REQUEST FOR SCANNING SERVICES

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REQUESTER: Bev Michael, Operator Licensing, DRS ext. 2-4640 Thx, Bev	DATE OF REQUEST February 12, 2002
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Nuclear Regulatory Commission Senior Reactor Operator Licensing Examination

McGuire Nuclear Station

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Bank Question: 51 ***Answer: C***

1 Pt(s)

A large break LOCA is in progress and the operators are responding in E-1 (Reactor Trip or Safety Injection). Given the following conditions:

- ND pump 1A is tagged out of service for maintenance.
- Containment pressure is 14 psig.
- FWST level is below the swap over setpoint.

When shifting to cold leg recirc using ES-1.3 (Transfer to Cold Leg Recirc), valve 1NI-184B (RB Sump to Train 1B ND & NS) fails to open. The operators implement ECA-1.1 (Loss of Emergency Coolant Recirculation).

FR-Z.1 (Response to High Containment Pressure) requires both NS pumps to be in operation. ECA-1.1 limits the operators to only one NS pump in step 11. Which of these two procedures takes priority under these conditions and what is the basis for this requirement?

A. FR-Z.1 takes priority because a total loss of ND causes the NS system to become relatively more important to reduce containment pressure.

B. FR-Z.1 takes priority because it was implemented in response to a red path and FRPs always have priority over ECA procedures.

C. ECA-1.1 takes priority because it conserves FWST water level as long as possible for injection while providing sufficient NS flow to mitigate containment pressure.

D. ECA-1.1 takes priority because ECA procedures always have priority over FRPs.

Distracter Analysis:

A. **Incorrect**: ECA-1.1 takes priority over FR-Z.1
Plausible: Although a loss of ND and containment sump recirc causes a loss of the containment heat sink, the supply for NS comes from the FWST which will be drawn down until containment sump recirculation can be established.

B. **Incorrect**: ECA-1.1 takes priority over FR-Z.1
Plausible: FRPs normally take priority over most EOPs

C. **Correct answer**

D. **Incorrect**: ECAs do not always have priority over FRPs.

Plausible: Some ECAs take priority e.g. ECA-0.0 has priority over FRPs in that F-0 is not applicable until transition out of ECA-0.0.

Bank Question: 60 ***Answer: A***

1 Pt(s) Unit 2 was operating at 100% power when an electrical fire started inside the auxiliary building cable spreading room. What type of fire suppression system is installed inside the cable spreading area and what are the hazards to personnel if they enter this room?

A. **A manual deluge (Mulsifyre) System is installed. An electrical shock hazard exists due to the use of water to combat an electrical fire.**

B. **An automatic sprinkler system is installed. An electrical shock hazard exists due to the use of water to combat an electrical fire.**

C. **An automatic Halon system is installed. An asphyxiation hazard exists due to the presence of Halon gas.**

D. **A manual Cardox system is installed. An asphyxiation hazard exists due to the presence of carbon dioxide gas.**

Distracter Analysis:

A. **Correct Answer**:

B. **Incorrect**: A manual deluge Mulsifyre system is installed
Plausible: an electrical shock hazard exists

C. **Incorrect**: A manual deluge Mulsifyre system is installed
Plausible: Halon gas is generally used in areas in which electrical fires are the predominant risk because it does not create a shock hazard

D. **Incorrect**: A manual deluge Mulsifyre system is installed
Plausible: Cardox gas is a personnel hazard – although all the CARDOX systems have been replaced with HALON, the pull switches still say CARDOX in some areas (like the diesel generators)

Bank Question: 82 ***Answer: D***

1 Pt(s)

Unit 2 is recovering from a loss of 120 VAC instrument bus 2EKVA due to the loss of the 2EVIA static inverter. 2EKVA has been reenergized from the alternate supply. After repairs to inverter 2EVIA are completed, the operator is directed to restore the 2EKVA bus to the normal line up.

Which one of the following actions is necessary to restore the electrical lineup to a normal operating configuration after tags are cleared?

A. **Manually transfer bus power from static inverter 2EVIB back to static inverter 2EVIA.**

B. **Enable the automatic transfer of power from static inverter 2EVIB back to 2EVIA.**

C. **Enable the automatic transfer of power from regulated power center 2KRP back to static inverter 2EVIA.**

D. **Manually transfer power from regulated load center 2KRP back to static inverter 2EVIA.**

Distracter Analysis:

A. **Incorrect**: Power is not transferred to 2EVIB – this is the normal power supply to 2EKVB
Plausible: If the candidate thinks that 2EKIB can be used to supply 2EKVA

B. **Incorrect**: There is no automatic transfer associated with these static inverters
Plausible: There are automatic bus transfers for some of the 120 VAC power supply breakers

C. **Incorrect**: There is no automatic transfer between 2KRP and 2EVIA
Plausible: 2KRP is the correct alternate supply for 2EKVB if 2EVIA is not operating and there is an auto transfer switch between the normal and alternate power supplies for 2KRP.

D. **Correct answer**

Bank Question: 97 ***Answer: B***

1 Pt(s)

Unit 1 is in the process of preparing to conduct a plant cooldown in Mode 3 in preparation for a refueling period. The OSM denied a request from maintenance to tag shut 1ND-30A (TRAIN A ND TO HOT LEG ISOL) for valve stroke time testing.

What is the reason for his decision?

A. **Although this would be permitted in mode 3, closing 1ND-30A would cause one ND train to be inoperable and is prohibited in Mode 4.**

B. **This action would isolate the cross tie between the ND trains which is assumed to be open in the FSAR ensuring one ND pump can inject into all four NC loops.**

C. **1ND-15B is interlocked with 1NI-136B (B NI PUMP SUCTION FROM ND) to provide an alternate boration path. Closure of 1ND-30A will defeat this interlock.**

D. **1ND-30A is interlocked with 1ND-58A (TRAIN A ND TO NV & NI PUMPS) and this will defeat the interlock and prevent 1ND-58A from opening and establishing an alternate boration path from the NV system**

Distracter Analysis:

A. **Incorrect**: permitted in mode 4
Plausible: based on transposition of operating requirements

B. **Correct Answer**:

C. **Incorrect**: 1ND-15B is not interlocked with 1NI-136B
Plausible: based on misunderstanding of the 1NI-136B interlock

D. **Incorrect**: 1ND-30A is not interlocked with 1ND-58A.
Plausible: based on misunderstanding of the 1ND-58A interlock

Bank Question: 120 ***Answer: C***

1 Pt(s) Unit 1 was operating at 100% power. Given the following motor driven auxiliary feedwater pump operating parameters:

	0200	0210	0220	0230
Discharge Pressure (ft water)	3325	3325	3010	2950
Suction Pressure (ft water)	75	75	75	75
Pump flow rate (gpm)	420	480	520	560

What is the onset (earliest time) of pump cavitation conditions?

REFERENCES PROVIDED
Curve 8.4 of enclosure 4.3 to OP/1/A/6100/22

A. **0200**

B. **0210**

C. **0220**

D. **0230**

Distracter Analysis:

A. **Incorrect**: - cavitation has not yet occurred: 3325 - 75 = 3250 in
Plausible: - below pump characteristic curve but above NPSH requirement

B. **Incorrect:** - cavitation has not yet occurred: 3325-75 = 3250
Plausible: - point is above the pump characteristic curve – but still below NPSH curve

C. **Correct answer**: below NPSH curve 3010 – 75 = 2935 in

D. **Incorrect**: cavitation has already occurred
Plausible: - – if the candidate does not remember to subtract the suction pressure from the discharge pressure: 3010 from answer C is above the NPSH curve

Bank Question: 125 ***Answer: C***

1 Pt(s)

A worker is preparing to enter a high radiation area to work on a valve in the reactor building. During the pre-job briefing, RP states that the expected whole body radiation level are as follows:

- Dose rate in the center of the room 20 ft away = 200 mrem/hr
- Dose rate 18 inches from valve = 700 mrem/hr
- Contact reading = 1100 mrem/hr

How should the area around the valve be classified?

A. **The room is a radiation area; the valve is a hot spot**

B. **The room is a high radiation area; valve is NOT a hot spot**

C. **The room is a high radiation area; the valve is a hot spot**

D. **The room is an extra high radiation area; the valve is NOT a hot spot**

Distracter Analysis: A hot spot is an area where the dose rate on contact is > 5x general area radiation but > 100 mrem/hr. In this case 5 x 200 = 1000 mrem/hr < 1100 mrem/hr on contact.

A. **Incorrect**: 200 mrem/hr general area dose rate > 100 mrem/hr = high radiation area
Plausible: if the candidate does not know that the lower limit for a high radiation area is 100 mrem/hr – and the valve is a hot spot

B. **Incorrect**: The valve is a hot spot
Plausible: the room is a high radiation area – if the candidate thinks that the definition of a hot spot is > 5x general area dose rate when measured 18 inches from the contact reading

C. **Correct Answer**:

D. **Incorrect:** The room is not an extra high radiation area
Plausible: if the candidate thinks that the definition of a hot spot is > 5x general area dose rate when measured 18 inches from the contact reading

Bank Question 152 ***Answer: C***

1 Pt(s) The operators are conducting a reactor startup.

Given the following indications on the source range (SR) and intermediate range (IR) excore nuclear instruments:

Time	0200	0205	0210	0215
SR "A" (cps)	$1.5x10^4$	$2.5x10^4$	$2.8x10^4$	$1.0x10^5$
SR "B" (cps)	$1.4x10^4$	$2.3x10^4$	$2.7x10^4$	$9.8x10^4$
IR "A" (amps)	$7.6x10^{-11}$	$1.1x10^{-10}$	$1.5x10^{-10}$	$7.0x10^{-10}$
IR "B" (amps)	$7.9x10^{-11}$	$9.0x10^{-11}$	$1.1x10^{-10}$	$7.5x10^{-10}$

What is the earliest time that the operators should block the source range nuclear instruments?

A. 0200

B. 0205

C. 0210

D. 0215

Distracter Analysis:

The objective behind this question is to determine if the candidate can differentiate between when they CAN block SR high flux (because P-6) is in and when they are ALLOWED to block SR high flux - after observing "proper" overlap between IR and SR - i.e. one decade. They will not observe one decade of overlap until both IR NI channels are > 1E-10 amps because they come on scale at 1E-11 amps.

