1.

E010020501K03 - LIST THE MAJOR STEPS IN THE HIGH LEVEL ACTION SUMMARY FOR ES-1.1, SI TERMINATION, PER THE IPEOP BACKGROUND DOCUMENT.

89 . a

			RO Value	SRO Value			
KA	E03	EA2.1	3.4	4.2	Ref. Section	Ref. Page	Ref. Revision
Refs.	Post LOCA Cooldown And Depressurization			ES-1.2	Step 24	15	M
	Quick Reference Foldout Section E-1			E-1 QRF	2	1	H
	User's Guide For IPEOPs			UG-0	m	4	C

Objs. E010010501K07 - LIST THE SI REINITIATION CRITERIA, GIVEN A LOSS OF REACTOR OR SECONDARY COOLANT, IN ACCORDANCE WITH THE E-1 QRF.

E010010501A02 - WHILE RESPONDING TO A LOSS OF REACTOR OR SECONDARY COOLANT, IMPLEMENT THE E-1 QRF

90 . d

			RO Value	SRO Value			
KA	E03	EK2.1	3.6	4.0	Ref. Section	Ref. Page	Ref. Revision
Refs.	System Description - Safety Injection (SI)			Number 33	3.8	33-20	C
	Integrated Logic - SIS			E-2032			V

Objs. 0330000001K04 - EXPLAIN THE RESPONSE TO THE OPERATION OF ALL SAFETY INJECTION SYSTEM CONTROLS IN ACCORDANCE WITH LOGIC DIAGRAMS/SCHEMATICS E1635, E2032, E2033, E2034, E2035, XK100-148 AND XK100-150.

Contributor to Core Damage Sequence: Large LOCA – 9%

91 . c

			RO Value	SRO Value			
KA	E04	EK2.2	3.8	4.0	Ref. Section	Ref. Page	Ref. Revision
Refs.	Flow Diagram CVCS			OP XK100-35			AA
	Integrated Logic - CVCS			E2023	LD-4B		W

Objs. E010050501K03 - Explain the applicability and entry conditions of ECA-1.1, Loss of Emergency Coolant Recirculation.

0350000001K02 - EXPLAIN THE RESPONSE TO THE OPERATION OF ALL CHEMICAL AND VOLUME CONTROL - CHARGING AND LETDOWN SYSTEM CONTROLS IN ACCORDANCE WITH LOGIC DIAGRAMS E2020, E2023 THROUGH E2028, E2030, E3000, AND E2039.

0360000001K32 - DISCUSS THE DESIGN CHARACTERISTICS OF THE REACTOR COOLANT SYSTEM FOR ANY MODE OF OPERATION PER SYSTEM DESCRIPTION 36 AND USAR SECTIONS 4.0 AND 6.5.

92 . d

			RO Value	SRO Value			
KA	E04	EK3.2	3.4	4.0	Ref. Section	Ref. Page	Ref. Revision
Refs.	Loss of Emergency Coolant Recirculation			ECA-1.1	Step 21	10	K
	PLANT SPECIFIC BACKGROUND INFORMATION FOR KNPP IPEOP ECA-1.1			ECA-1.1	3.1.4	5	G

Objs. E010050501K02 - LIST THE MAJOR STEPS IN THE HIGH LEVEL ACTION SUMMARY OF ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION, PER THE IPEOP BACKGROUND DOCUMENT.

E010050501K04 - GIVEN A LOSS OF EMERGENCY COOLANT RECIRCULATION, EXPLAIN THE BASIS FOR ACTIONS TAKEN, PER ECA-1.1 BACKGROUND DOCUMENT.

Contributor to Core Damage Sequence: Medium LOCA – 13%

93 . b

			RO Value	SRO Value			
KA	E08	2.4.45	3.3	3.6	Ref. Section	Ref. Page	Ref. Revision
Refs.	Transfer to Containment Sump Recirculation			ES-1.3	2.1.c	1	Q
	Transfer to Containment Sump Recirculation			ES-1.3	CAUTION step 1	2	Q
	PLANT SPECIFIC BACKGROUND INFORMATION FOR KNPP IPEOP ES-1.3			ES-1.3	4 - Step 1	6	N

Objs. E010010501K10 - LIST THE SUMP RECIRCULATION SWITCHOVER CRITERION, GIVEN A LOSS OF REACTOR OR SECONDARY COOLANT, IN ACCORDANCE WITH THE E-1 QRF.

E010040501K02 - Explain the applicability and entry conditions of ES-1.3, Transfer to Containment Sump Recirculation.

E010040501K04 - GIVEN A TRANSFER TO CONTAINMENT SUMP RECIRCULATION, EXPLAIN THE BASIS FOR ACTIONS TAKEN, PER ES-1.3 BACKGROUND DOCUMENT.