A. **Incorrect**: < 1E-10 amps in both IR channels
Plausible: Source range nuclear instruments should be blocked by the time the level is 10^4 CPS by the training material

B. **Incorrect**: only IR A channel has reached 1 decade of observed overlap with the SR
Plausible: this is when P6 is in and they can physically block SR high flux

C. **Correct answer:** proper overlap has been observed on IR B

D. **Incorrect:** - will reach the SR high flux trip setpoint at 1E10 cps
Plausible: - if the candidate does not know P6 or if he is confusing the high flux trip setpoint with the P6 setpoint

Bank Question: 191 ***Answer: B***

1 Pt(s)

Unit 1 was operating at 100% power when a total loss of feedwater occurred. The operators reached step 35 of FR-H.1(Response to Loss of Secondary Heat Sink) which states:

> ***<u>IF AT ANY TIME</u> while in this procedure any S/G W/R level goes below 12% (17% ACC), <u>THEN GO TO</u> Enclosure 10 (Hot/Dry Steam Generator Limits)***

Given the following conditions:

	Loop A	Loop B	Loop C	Loop D
S/G (WR) [%]	0	15	9	10
NC T_{Hot} [°F]	150	555	530	545

- Containment pressure is 3.4 psig
- The TD CA pump is available to feed the S/Gs

Which one of the following statements correctly describes the bases for the restrictions for restoring feedwater flow following feed and bleed in FR-H.1?

A. **Restore flow to the A S/G because loop A T-hot is the lowest of the loops and this will reduce the chance of thermal shocking the S/G tube sheet. Flow should not be restored to the B and C S/Gs because they will be reserved for use later to provide a steam supply for the TD CA pump.**

B. **Restore flow to the B S/G because B S/G level is the highest and this will reduce the chance of thermal shocking the S/G tube sheet. Flow should be preferentially restored to the B or C S/G to maintain the TD CA pump steam supply.**

C. **Restore flow to the C S/G because loop C T-hot is less than loop B T-hot and this will reduce the chance of thermal shocking the S/G tube sheet. Flow should be preferentially restored to the B or C S/G to maintain the TD CA pump steam supply.**

D. **Restore flow to the D S/G because the D S/G is higher than A S/G level, which will reduce the risk of thermal shock. Flow should not be restored to the B and C S/Gs because they will be reserved for use later to provide a steam supply for the TD CA pump.**

Distracter Analysis: There was a change in this procedure since the last NRC exam. The previous guidance was not to feed a S/G when T-hot > 550 °F and to select the B and C S/Gs for restoration of flow. Now the guidance is to select the S/G that has the highest apparent level and to preferentially select the B or C S/G.

A. **Incorrect**: T-hot should not be used to determine which S/G should receive flow. It is not a reliable means of determining S/G shell temp in a dry stagnant loop.
Plausible: The apparent temp of the A loop is the lowest and it may appear that the chance of thermal shock is lessened.

B. **Correct answer**: feed the S/G that has the highest level and preferentially feed B & C S/Gs to maintain steam supply to the TD CA pump.

C. **Incorrect**: C S/G has a lower S/G level than B S/G
Plausible: C S/G has a lower T-hot than B S/G

D. **Incorrect:** No basis for reserving the B & C S/Gs for restoring flow
Plausible: There is a high probability that restoring feed to a dry S/G could rupture the tube sheet due to thermal stress. It makes sense to select a S/G that is NOT used to supply steam to the TD CA pump for the initial restoration of the heat sink.

Bank Question: 207 ***Answer: B***

1 Pt(s) Unit 1 is preparing for a reactor start up following a refueling outage. Given the following conditions:

- T_{avg} = 515 °F
- Plant heatup in progress using NCPs

At 0200, a Station Engineer reports that a mistake had been made in analyzing the containment Appendix J Leak Rate Test results that were conducted prior to exceeding 200 °F in Mode 5. Reanalysis indicated that the combined containment leak rate (Type A) had exceeded 1.0 L_a.

Which one of the following actions is required by Tech Specs in response to this situation?

REFERENCES PROVIDED
Tech Spec 3.6.1PAGES 1, 2
Tech Spec Bases B3.6.1-3, 4

A. Commence a plant cooldown to reach Mode 5 within 30 hours.

B. Commence a plant cooldown to reach Mode 5 within 36 hours.

C. Commence a plant cooldown to reach Mode 5 within 37 hours.

D. Commence a plant cooldown to reach Mode 5 within 43 hours.

Distracter Analysis:
This question tests the application of ITS completion times. Tech Spec 3.6.1 allows 1 hour to restore containment to an operable status, 6 hours to be in mode 3 and 36 hours to be in mode 5. However, the plant is currently in mode 3 - but this does not change the 36 hour completion time to enter mode 5. This represents a change over the old method of computing completion times.

A. **Incorrect**: 36 hours is allowable
Plausible: if the candidate deducts the 6 hours to be in mode 3 from the total 36 hours to enter mode 5 - would have been a correct method of determining completion time under old tech specs

B. **Incorrect**: 36 hours is allowable
Plausible: if the candidate takes the total 36 hours to enter mode 5 and adds in 1 hour to restore containment to operability - would have

been a correct method of determining completion time under old tech specs

C. **Correct answer** ITS completion times are different than old Tech Specs

D. **Incorrect**: 36 hours is allowable
Plausible: if the candidate adds the 6 hours to enter mode 3 to the 36 hours to reach mode 5 and adds 1 hour to restore containment to operability. This would have been the old Tech Spec method of doing completion times if the candidate made the mistake of NOT recognizing he/she was already in mode 3.

Bank Question: 216 ***Answer: B***

1 Pt(s) Which one of the following describes a responsibility associated with the Fuel Handling SRO during Fuel Handling operations?

A. Operate Fuel Handling Equipment, in accordance with approved procedure(s).

B. Directly observe Fuel Handling activities from the reactor building operating deck.

C. Physically latch/unlatch each fuel assembly, in accordance with approved procedure(s).

D. Supervise reactor vessel in-service inspections immediately after core off-load.

Distracter Analysis:

A. **Incorrect**: Fuel Handling SRO does not operate the FH equipment
Plausible: The FH SRO is responsible to supervise the operation of the FH equipment

B. **Correct Answer:**

C. **Incorrect**: Latching/unlatching fuel assemblies conducted by fuel handlers
Plausible: FH SRO supervises this operation

D. **Correct answer:**

Bank Question: 217 ***Answer: A***

1 Pt(s)

Unit 1 was operating at 100% power in Mode 1. Given the following conditions:

- 2 main steam safety valves (MSSVs) on the 1D S/G have been gagged shut to prevent chattering

Which one of the following statements describes the required action(s) and a basis for these actions?

REFERENCES PROVIDED
Tech Spec 3.7.1 pages 1-3

A. Reduce power below 39% to ensure that the reactor coolant pressure boundary is not over-pressurized.

B. Reduce power below 39% to ensure that the positive reactivity effect on NCS cooldown associated with the operation of the main steam system safety valves is minimized.

C. Reduce power below 19% to ensure that the reactor coolant pressure boundary is not over-pressurized.

D. Reduce power below 19% to ensure that the positive reactivity effect on NCS cooldown associated with the operation of the main steam system safety valves is minimized.

Distracter Analysis:

A. **Correct Answer**:
B. **Incorrect**: incorrect basis
Plausible: power level is correct
C. **Incorrect**: power level too low – 39% is correct
Plausible: basis is correct, will select 19% power if they look up tech spec actions required for 2 safety valves operable (rather than 2 inoperable)
D. **Incorrect**: power level too low – 39% is correct, basis is incorrect
Plausible: opening MSSVs will cause a positive reactivity transient – will select 19% power if they look up tech spec actions required for 2 safety valves operable (rather than 2 inoperable)

Bank Question: 241 ***Answer: D***

1 Pt(s)

Unit 1 is operating at 100% power when the supply breaker from 1LXG to Control Rod Drive MG set #2 opens. Which one of the following sequence of events will occur to the reactor trip breakers A or B (RTA/B) and the reactor trip bypass breakers A or B (BYA/B)?

A. RTA and BYB will open

B. RTB and BYA will open

C. BYA and BYB will open

D. No breakers will open

Distracter Analysis:

A. **Incorrect**: the rod drive MG sets are run in parallel – losing one MG set will not cause any reactor trip breakers to open.
Plausible: If the candidate thinks that the 1B rod drive MG set provides control power RTA and BYB.

B. **Incorrect**: the rod drive MG sets are run in parallel – losing one MG set will not cause any reactor trip breakers to open.
Plausible: If the candidate thinks that the 1B rod drive MG set provides control power RTB and BYA.

C. **Incorrect**: the rod drive MG sets are run in parallel – losing one MG set will not cause any reactor trip breakers to open.
Plausible: If the candidate thinks that the 1B rod drive MG set provides control power BYA and BYB.

D. **Correct answer:** the rod drive MG sets are run in parallel – losing one MG set will not cause any reactor trip breakers to open.

Bank Question: 242 ***Answer: A***

1 Pt(s) Unit 1 is operating at 100% power. Given the following conditions on the 1A NCP:

Time	0200	0210	0220	0230
Motor winding temp (F°):	312	315	320	324
Pump shaft vibration (mils):	15	16	18	21
#1 seal ΔP (psid):	201	196	223	235
#1 seal outlet temp (F°):	201	226	236	240

What is the earliest time that the operators are required to trip NCP-1A?

A. **0200**

B. **0210**

C. **0220**

D. **0230**

Distracter Analysis: Objective – to determine if the candidate can analyze the above conditions and select the correct time to trip the NCP. Parameters are selected to plausibly distract on the basis of the different set points.

A. **Correct:** Must trip when motor winding temperature exceeds 311 °F

B. **Incorrect**: reached trip set point at 0200
Plausible: NCP #1 seal differential pressure is less than limit of 200 psid

C. **Incorrect**: reached trip set point at 0200
Plausible: NCP #1 seal outlet temp exceeds the limit of 235 °F

D. **Incorrect:** reached trip set point at 0200
Plausible: pump shaft vibration limit reached at 20 mils

Bank Question: 243 ***Answer: C***

1 Pt(s) Unit 1 is conducting a plant startup in Mode 1. The operators have reached 8% power when a momentary electrical transient occurs resulting in the following conditions:

Bus	1TA	1TB	1TC	1TD
Frequency (Hz)	55	60	55	60
Voltage (VAC)	6410	6900	6410	6900

Which one of the following sequences would occur?

A. **A reactor trip does NOT occur and NCPs 1A and 1C trip on under-frequency while NCPs 1B and 1D continue running.**

B. **A reactor trip occurs and NCPs 1A and 1C trip on under-voltage while NCPs 1B and 1D continue running.**

C. **A reactor trip does NOT occur and all four NCPs trip on under-frequency.**

D. **A reactor trip occurs and all four NCPs trip on under-frequency**

Distracter Analysis:

A. **Incorrect**: all 4 NCPs trip due to the NC pump monitor system action
Plausible: only 2 pumps have a low frequency condition

B. **Incorrect**: all 4 NCPs trip due to the NC pump monitor system action - the reactor dies not trip as power is below P-7 (10%)
Plausible: only 2 pumps have a low voltage condition

C. **Correct answer**

D. **Incorrect**: the reactor does not trip below P-7 (10%)
Plausible: all four NCPs trip due to under-frequency on 2/4 NCPs

Bank Question: 298 ***Answer: D***

1 Pt(s) Unit 1 was operating at 100% power. Given the following conditions:

- Pressurizer pressure controller is selected to "1-2"
- Pressurizer pressure controls are in AUTO
- Pressurizer pressure channel I detector fails LOW

Which one of the following describes the plant response with no operator action?

A. High pressurizer pressure reactor trip will occur.

B. PORV 1NC-34A will maintain NC system pressure 80 to 100 psig above normal.

C. PORV 1NC-34A will maintain NCS pressure from 100 psig above normal to 50 psig below normal.

D. PORVs 1NC-32B and 1NC-36B maintain NC system pressure 80 to 100 psig above normal.

Distracter Analysis: This question was modified from a question on the Catawba NRC Exam from 1997. The stem was changed from position 3-2 to position 1-2 and distracter C was changed and the correct answer was changed.

A. **Incorrect**: no trip will occur
Plausible: would be the correct answer for pressure control in the 3-2 position – this was the correct answer on the 1997 Catawba NRC exam

B. **Incorrect**: NC-34A will not open
Plausible: the plant pressure control band is correct but NC-34A only opens if pressurizer pressure channel I fails high, not low - right pressure, wrong PORV

C. **Incorrect:** NC-34A will not open
Plausible: NC-34A opens if pressurizer pressure channel I fails high and the pressure control band is correct for NC-34A

D. **Correct answer**

Bank Question: 307 ***Answer: C***

1 Pt(s) Unit 1 was operating at 100% power when a crud burst occurred. Given the following events and conditions:

- EMF-48 (Reactor Coolant Hi Rad) trip 2 alarm
- 1EMF-18 (Reactor Coolant Filter 1A) trip 2 alarm

Which one of the following actions is required to reduce coolant activity due to a crud burst in the NC system?

A. Purge the VCT with nitrogen

B. Place/ensure both mixed bed demineralizers are in service

C. Increase letdown flow

D. Add hydrogen to the reactor coolant

Distracter Analysis:

A. **Incorrect**: Will not correct a high NC activity from a crud burst
Plausible: One of the subsequent actions in AP/18 is to purge the VCT to the waste gas system with Hydrogen. In addition, Nitrogen is used to purge the VCT for shutdown. It is likely that a candidate could mix up these purges.

B. **Incorrect**: Do not want to load crud particles into BOTH mixed bed demineralizers
Plausible: Mixed bed demins will filter crud particles and remove fission product ionic impurities - this action required for fuel element failure/high fission product activity in AP/18 - but not for crud burst

C. **Correct**: Will increase removal rate of crud particles by increased filtration.

D. **Incorrect**: Will not remove crud burst particulate activity
Plausible: Used to scavenge Oxygen from the NC coolant and thus reduce the corrosion rates and crud production in the RCS. However, this does not affect crud burst particulates that are already in the NC system coolant.

Bank Question: 308 ***Answer: D***

1 Pt(s)

Unit 1 is operating at 100% power. Given the following conditions:

- Rod control is in manual
- Control Bank D is at 200 steps

If the rods in control bank D start stepping out at 8 steps per minute, what one of the following actions is required at this time?

A. **Select Control Bank D on the rod selector switch and manually insert Control Bank D**

B. **Select "AUTO" on the Bank Select Switch and see if rod motion stops**

C. **Commence emergency boration**

D. **Trip the reactor**

Distracter Analysis:

A. **Incorrect**: The correct response is to trip the reactor for a rod withdrawal
Plausible: this action could stop the rod withdrawal, as the rods in signal should over-ride the rods out signal

B. **Incorrect**: Trip the reactor is the correct response.
Plausible: If the malfunction was in the manual section of the rod control circuitry, this could stop the rods. If the rod control was in auto – then going to manual would be the correct answer. This reverses that thought process.

C. **Incorrect**: Trip the reactor is the correct response
Plausible: This action would be required to insert negative reactivity if the trip did not work

D. **Correct answer**: Immediate action in step 3 of AP-14

Bank Question: 311 ***Answer: A***

1 Pt(s) Unit 1 is operating at 50% power. Given the following conditions:

- Pressurizer pressure is 2235 psig
- Pressurizer Relief Tank (PRT) pressure is 20 psig
- PRT temperature is 125 °F
- PRT level is 81%
- The PRT is being cooled by spraying from the RMWST
- A pressurizer code safety valve is suspected of leaking by it's seat

What temperature would be indicated on the associated safety valve discharge RTD if the code safety were leaking by?

REFERENCES PROVIDED: Steam Tables

A. 258-262 °F

B. 228-232 °F

C. 161-165 °F

D. 123 -127°F

Distracter Analysis:

A. Correct answer

B. Incorrect: Temp is too low - the correct temp is 260 °F
Plausible: If the candidate makes the mistake of not correcting for atmospheric pressure by failing to adding 14.6 psi to the PRT pressure and uses 20 psia.

C. Incorrect: Temp is too low - the correct temp is 260 °F
Plausible: If the candidate reverses the correction for atmospheric pressure by subtracting 14.6 psi from PRT pressure of 20 psig to get 5 psia.

D. Incorrect: Temp is too low - the correct temp is 260 °F
Plausible: If the candidate thinks that the discharge temperature will be at the same temperature as the PRT fluid.

Bank Question: 338 ***Answer: C***

1 Pt(s) Which one of the following statements complies with the requirements of OMP 4-3 regarding the rules of usage for abnormal procedures (APs) when the EOPs have been implemented?

A. **APs may not be implemented when EOPs have been entered.**

B. **Only one AP at a time may be implemented when EOPs have been implemented. Concurrent implementation of APs when EOPs are in use is not allowed.**

C. **APs may be implemented concurrently with EOPs. However, the APs were written assuming that SI has not actuated and operators must be careful when using APs if SI has occurred.**

D. **APs may be implemented concurrently with EOPs with the exception of events where SI has actuated. APs were written assuming the SI had not occurred and cannot be used if SI has actuated.**

Distracter Analysis:

A. **Incorrect**: APs may be entered after EOPs have been started
Plausible: Many plants have this provision - symptomatic EOPs should address all significant safety challenges without requiring APs

B. **Incorrect**: No limitation on the number of APs
Plausible: Makes sense to limit the number of concurrent procedures in use

C. **Correct answer**

D. **Incorrect**: No explicit prohibition against use of APs when SI has actuated BUT there is a caution and the APs were written for the situation where SI has NOT occurred.
Plausible: APs were written for the situation where SI has NOT occurred.

Bank Question: 353 ***Answer: D***

1 Pt(s)

A male worker needs to repack a valve in an area that has the following radiological characteristics:

- The worker's present exposure is 1800 mrem for the year.
- General area dose rate = 65 mrem/hr
- Airborne contamination concentration = 20 DAC

The job will take 4 hours with a mechanic wearing a full-face respirator. It will only take 2 hours if the mechanic does NOT wear the respirator.

Which of the following choices for completing this job would maintain the workers exposure within the Station ALARA requirements?

A. The worker should wear the respirator otherwise he will exceed 25% of the DAC limit.

B. The worker should NOT wear the respirator because the dose received will exceed neither NRC nor site dose limits.

C. The worker should wear the respirator because the total TEDE dose received will be less than if he does not wear one.

D. The worker should NOT wear the respirator because the total TEDE dose received will be greater than if he wears one.

Distracter Analysis:

Radiation exposure comparison:
Without respirator
DDE = 65 mrem/hr x 2 hr = 130 mrem
From airborne contamination:
CEDE = 20 DAC 2 hr x 2.5 mrem/DAC-hr = 100 mrem
TEDE = 130 + 100 = 230 mrem from job
Total exposure for year = 1800 + 230 = 2030 mrem

With respirator
DDE = 65 mrem/hr x 4 hr = 260 mrem
CEDE = 0
TEDE = 260 mrem
Total exposure for year = 260 + 1800 = 2060 mrem

(with respirator) (without respirator)
TEDE = 2060 mrem > 2030 mrem = do NOT use a respirator

A. **Incorrect**: Will not exceed 25% the DAC limit - this is not how DAC is applied to exposure limits
Plausible: 25% DAC is the limit at which an area requires posting as a high airborne contamination area.

B. **Incorrect**: The dose will exceed station admin limits of 2000 mrem
Plausible: if the candidate does not know the station admin limit or miscalculates the dose received

C. **Incorrect**: The exposure will be greater if you wear the respirator
Plausible: If the candidate incorrectly computes the exposure - this was the correct answer on the 1997 Catawba NRC exam

D. **Correct answer**

Bank Question: 372 ***Answer: C***

1 Pt(s)

Unit 2 was responding to a faulted steam generator event. The operators entered FR-P.1 (Response to Imminent PTS) and reach step 15 where they are directed to isolate cold leg accumulators (CLAs).

What is the EOP basis for isolating the CLAs in FR-P.1?