94 . b

			RO Value	SRO Value			
KA	E08	EK1.2	3.4	4.0	Ref. Section	Ref. Page	Ref. Revision
Refs.	Response To Imminent Pressurized Thermal Shock Condition			FR-P.1	Step 24	11	N
	PLANT SPECIFIC BACKGROUND INFORMATION FOR KNPP IPEOP FR-P.1			FR-P.1	4, Step 24	39-40	K

Objs. FRP0010501A01 - GIVEN AN IMMINENT PRESSURIZED THERMAL SHOCK CONDITION, RESPOND IN ACCORDANCE WITH FR-P.1.

FRP0010501K04 - GIVEN AN IMMINENT PRESSURIZED THERMAL SHOCK CONDITION, EXPLAIN THE BASIS FOR ACTIONS TAKEN, PER FR-P.1 BACKGROUND DOCUMENT.

95 . c

			RO Value	SRO Value			
KA	E09	EA2.1	3.1	3.8	Ref. Section	Ref. Page	Ref. Revision
Refs.	Natural Circulation Cooldown			ES-0.2	Figure ES-0.2-2	12	N

Objs. E000050501A01 - WHEN REQUIRED TO COOLDOWN WITH NATURAL CIRCULATION, PERFORM THE ACTIONS IN ACCORDANCE WITH ES-0.2.

E000050501K02 - GIVEN A NATURAL CIRCULATION COOLDOWN, EXPLAIN THE BASIS FOR ACTIONS TAKEN, PER ES-0.2 BACKGROUND DOCUMENT.

Contributor to Core Damage Sequence: Loss of Offsite Power – 38%

96 . a

			RO Value	SRO Value			
KA	E10	2.4.7	3.1	3.8	Ref. Section	Ref. Page	Ref. Revision
Refs.	PLANT SPECIFIC BACKGROUND INFORMATION FOR KNPP IPEOP ES-0.3			ES-0.3	3.	3	J

Objs. E000060501K03 - LIST THE MAJOR STEPS IN THE HIGH LEVEL ACTION SUMMARY FOR ES-0.3, NATURAL CIRCULATION COOLDOWN WITH STEAM VOID IN VESSEL, PER THE IPEOP BACKGROUND DOCUMENT.

97 . c			RO Value	SRO Value			
KA	E11	EK3.2	3.5	4.0	Ref. Section	Ref. Page	Ref. Revision
Refs.	Loss of Emergency Coolant Recirculation			ECA-1.1	Figure ECA-1.1-1	17	K
	USAR				Figure 14.2.4-1 (SI Pumps Flow)		15

Objs. E010050501A01 - GIVEN A LOSS OF EMERGENCY COOLANT RECIRCULATION CONDITION, RESPOND IN ACCORDANCE WITH ECA-1.1.

98 . b			RO Value	SRO Value			
KA	E12	2.4.30	2.2	3.6	Ref. Section	Ref. Page	Ref. Revision
Refs.	Reportability Determinations			GNP 11.8.4	Figure GNP 11-8-4-1	6, 9, 10	A

Objs. 1190210302A01 - Evaluate internal and external problems for reportability in accordance with GNP 11.8.4.

1190190302K01 - Given a System or Component, recall the LCO, and explain the bases per Technical Specifications.

99 . a

			RO Value	SRO Value			
KA	E16	EK1.3	3.0	3.3	Ref. Section	Ref. Page	Ref. Revision
Refs.	Abnormal Radiation Monitoring System			A-RM-45	3.3 & 3.4	2-3	AC
	System Description - Radiation Monitoring (RM)			Number 45	1.3 Table	45-7	Orig
	Integrated Logics - Process Radiation Monitors			E-3748	Table 1 (8)		B

Objs. 0450000001K02 - EXPLAIN THE RESPONSE TO THE OPERATION OF ALL RADIATION MONITORING SYSTEM CONTROLS IN ACCORDANCE WITH THE FOLLOWING LOGIC DIAGRAMS: E2013, E2018, E2019, E2021, E2951, E3745 and E3748.

0450110401K01 - LIST THE IMMEDIATE OPERATOR ACTIONS, GIVEN AN R-13 RADIATION INDICATION HIGH CONDITION IN ACCORDANCE WITH A-RM-45.

100 . b

			RO Value	SRO Value			
KA	E16	EK3.3	3.0	3.0	Ref. Section	Ref. Page	Ref. Revision
Refs.	Response to High Containment Radiation Level			FR-Z.3	Step 2	2	F
	PLANT SPECIFIC BACKGROUND INFORMATION FOR KNPP IPEOP FR-Z.3			FR-Z.3	4, Step 3	6A	C

Objs. FRZ0030501K01- GIVEN A CONTAINMENT HIGH RADIATION LEVEL CONDITION, EXPLAIN THE BASIS FOR ACTIONS TAKEN PER FR-Z.3 BACKGROUND DOCUMENT.