A. To prevent injecting the CLA nitrogen bubble into the reactor and creating a gas bubble in the vessel head region.

B. To prevent repressurizing the reactor vessel and adding pressure stress to thermal stress.

C. To prevent adding more cold water to the reactor vessel and increasing the thermal stress.

D. To prevent depleting CLA volume and to preserve a source of highly borated water to prevent recriticallity during cooldown.

Distracter Analysis:

A. **Incorrect**: - the CLAs are isolated to prevent adding to the thermal stress. The gas bubble would not be a limiting concern in FR-P.1
Plausible: - this is a valid limiting condition for isolating the CLAs during LOCA depressurizations - good answer, wrong event.

B. **Incorrect**: - adding the CLA volume would not cause an increase in pressure because the addition of the CLA volume is caused by pressure in the system decreasing below CLA pressure.
Plausible: - the basis for terminating SI is to prevent adding water to the system and increasing pressure thereby adding pressure stress to thermal stress. Not applicable to the CLAs.

C. **Correct answer**

D. **Incorrect:** - the CLAs are not require for Boron addition for this scenario.
Plausible: - re-criticality this is a valid concern for scenarios that involve adding large quantities of unborated water. Good answer - wrong event.

Bank Question: 390 ***Answer: B***

1 Pt(s) Unit 1 is recovering from a LOCA. The operators started the process of terminating safety injection at 2:00 AM. Given the following indications at the following times:

	Parameter	2:00	2:05	2:10	2:15
1)	Pressurizer level (%)	40	29	15	11
2)	NC pressure (psig)	280	285	290	295
3)	ND Flow	1000	1025	1075	1085
4)	Core exit T/Cs (°F)	690	702	695	685
5)	FWST level (inches)	183	179	149	113
6)	Containment Pressure (psig)	3.5	2.3	1.2	1.1

What is the earliest time that the operators should transition to ES-1.3, transfer to cold leg recirculation.

A. **0200**

B. **0205**

C. **0210**

D. **0215**

Distracter Analysis: This question is designed to test the candidate's ability to identify the criteria for switchover to cold leg recirc from a list of plant parameters. A change to the switchover criteria was changing the FWST level from 150 inches to 180 inches. In addition, he meets the S/I reinitiation criteria at 0205 and this will check if he recognizes that ES-1.3 has priority

A. **Incorrect**: - does not meet criteria for switchover, FWST > 180 inches
Plausible: - .if candidate does not know foldout criteria for switchover

B. **Correct answer** - FWST level < 180 inches - reaches ACC value for S/I reinitiation criteria - needs to decide if ES-1.3 needs to be delayed until S/I reinitiation is completed

C. **Incorrect**: - too late
Plausible: - if candidate does not know foldout criteria for switchover - recent change (since last exam) to the switchover criteria - used to be 150 inches - now is 180 inches

D. **Incorrect:** - too late
Plausible: - if candidate thinks that SI termination must be competed before switchover to cold leg recirc or if he thinks that S/I reinitiation takes priority over switchover - reaches non-ACC value for S/I reinitiation.

Bank Question: 401 ***Answer: D***

1 Pt(s)

Unit 2 was operating at 100% when a single control rod in control bank D drops into the core due to a failed CDRM. The SRO directs that the dropped rod be recovered.

Which one of the following prevents the remaining rods in the control bank from being withdrawn while the dropped rod is being recovered?

A. The rod control non-urgent failure alarm will actuate when the dropped rod is withdrawn blocking all rod motion.

B. C-11 actuated when the rod dropped and will prevent outward rod motion by control bank D

C. The Lift Coil Disconnect Switch is opened on the dropped rod to electrically isolate it from control bank D

D. The Lift Coil Disconnect Switches are opened on control bank D rods that did not drop

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Distracter Analysis:

A. **Incorrect**: the non-urgent failure alarm does not actuate unless you lose a power supply to a logic or power cabinet
Plausible: - the urgent failure alarm actuates when the rod is withdrawn and this would block rod motion for all rods on the opposite power cabinet to the dropped rod.

B. **Incorrect**: - C-11 will not allow any **auto** rod motion but allows manual motion – C-11 has not actuated under these conditions (bank D at top of core)
Plausible: - C-11 will prevent outward rod motion in bank D

C. **Incorrect**: - will not be able to pick up the dropped rod
Plausible - if the candidate was not familiar with the actions of the lift coil disconnect switch - it will disconnect the rod from bank D but the rod cannot then be withdrawn with the switch open

D. **Correct answer**

Bank Question: 404 ***Answer: A***

1 Pt(s) Unit 1 was responding to a small break LOCA. Containment pressure reached 3.5 psig. The Subcooling Margin Monitor currently indicated +35 °F. Which of the following statements best describes the status of subcooling in the core?

A. The core is subcooled by 35 °F

B. The core is superheated by 35 °F

C. The core is superheated by more than 35 °F due to the effects of adverse containment conditions

D. The core is subcooled by more than 35 °F due to the effects of adverse containment conditions

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Distracter Analysis: This is a modified question from a previous NRC exam. The original question asked what the core conditions were if ICCM was reading –35 °F. The original answer was "C". Note: the upper limit for measuring superheat is –35 °F. The lower limit is +200 F.

Although the ICCM was designed for ACC inputs, this option was never used because the pressure transmitters were located outside of containment

A. **Correct**:

B. **Incorrect**: - subcooling is 35 °F
Plausible: - if the candidate reverses the meaning of the indication (i.e. – means subcooled, + means superheated)

C. **Incorrect**: - subcooling is 35 °F -
Plausible: - if the candidate reverses the meaning of the indication – this was the answer on the NRC exam in 1997.

D. **Incorrect**: - subcooling is 35 °F
Plausible: - if the candidate reverses the meaning of the indication.

Bank Question: 407 ***Answer: C***

1 Pt(s)

Unit 1 has a liquid radioactive waste release in progress from the Ventilation Unit Condensate Drain Tank (VUCDT) through the RC system. All lineups and authorizations have been properly made in accordance with OP/0/B/6200/35 using the normal path. 2 RC pumps are the minimum required under LWR document.

Given the following initial conditions:

- 2 RC pumps are running
- Controlling EMF properly adjusted for trip 1 and trip 2 settings
- No other releases are in progress

What automatic actions would terminate the release?

A. WM-46 will close automatically if 1 RC pump trips

B. WM-46 will close automatically when EMF-44 (VUCDT) reaches the trip 2 setpoint

C. WL-320 and WP-35 will close automatically if 1 RC pump trips

D. WL-320 and WP-35 will close automatically when EMF-49 (Liquid Waste) reaches the trip 2 setpoint

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Distracter Analysis: Used a similar question on the last NRC exam – but modified the stem and the answer. The answer from the previous exam was "D". 1EMF-44 is the controlling EMF, not 1EMF-49.

A. **Incorrect**: WM-46 is isolated and not used anymore as a release path.
Plausible: - RC pump interlock will actuate - set at 2 pumps (minimum required on LWR document). – WM-46 was formerly the normal release path

B. **Incorrect**: - WM-46 receives a closing signal from EMF-44 but this is not the normal path for a release. WM-46 is isolated and not used anymore.
Plausible: - this was formerly the normal release path – EMF-44 sends a closing signal to WM-46 but the valve is no longer in service.

C. **Correct**: - RC pump interlock will actuate - set at 2 pumps (minimum required on LWR document).

D. **Incorrect**: - EMF-49 does not trip WL-320 is not used to monitor the release from the VUCDT
Plausible: - EMF-49 would monitor and isolate a liquid release from the Waste Monitor Tank (WMT) This was the correct answer from the last NRC exam – except the monitor referenced was 1EMF-44 instead of 1EMF-49.

Bank Question: 412 ***Answer: A***

1 Pt(s) Unit 1 is responding to a LOCA. Given the following initial conditions:

- A reactor trip and safety injection actuation occurred at 0150
- MSIVs are shut.
- Phase B containment isolation has occurred

The operators reach step 2 in ES-1.1 (SI Termination) requiring a reset of the safety injection signal.

Given the following parameter trends at 0200:

- NC pressure = dropped to 1850 psig then stabilized at 1951 psig
- Steamline pressure = 771psig - decreasing slowly
- Containment pressure = 2.2 psig - decreasing slowly

Given the following sequence of operator actions:

0202 Blocks the low steam line pressure MSI signal
0203 Blocks the low PZR pressure SI signal
0204 Resets the phase B isolation signal

What is the earliest time that depressing the SI reset pushbuttons (trains A and B) would reset safety injection?

A. 0200

B. 0202

C. 0203

D. 0204

Distracter Analysis:
This question will test if a candidate understands that safety injection can be reset even with valid SI actuation signals still present (not blocked). The only restrictions are the 60-second timer and P-4 (RTBs open).

A. **Correct answer** - safety injection can be reset after a 60 second timer has elapsed and the train related reactor trip breaker has opened (P-4). None of the SI signals being present will prevent reset of SI. Once reset, only manual SI is available

B. **Incorrect**: - SI already reset at 0200
Plausible: - the steam line low pressure MSI can be blocked < P-11 - doesn't effect SI

C. **Incorrect**: - SI already reset at 0202
Plausible: - this will block the low pressurizer pressure SI signal

D. **Incorrect**: - SI already reset at 0202
Plausible: - this action will block hi-hi containment pressure SI signal

Bank Question: 430 ***Answer: A***

1 Pt(s)

Unit 1 is responding to a large break LOCA into containment. Given the following events and conditions:

- Containment spray auto-actuated and reduced containment pressure
- Containment pressure is now at 0.24 psig and continues to increase
- NS actuation logic has been reset by the operators

Which one of the following describes the NS system response to an increase in containment pressure?

A. **NS pumps will start and discharge valves will open after containment pressure reaches 3.0 psig.**

B. **NS pumps will start and discharge valves will open after containment pressure exceeds 0.8 psig.**

C. **NS pumps will start when containment pressure reaches 0.35 psig and discharge valves will open when containment pressure exceeds 0.8 psig.**

D. **Discharge valves will open when containment pressure exceeds 0.35 psig and NS pumps will start when containment pressure exceeds 0.8 psig.**

Distracter Analysis: This is a modification of a question used in the 1999 exam. The changes were to modify the condition that NS spray was reset. The previous correct answer was D. Answer A was also modified. The normal expected behavior of NS is as stated in D. If Containment pressure did not fall below 0.35 psig, then D would still be correct because the CPCS interlock would not be satisfied.

A. **Correct answer** – with containment spray actuation reset, the valves will open and the pumps will start at 3.0 psig

B. **Incorrect**: - valves do not open at 0.8 psig
Plausible: - pumps start at 0.8 psig by CPCS

C. **Incorrect**: - pumps do not start at 0.35 psig and valves do not open at 0.8 psig
Plausible: - this is backwards to what happens

D. **Incorrect**: - containment spray actuation was reset
Plausible: - this is how the system would function if containment spray actuation was NOT reset.

Bank Question: 447 ***Answer: D***

1 Pt(s)

Unit 1 is shutdown, Mode 6, in a refueling outage. Given the following conditions:

- Containment airlock doors are both open
- A full shift of qualified maintenance personnel are available inside containment
- The Refueling SRO is in the control room
- The Fuel Handling Supervisor is inside containment

Refueling has been completed and the Fuel Handling Supervisor (who is not a qualified SRO) requests permission to latch all control rods to prepare for the reactor startup. What additional requirements must be met (if any) to proceed with latching rods?

A. Latching rods may proceed at the discretion of the Fuel Handling Supervisor.

B. Latching rods may not proceed until after containment integrity has been restored.

C. Latching control rods may not proceed until after the Refueling SRO arrives inside containment to supervise.

D. Latching control rods may not proceed until after the Refueling SRO arrives inside containment and containment integrity has been restored.

Distracter Analysis:

A. **Incorrect**: - the Refueling SRO is required to supervise this evolution and containment integrity must be restored
Plausible: - if the candidate does not recognize that latching rods is a core alteration or doesn't recognize that this requires containment integrity to be established

B. **Incorrect:** - the Refueling SRO is required to supervise this evolution
Plausible: - if the candidate does not recognize that latching rods is a core alteration

C. **Incorrect**: - containment integrity must first be established
Plausible: - core alterations requires SRO coverage and containment integrity

D. **Correct answer**

Bank Question: 451 ***Answer: B***

1 Pt(s) Unit 1 is shutdown in a refueling outage. Given the following events and conditions:

- A VI header rupture occurs
- The VI system completely depressurizes.
- VI-820 was open at the time of the rupture.
- The VS system was in a normal lineup

What effect does a total loss of the VI system have on the VS system?

A. VI-820 will auto-close as VI header pressure decreases below 90 psig and the VS air compressor will start automatically at 82 psig to maintain VS header pressure

B. VI-820 will auto-close as VI header pressure decreases below 82 psig and the VS air compressor must be manually started to maintain VS header pressure

C. Check valves in the VI - VS cross-connect line will close to isolate VS system pressure before it drops below 90 psig

D. VS pressure in the Fire Protection Pressurizer Tank will be lost until a VS air compressor can be started.

Distracter Analysis:

A. **Incorrect**: - the VS air compressor does not automatically start to maintain pressure - VI-820 auto-closes at 82 psig not 90 psig
Plausible: - The VI system is safety significant, VI-820 does close at 82 psig and there is a separate VS air compressor which has an automatic startup feature – but it just is normally in "off" and requires operator action to start.

B. **Correct answer**

C. **Incorrect**: - there are no check valves in this line
Plausible: - this is another possible method to prevent depressurizing the VS header at some plants.

D. **Incorrect**: - the RF system tank is pressurized with VS air - but is maintained isolated from the VI header
Plausible: - if the candidate does not know that the RF system air tank is isolated from the VS header.

Bank Question: 465 ***Answer: A***

1 Pt(s) Unit 1 is operating at 15% power going to 100% power. The operators just completed synchronizing the main generator on the power grid. Which one of the following sequences describes the correct operator actions for increasing the main generator load?

A. **Select MW IN**
Raise the GV limit from 17% to 120%
Depress LOAD RATE pushbutton and enter desired load rate
Depress the REFERENCE pushbutton and enter the load
Depress the GO pushbutton

B. **Select MW IN**
Raise the GV limit from 17% to 120%
Depress STANDARD pushbutton and enter desired load and load rate using the keypad
Depress the GO pushbutton

C. **Select MW OUT**
Raise the GV limit from 17% to 100%
Depress STANDARD pushbutton and enter desired load and load rate using the keypad
Depress the GO pushbutton

D. **Select MW OUT**
Raise the GV limit from 17% to 100%
Depress the REFERENCE pushbutton and enter the load
Depress LOAD RATE pushbutton and enter desired load rate
Depress the GO pushbutton

Distracter Analysis:

A. **Correct answer**:

B. **Incorrect**: - do not use the STANDARD button
Plausible: - this is designed to work correctly but is not used at McGuire

C. **Incorrect**: - MW must be IN - do not use the STANDARD button
Plausible: - if the candidate does not understand the MW IN feedback loop

D. **Incorrect:** - MW must be IN for feedback loop
Plausible: - if the candidate does not understand the MW IN feedback loop

Bank Question: 471 ***Answer: D***

1 Pt(s) Unit 1 is responding to a LOCA. Given the following events and conditions:

- Completed E-0 (Reactor Trip or Safety Injection)
- Entered E-1 (Loss of Reactor or Secondary Coolant)
- The STA reported the following valid critical safety functions:
  - Subcriticality - orange path
  - Integrity - red path
  - Heat Sink - red path
  - All other CSFs are green or yellow

What procedure should be operator select?

A. Remain in E-1 (Loss of Reactor or Secondary Coolant)

B. Transition immediately to FR-S.1 (Response to Nuclear Generation /ATWS)

C. Transition immediately to FR-P.1 (Response to Imminent Pressurized Thermal Shock Condition)

D. Transition immediately to FR-H.1 (Response to Loss of Secondary Heat Sink)

Distracter Analysis:

A. **Incorrect**: - must transition to CSFs
Plausible: - if candidate does not know restrictions and applicability of F-0

B. **Incorrect**: - Orange path does not have priority over red paths
Plausible: - if candidate does not know rules of usage

C. **Incorrect:** - Integrity does not have priority over Heat Sink
Plausible: - if candidate does not know CSF rules of usage

D. **Correct answer**: - Heat sink does not have priority over core cooling

Bank Question: 479 ***Answer: A***

1 Pt(s)

Unit 1 is in the process of making a radioactive gaseous waste release from the waste gas decay tank in accordance with OP/0/A/6200/18. Given the following conditions:

- MRIRR = 21 CFM
- MOSRR = 40 CFM
- 1EMF-50 trip 1 setpoint = 1.0E5 CPM
- 1EMF-50 trip 2 = 2.0E5 CPM
- 1EMF-36 is in service

Time	0200	0215	0230	0245
Release rate (CFM)	22	25	41	37
EMF-50 (CPM)	1.8E5	2.2E5	2.1E5	3.2E5

If the operators reset 1EMF-50 whenever allowed by procedure, what is the earliest time (if any) that the operators are **required** to terminate the gaseous release.

A. 0200

B. 0215

C. 0230

D. 0245

Distracter Analysis:

A. **Correct** answer - the release rate (21 CFM) > MRIRR (20 CFM) (most restrictive instantaneous release rate)

B. **Incorrect**: - exceeded MRIRR at 0200
Plausible: - EMF-50 tripped WG-160 for the first time

C. **Incorrect**: - exceeded MRIRR at 0200
Plausible: - exceeded MOSRR (maximum observed system release rate) - the operator is allowed to reset EMF-36 and restart the release 3 times before being required to terminate the release - this is the 2nd time EMF-36 has reached trip 1

D. **Incorrect**: - exceeded MRIRR at 0200
Plausible: - exceeded trip 2 on EMF-50 for the 3rd time - allowed to reset this trip 3 times before terminating release

Bank Question: 501 ***Answer: D***

1 Pt(s)

Unit 2 was operating at 100% power when a terrorist attack in the control room caused the operators to rapidly evacuate to the Auxiliary Shutdown Panel. The operators were not able to perform AP/17 (Loss of Control Room) actions prior to evacuation at 0200.

The terrorists tripped the turbine but did not operate any other controls. There are no other local operator actions taken. Given the following steam generator narrow range levels:

	0200	0202	0204	0206	0208
2A S/G NR	65%	37%	22%	15%	25%
2B S/G NR	64%	38%	23%	18%	26%
2C S/G NR	63%	39%	25%	16%	24%
2D S/G NR	65%	38%	26%	20%	27%

Which one of the following statements describes the complete list of running feedwater pumps when the operators first arrive at the ASP at 0210 to take local control of the plant?

A. Both motor driven CA pumps

B. Both motor drive CA pumps and the turbine drive CA pump

C. Both motor driven CA pumps and both CF pumps (in roll-back hold)

D. Both motor driven CA pumps, the turbine driven CA pump and both CF pumps (in roll-back hold)

Distracter Analysis: The lo-lo setpoint for SGWL is 17%. This causes:

- Reactor trip - on 1 of 4 S/Gs in 2 of 4 channels
- MD CA pumps auto-start - on 1 of 4 S/Gs in 2 of 4 channels
- TD CA pump auto-start – on 2 of 4 S/Gs in 2 of 4 channels

A. **Incorrect**: CF pumps will not trip – this is done by a local operator action in AP-17, TD CA pump auto-starts.
Plausible: MD CA pumps will start when S/G levels < 17% on 1/4 S/Gs

B. **Incorrect**: The CF pumps will continue to run until tripped by local operator action in AP-17
Plausible: The MD and TD CA pumps auto start

C. **Incorrect**: The TD CA pump will auto start.
Plausible: The MD CA pumps auto start and the CF pumps remain running

D. **Correct answer**:

Bank Question: 504 ***Answer: C***

1 Pt(s)

Unit 1 was operating at 100% power when a reactor trip occurred due to a feedwater control valve malfunction. Given the following events and conditions:

- Both motor-driven CA pumps started
- The operators have entered E-0 (Reactor Trip or Safety Injection)
- Feedwater flow to each generator is greater than 450 gpm
- Given the following steam generator levels

Time	0200	0201	0202	0203	0205*
Steam Generator Level					
1A S/G (% NR)	8	12	22	39	45
1B S/G (% NR)	7	12	24	39	46
1C S/G (% NR)	5	13	25	40	47
1D S/G (% NR)	5	16	28	39	45
Containment pressure (psig)	1.5	3.5	2.5	1.5	0.6

***At 0205**, the operators reach step 16.c of E-0 that reads:

***WHEN** N/R level in any S/G greater than 11% (32% ACC), **THEN** control CA flow to maintain N/R levels between 11%(32%) and 50%.*

Which one of the following statements correctly describes when the operators are allowed to reset and reduce CA flow to the steam generators?

A. Any time after 0201

B. Any time after 0202

C. Any time after 0203

D. Any time after 0205

Distracter Analysis: OMP 4-3 allows reset and control of CA flow when any S/G level reaches its normal setpoint (39%). Step 16 of E-0 directs the operators to control CA flow between 11% (32% ACC) and 50% S/G NR level. Until the operators reach step 16, they are allowed to control S/G level between 39% and 55% per OMP 4-3.

Note: This question was significantly modified – in the previous version of this question, the correct answer was A, can control S/G levels immediately.

A. **Incorrect**: no S/Gs have reached their normal control bands
Plausible: S/G levels exceed 11%, the "normal" setpoint for the controlling band after reaching step 16. If the candidate thinks that the S/Gs are within their control band or if the candidate thinks that OMP 4-3 allows control of levels at any time.

B. **Incorrect**: no S/Gs have reached their normal control bands.
Plausible: OMP 4-3 allows reset and control of CA flow when any S/G level reaches its normal control band which is 39%. However, the candidate may become confused with the control bands given in step 16. S/G levels exceed 11% but ACC values are now in effect and the minimum band for S/G level is 32%– even though containment pressure < 3.0 psig.

C. **Correct Answer**: OMP 4-3 allows control of S/G levels when they reach their "normal level setpoint" – in this case > 39% NR level

D. **Incorrect**: can reset CA at 0203 when S/G NR level reaches its control band.

Plausible: If candidates do not know OMP 4-3 guidance that allows resetting CA flow prior to reaching step 16.

Bank Question: 507 ***Answer: B***

1 Pt(s)

Unit 2 is responding to a LOCA into the Auxiliary Building in ECA-1.2 (LOCA Outside of Containment). Upon completion of ECA-1.2, NC system pressure continues to decrease. Which one of the following statements correctly describes the correct mitigating strategy to assure continued removal of decay heat under these conditions?

A. Transition back to E-1 (Loss of Reactor or Secondary Coolant).

B. Transition to ECA-1.1 (Loss of Emergency Coolant Recirc).

C. Transition to ES-1.2 (Post LOCA Cooldown and Depressurization)

D. Transition to ES-1.3 (Transition to Cold Leg Recirc).

Distracter Analysis:

A. **Incorrect**: Not the correct procedural transition if the NC system pressure continues to decrease (ie leak path not isolated)
Plausible: This IS the correct procedure if the NC system pressure was stable or increasing.

B. **Correct answer**

C. **Incorrect**: Transition to ES-1.2 not allowed as the leak is not isolated
Plausible: The name of the procedure is appropriate for the situation.

D. **Incorrect**: Transition to ES-1.3 not in accordance with the major action steps
Plausible: Although many actions are the same, it is not the correct procedure.

Bank Question: 512 ***Answer: B***

1 Pt(s)

Unit 2 was in the process of starting up the reactor following a refueling outage. Given the following plant conditions and events:

- Reactor trip breakers are closed
- Shutdown bank rod withdrawal has commenced
- Train A of Wide Range Shutdown Monitoring is inoperable

If source range N-32 fails, which one of the following actions is required?

A. **Startup may continue with train B of the Gamma-Metrics Shutdown Monitor System substituting for the failed N-32 source range channel**

B. **Immediately stop withdrawal of shutdown banks**

C. **Immediately open the reactor trip breakers**

D. **Immediately reinsert shutdown banks and open the reactor trip breakers**

Distracter Analysis:

A. **Incorrect**: Cannot substitute gamma metrics for SR channel
Plausible: Allowed to substitute SR channels for gamma metrics

B. **Correct answer**: immediate action per AP/16 case I

C. **Incorrect**: Not required unless 2 SR channels fail
Plausible: If candidate does not know tech spec requirements

D. **Incorrect**: Not a tech spec action
Plausible: If candidate does not know tech spec requirements

Bank Question: 531 ***Answer: C***

1 Pt(s) Unit 2 is operating at 75% power when a load rejection occurs. Which one of the following statements correctly describes the response of 2CM-420 (Load Rej Byp) to this transient?

A. **2CM-420 closes to prevent condensate water from being diverted to the suction of the hotwell booster pumps from the condensate booster pumps to assure minimum flow to the CF pumps.**

B. **2CM-420 closes to prevent diversion of water from the "C" heater drain tank back to the UST thereby ensuring sufficient CF pump suction pressure.**

C. **2CM-420 opens to divert condensate flow directly to the condensate booster pump suction to ensure that CF pumps have sufficient suction pressure.**

D. **2CM-420 opens to divert condensate flow, bypassing around the condensate booster pumps, directly to the CF pumps to assure minimum flow requirements.**

Distracter Analysis:

A. **Incorrect**: CM-420 opens - does not close. Does not prevent water from being recirculated around the hotwell pumps.
Plausible: this function is performed by CM-407 – which opens to assure minimum flow around the hotwell pumps to prevent water hammer on the CM system during startup.

B. **Incorrect**: CM-420 opens - does not close. Does not prevent a loss of water to the condensate booster pump suction.
Plausible: CM-227 opens to recirc condensate from the C feedwater heater to the USTs to assure minimum recirc flow on the CBPs

C. **Correct answer**

D. **Incorrect**: CM-420 does not provide a flow path around the condensate booster pumps directly to the CF pumps to meet minimum flow requirements
Plausible: CM-420 opens to provide bypass flow – but directly to the CBPs – not the CF pumps

Bank Question: 538 ***Answer: C***

1 Pt(s)

Unit 1 is operating at 28% power during a plant startup to 100%. Given the following conditions on the 1C steam generator:

- Main feedwater regulating valve (FRV) is in AUTO control at 25% open
- Bypass FRV is in MANUAL control at 100% open
- Steam flow channel I fails high

Which one of the following statements correctly describes the plant response for the 1C steam generator FRVs??

A. **Main FRV modulates open to increase feedwater flow and steam generator water level increases to the high level alarm setpoint.**

B. **Main FRV modulates shut to reduce feedwater flow and steam generator level decreases to the low level alarm setpoint.**

C. **Main FRV modulates open to increase feedwater flow but sufficient level error signal develops to restore CF flow to normal without reaching the high level alarm setpoint.**

D. **Main FRV modulates shut to reduce feedwater flow but sufficient level error signal develops to restore CF flow to normal without reaching the low level alarm setpoint.**

Distracter Analysis:

A. **Incorrect**: S/G water level does not increase to the high level alarm as level error quickly overcomes flow mismatch
Plausible: CF control valves open to 120%

B. **Incorrect**: FRVs do not modulate shut and SG water level does not fall to the low level alarm
Plausible: If the candidate reverses the effect of the instrument failure - this is what happens for a steam flow transmitter failing low

C. **Correct answer**

D. **Incorrect**: FRVs do not modulate shut
Plausible: level error does overcome flow mismatch and level will be restored

Bank Question: 591 ***Answer: D***

1 Pt(s)

Unit 2 was operating at 5% power during a plant startup when the following sequence of actions occurred.

- Opened 2NV-265B
- Started Boric Acid Transfer pump #2A

If no other operator actions occurred, which of the following statements correctly describes the response of reactor power and control rods?

A. **Power remains at 5%**
Control rods drive in

B. **Power remains at 5%**
Control rods do not move

C. **Power decreases**
Control rods drive in

D. **Power decreases**
Control rods do not move

Distracter Analysis: The candidate must recognize that:

- The sequence of operations amounts to emergency boration of the reactor.
- Control rods are in manual at this point during the startup
- Power decreases due to boron addition

A. **Incorrect**: power will decrease
Plausible: control rods would drive in – IF they were in auto control

B. **Incorrect**: power will decrease
Plausible: control rods will not move

C. **Incorrect**: control rods are in manual and will not move
Plausible: power will decrease

D. **Correct answer**

Bank Question: 592 ***Answer: B***

1 Pt(s)

Unit 1 was cooling down in Mode 4 when the 1A1 KC pump trips. Given the following conditions:

- Both trains of KC were initially in operation
- 1A2 KC pump was secured due to high KC flow
- Both trains of ND were aligned for RHR shutdown cooling
- NCS temperature was 205 °F

If train A KC pumps cannot be restarted, which one of the following list of actions is the **complete list** of actions that must be taken to prevent damage to equipment?

A. **Stop ND pump 1A**

B. **Stop ND pump 1A**
Isolate ND flow through the 1A ND heat exchanger

C. **Cross-connect KC flow to the 1A ND heat exchanger**
Cross-connect KC flow through the 1A ND Pump mechanical seal heat exchanger

D. **Stop ND pump 1A**
Isolate KC flow through the letdown heat exchanger

Distracter Analysis: Upon a loss of KC to an operating ND train, AP/21 requires two actions (per Foldout page):

- Stop the associated ND pump
- Isolate flow to the associated ND HX

A. **Incorrect**: Must also stop flow to the ND HX per AP/21
Plausible: action to stop the 1A ND pump is correct. There is a separate operating precaution to maintain flow through the ND HX > 2000 gpm to prevent water hammer – but it does not apply to this case.

B. **Correct answer**

C. **Incorrect**: cannot cross-connect B train KC flow to the A train ND HX under these conditions – AP/21 specifies that flow must be stopped to the ND HX.
Plausible: There is a precaution to ensure that KC flow is maintained to ND mechanical seal HX for all operating ND pumps

D. **Incorrect**: no need to secure flow the letdown HX

Plausible: this would be required if KC was lost when the plant was at power and NCS temp was higher to prevent flashing in the letdown line.

Bank Question: 593 ***Answer: B***

1 Pt(s) Unit 2 was operating at 99% power when a steamline rupture occurred. Given the following events and conditions:

0200 The operators enter AP/01 (Steam Leak)
0200 The operators reduce turbine load to match Tave and Tref
0201 The operators start a second NV pump and isolate letdown
0202 NLOs start investigating for the location of the steam leak
0203 "P/R OVER POWER ROD STOP" alarm – the RO reports that power has turned and is decreasing.
0204 STA reports pressurizer level is decreasing and cannot be maintained
0205 The turbine building operator reports that the line to the atmospheric dump valves has a steam leak and cannot be isolated

If no safety injection has occurred, pressurizer pressure is maintained and no reactor trip signals are received prior to 0205, which one of the following operator responses is correct?

A. Manually trip the reactor at 0203

B. Manually trip the reactor at 0204

C. Manually trip the reactor at 0205

D. Commence a rapid down power using AP/04 at 0205

Distracter Analysis:

A. **Incorrect**: no requirement to trip the reactor because reactor power has turned and is decreasing. Not approaching the overpower automatic reactor trip at 109% in 2 of 4 channels.
Plausible: shows a power mismatch – reactor power reaches 103% on 1 of 4 PR channels to cause C-2. OMP 4-3 requires the operator to trip when an automatic safeguards action setpoint is approached to avoid challenging the automatic safeguards function.

B. **Correct answer** required to trip under AP/01 (and many other procedures) if you cannot maintain pressurizer level with 2 NV pumps

C. **Incorrect**: required to trip when PZR level cannot be maintained
Plausible: if the candidate thinks that a reactor trip is required because the steam leak was not isolated.

D. **Incorrect**: required to trip when PZR level cannot be maintained

Plausible: this would be the correct answer if not required to trip at 0204.

Bank Question: 594 ***Answer: C***

1 Pt(s) Which one of the following statements correctly describes the operation of the condenser dump valves during a loss of condenser vacuum?

A. **Condenser steam dump valves do not open because the C-7A arming signal is blocked.**

B. **Condenser steam dump valves isolate on a P4 signal when the reactor trips.**

C. **Condenser steam dump valves isolate upon a loss of C-9 signal when condenser pressure drops below 20 inches of vacuum.**

D. **Condenser steam dump valves continue to dump steam to the condenser until condenser reaches atmospheric pressure.**

Distracter Analysis:

A. **Incorrect**: C-7A will arm on a 10% step change in load
Plausible: If the C-7A interlock did not pick up and arm the condenser dump valve, they would not open

B. **Incorrect**: The P4 signal does not close the condenser dump valves
Plausible: A P4 signal would block the atmospheric dump valves

C. **Correct answer**

D. **Incorrect**: The condenser dump valves would close on loss of C-9
Plausible: The condenser dump valves normally open for a reactor trip.

Bank Question: 595 ***Answer: D***

1 Pt(s)

During step 22 of ECA-0.0 (Loss of All AC Power), the operators are directed to depressurize intact S/Gs to 210 psig at the maximum rate if the standby makeup pump cannot be started. What is the basis for depressurizing at the maximum rate instead of a slower more controlled rate?

A. To cooldown as quickly as possible to prevent the loss of pressurizer level.

B. To reduce NC pressure as rapidly as possible to prevent voiding in the upper head region.

C. To maximize natural circulation flow to prevent excessive thermal stratification in the NC loops.

D. To minimize the loss of NC system inventory through the NCP seals.

Distracter Analysis:

A. **Incorrect**: Depressurization at the maximum rate will actually increase the chances of losing pressurizer level as the cooldown causes NC system inventory contraction.
Plausible: ECA-0.0 has a specific note that the cooldown should be continued even of pressurizer level is lost. The loss of PZR level is often a termination criterion for depressurization in other procedures.

B. **Incorrect**: Depressurization at the maximum rate will actually increase the chances of voiding in the upper head region as the cooldown causes NC system inventory contraction.
Plausible: This is an EOP basis for maximizing the cooldown rate in other EOPs – not the depressurization rate.

C. **Incorrect**: Maximizing the cooldown rate will increase the thermal driving head but this prevent, not enhance thermal stratification in the NC loops. Thermal stratification in the NC loops occurs when natural circulation flow is lost due to heat losses to ambient.
Plausible: Maximizing the cooldown rate will increase the temperature difference across the core because the SG temperatures will decrease.

D. **Correct answer**. The standby makeup pump provides flow to the NCP seals. If the pump cannot be started, rapid seal failure will occur.

Bank Question: 596 ***Answer: C***

1 Pt(s)

Units 1 and 2 were operating at 100% power when a fire broke out in the back of the control room. Given the following conditions:

- The fire has not effected or degraded any control systems
- Heavy black smoke is throughout the control room
- The SRO implements AP/17 (Loss of Control Room)

Which one of the following statements correctly describes the operator response to this event?

A. **Immediately trip both unit turbines and reactors and evacuate the control room to the auxiliary shutdown panels.**

B. **Evacuate the control room; trip both unit turbines and reactors on the way to the auxiliary shutdown panel.**

C. **Evacuate the control room: proceed to the auxiliary shutdown panels and trip both unit turbines and reactors when directed by the SROs at the auxiliary shutdown panels.**

D. **Evacuate the control room, proceed to the safe shutdown facility and trip both unit turbines and reactors when directed by the SRO at the standby shutdown facility.**

Distracter Analysis:

A. **Incorrect**: There is a specific caution in AP-17 that warns against tripping the reactor until the SRO is stationed at the ASP and the SRO directs the reactor be tripped.
Plausible: This could be a conservative thing to do before evacuating. Many plants require the reactor to be tripped prior to evacuation.

B. **Incorrect**: There is a specific caution in AP-17 that warns against tripping the reactor until the SRO is stationed at the ASP and the SRO directs the reactor be tripped.
Plausible: If the candidate does not recognize this caution. This would be a convenient and expeditious action to take. This was the old AP-17 response and is now the AP/24 response.

C. **Correct answer**

D. **Incorrect**: Evacuate to the ASP not the SSF
Plausible: The operators would evacuate to the SSF if the fire degraded control systems

Bank Question: 597 ***Answer: D***

1 Pt(s)

Unit 2 was shutdown in Mode 4, cooling down to a refueling outage. The following annunciator lights are provided for identification purposes in answering the question below:

Annunciators on panel 2AD-10:
E-1 = *Upper Cont. Airlock Aux. Door Open*
F-1 = *Upper Cont. Airlock Rx. Door Open*
E-2 = *Lower Cont. Airlock Aux. Door Open*
F-2 = *Lower Cont. Airlock Rx. Door Open*

Annunciators on panel 2AD-13:
A-8 = *VE Door Open*

Approval was given for normal passage into the containment to perform work – no approval has been given for any compensatory measures.

Which one of the following alarm conditions requires corrective action under MSD 585, (Reactor Building Personnel Access and Material Control for Modes 1, 2, 3 and 4)?

A. 2AD-10 E-1 and 2AD-10 E-2 actuated

B. 2AD-10 F-1 and 2AD-10 F-2 actuated

C. 2AD-10 E-1 and 2AD-10 F-2 actuated

D. 2AD-10 E-2 and 2AD-13 A-8 actuated

Distracter Analysis:

A. **Incorrect**: Allowable to have one door open in each airlock
Plausible: 2 airlock doors are open at the same time

B. **Incorrect**: Allowable to have one door open in each airlock
Plausible: 2 airlock doors are open at the same time

C. **Incorrect**: Allowable to have one door open in each airlock
Plausible: 2 airlock doors are open at the same time

D. **Correct answer** although only one containment airlock door is open, the alarm on the VE annulus door – if left open for > 2minutes, requires compensatory security action because an ECCS phase B actuation will auto-start the VE system to establish a negative pressure in the annulus – which can't occur if the annulus door is open.

Bank Question: 598 ***Answer: C***

1 Pt(s)

Unit 1 was responding to an internal flow blockage condition in the core that required a reactor trip and entry into FR-C.2 (Response to Degraded Core Cooling).

Step 15.e of FR-C.2 states:

> *Dump steam to condenser from intact S/Gs while maintaining cooldown rate in NC T-colds less than* 100 °F *in an hour.*

Given the following times and temperatures during the event:

Time	**0200**	**0210**	**0220**	**0230**	**0240**	**0250**	**0300**
NC T-cold °F	557	560	565	558	540	530	520

Time	**0300**	**0310**	**0320**	**0330**	**0340**	**0350**	**0400**
NC T-cold °F	520	495	468	467	444	428	420

Time	**0400**	**0410**	**0420**	**0430**	**0440**	**0450**	**0500**
NC T-cold °F	420	405	390	371	350	320	310

If the cooldown started at 0230, what time did the operators **first** exceed the cooldown limit of FR-C.2?

A. **0240**

B. **0320**

C. **0350**

D. **0450**

Distracter Analysis:

A. **Incorrect**: cooldown rate was 45 °F for 1 hour - did not exceed 100 °F in one hour
Plausible: 108 °F/hr instantaneous cooldown rate for the 10-minute interval exceeded 100 °F/hr. In addition, the applicant has to consider the NCS temperature prior to the trip.

B. **Incorrect**: cooldown rate was 97 °F for 1 hour - did not exceed 100 °F in one hour but came very close.

Plausible: 162 °F/hr instantaneous cooldown rate for the 10-minute interval exceeded 100 °F/hr.

C. **Correct Answer**: cooldown rate was 102 °F in one hour – the instantaneous cooldown rate was only 96 °F/hr.

D. **Incorrect**: although the cooldown rate was 108 °F for 1 hour, the operators exceeded the limit at 0350 – not the first time
Plausible: Exceed both the instantaneous rate and the 1-hour rate. If the applicant misses the correct calculation for answer C, this is the next time when the cooldown rate is exceeded.

Bank Question: 599 ***Answer: D***

1 Pt(s)

Unit 1 was conducting a plant startup at 5% power when a control rod in control bank "A" drops into the core. Given the following events and conditions:

- The reactor remains critical during the recovery of the control rod
- Tave is allowed to drop to 550°F

Which one of the following statements correctly describes the adverse considerations?

A. **Reduced Tave could cause thermal shock on the pressurizer spray nozzle.**

B. **Thermal power best estimate would indicate higher than reactor power (by Power Range N/Is).**

C. **Moderator temperature coefficient (MTC) could exceed the minimum safety analysis value (i.e. become too positive) late in core life.**

D. **Moderator temperature coefficient (MTC) could exceed the minimum safety analysis value (i.e. become too positive) early in core life.**

Distracter Analysis:

A. **Incorrect**: The spray nozzle can withstand much higher temperature differentials
Plausible: One of the tech spec bases for the minimum temperature for critical operations is to ensure that adequate pressurizer spray capability is maintained.

B. **Incorrect**: Thermal power would indicate lower than reactor power due to increased thermalization of the neutrons.
Plausible: If the candidate reverses the logic and the effect.

C. **Incorrect**: MTC would become too positive EARLY in core life.
Plausible: The effect on MTC is correct – to reduce the coefficient

D. **Correct answer**

Bank Question: 601 ***Answer: A***

1 Pt(s) Which one of the following selections correctly describes reflux boiling flow path during a large break LOCA.

Steam enters the ___(1)_____ of S/G U-tubes where the steam condenses and re-enters the core area via the S/G __(2)____.

	(1)	(2)
A.	**hot leg**	**hot leg**
B.	**hot leg**	**cold leg**
C.	**cold leg**	**hot leg**
D.	**cold leg**	**cold leg**

Distracter Analysis:

A. **Correct answer**

B. **Incorrect**: steam returns via the hot leg
Plausible: the first part of the answer is correct

C. **Incorrect**: the steam enters the hot leg
Plausible: the second part of the answer is correct

D. **Incorrect**: cold legs are not affected during reflux boiling
Plausible: psychometric balance

Bank Question: 602 ***Answer: A***

1 Pt(s) Unit 2 is responding to a small break LOCA in ES-1.1 (SI Termination). Given the following plant conditions:

- NCPs tripped
- Pressurizer level is steady
- Only one train of ECCS is injecting
- Loop A temperatures are representative of all 4 loops
- Steam generator pressures are the same as steam header pressure

Which one of the following sets of plant parameters is indicative of natural circulation occurring in the steam generators per enclosure 2 of ES-1.1?

	Time	0200	0205	0210	0215
A.	**Steam Header Pressure (psig)**	**1042**	**1009**	**976**	**945**
	NC System Pressure (psig)	**1968**	**1964**	**1960**	**1958**
	Loop A T-hot (°F)	**579**	**574**	**569**	**564**
	Loop A T-cold (°F)	**548**	**544**	**540**	**536**
B.	**Steam Header Pressure (psig)**	**1042**	**1009**	**976**	**945**
	NC System Pressure (psig)	**1968**	**1972**	**1975**	**1981**
	Loop A T-hot (°F)	**579**	**582**	**585**	**595**
	Loop A T-cold (°F)	**548**	**544**	**540**	**536**
C.	**Steam Header Pressure (psig)**	**1042**	**1047**	**1050**	**1052**
	NC System Pressure (psig)	**1968**	**1964**	**1960**	**1958**
	Loop A T-hot (°F)	**579**	**574**	**569**	**564**
	Loop A T-cold (°F)	**548**	**549**	**548**	**550**
D.	**Steam Header Pressure (psig)**	**1042**	**1047**	**1050**	**1052**
	NC System Pressure (psig)	**1968**	**1972**	**1975**	**1981**
	Loop A T-hot (°F)	**579**	**582**	**585**	**595**
	Loop A T-cold (°F)	**548**	**544**	**540**	**536**

Distracter Analysis: The following conditions support natural circulation:

- S/G pressure stable of decreasing
- T-hot stable or decreasing
- T-cold stable or decreasing
- NC subcooling > 0 - NC pressure may trend up or down.

A. **Correct:** This shows indication of natural circulation flow occurring - decreasing S/G pressure, T-cold at S/G saturation conditions and decreasing, T-hot decreasing.

B. **Incorrect**: T-hot is increasing while steam pressure is decreasing
Plausible: Steam pressure and T-cold are both decreasing

C. **Incorrect**: Steam pressure is increasing and T-cold is tracking along with this trend. Temperature difference is decreasing indicating that heat removal rate is decreasing. This is a classic case of gas binding
Plausible: T-hot is decreasing.

D. **Incorrect**: Steam pressure increasing and T-hot is increasing.
Plausible: T-cold is decreasing

Bank Question: 603 ***Answer: B***

1 Pt(s)

Unit 1 was operating at 100% power when the 1A NV pump failed. Given the following events and conditions:

- 1B NV pump was tagged out of service for maintenance
- The Positive Displacement NV pump was tagged out of service
- The plant is at normal operating temperature, pressure and level
- Normal letdown is in service on the 75 gpm orifice
- Identified leakage is at the Tech Spec Limit
- Unidentified leakage is 1 cc/hr

If **no operator actions are taken**, how much time will elapse before the pressurizer level reaches the low level alarm and the heaters trip?

REFERENCES PROVIDED:
Curve Book Encl 7.38 (PZR Volume vs. Level)
ITS 3.4.13 page 1

A. Less than 45 minutes

B. 45 -55 minutes

C. 55 - 65 minutes

D. Longer than 65 minutes

Distracter Analysis:

The NC coolant will let down to the VCT at a rate of 75 gpm until 17% is reached in the pressurizer. At 17%, the PZR low-level alarm will isolate letdown.

Pressurizer level 55% = 7800 gal
Pressurizer at 17% = 2900 gal

Letdown flow = 75 gpm until isolation at 17% PZR level
Identified leakage = 10 gpm (includes NCP seal leakage to NCDT)
Unidentified leakage is negligible and may be ignored
NCP seal leak off = 4x3 = 12 gpm into VCT

Total flow rate out of the NC system = 97 gpm until letdown isolation
Time to reach 17% PZR level = (7800-2900 gal) / (97 gpm) = 50.5 minutes

A. **Incorrect**: too short
Plausible: if the candidate adds total NCP #1 seal injection flow instead of seal leak off (8x4=32 gpm) to the letdown leak rate 75 gpm and Tech Spec leak rate 10 gpm Time = 41 minutes, misreads the pressurizer level tank curve or makes another mistake.

B. **Correct answer**: Time to reach 17% PZR level = (7800-2900) / (75 + 10 +12 gpm) = 50.5 minutes

C. **Incorrect**: time is too long
Plausible: if candidate forgets to add in max allowable Tech Spec leakage or NCP seal leak off
Time to 17% PZR level = (7800-2900) / (75+10gpm) = 57.6 min
Time to 17% PZR level = (7800-2900)/(75+12 gpm) = 56.3 min

D. **Incorrect**: too long
Plausible: if the candidate does not consider the letdown rate of 75 gpm and only considers the Tech Spec leak rate 10 gpm and/or NCP seal leakoff, misreads the pressurizer level tank curve or makes another mistake.

Bank Question: 604 ***Answer: D***

1 Pt(s)

Unit 1 was operating at 25% power following a reactor startup when intermediate range channel N35 failed. Given the following conditions and events:

- N35 repairs have been made and N35 is being returned to service
- N36 reads1.5x10^{-4} amps
- The N35 "*level trip*" switch was returned to the "*normal*" position

If all power range nuclear instruments and N36 have been properly adjusted, which of the following operator conditions (if any) would cause the reactor to trip?

A. N35 "*Operation Selector*" switch was left in "10^{-3} " position after retesting

B. N35 was significantly under-compensated

C. N35 control power fuses were never reinstalled

D. A reactor trip would not occur

Distracter Analysis: At 10%, the operators manually block the hi IR Rx trip by procedure after P-10 is enabled on 2 of 4 PR detectors > 10%

A. **Incorrect**: - the *operation selector* switch is taken out of the circuit when the *level trip* switch is taken to *normal* – and all IR high flux Rx trips are blocked by P-10.
Plausible: - if the candidate thinks that a test signal can be inserted with *level trip* switch in the *normal* position

B. **Incorrect**: - The IR high flux trip is blocked by P-10
Plausible: - inserting a test signal can cause a trip signal to be generated from the IR drawer – but will not go to SSPS

C. **Incorrect**: - the reactor is above P-10 and although under-compensation of N35 could cause the high flux setpoint to be reached, the IR trips are disabled by P-10
Plausible: - If the candidate does not recognize that a N36 level of 1.5x10^{-4} amps is above P-10

D. **Correct answer**

Bank Question: 605 ***Answer: B***

1 Pt(s)

Unit 2 was operating at 100% power when a reactor trip occurred. The reactor trip caused the initiation of a tube leak in the 2B S/G. The leak rate was 100 gpm. Given the following conditions:

- 2EMF-33 (Condenser Air Ejector Exhaust) alarms in trip 2

If all the automatic features operate as designed (without operator intervention), which one of the following indications will provide the best indication (most sensitive and timely) to confirm that a S/G tube leak has occurred?

A. Comparing S/G feed flow to steam flow mismatch

B. Observing 2EMF-10, 11, 12 and 13 (steamline hi rad)

C. Observing 2EMF-34 (S/G sample line lo range)

D. Observing 2EMF-71, 72, 73, 74 (N16 leakage)

Distracter Analysis:

A. **Incorrect**: Not a sensitive method of comparison – requires large gpm leak rates before this is noticeable.
Plausible: This method will show gross SGTRs

B. **Correct answer**: normally, EMF-71-74 are the most sensitive monitors. But these monitors detect N^{16} γ radiation that has a high energy (7 MeV) γ that only is generated when the reactor is operating at power (requires a neutron flux).

C. **Incorrect**: S/G sample line will isolate at EMF-33 trip 2 – the sample line can only be lined up to 1 S/G at a time. If the leak is not in that S/G, there will be no indication of anything after isolation. Prior to isolation, it may show an increasing trend due to a general build up of activity in the feedwater.
Plausible: This would be a good answer if the automatic isolation did not occur

D. **Incorrect**: most sensitive method as it detects N^{16} γ radiation
Plausible: This was the correct answer for the 1997 NRC exam – when the premise of the question had the reactor was operating at 100% power. In this question, the reactor has tripped and neutron flux has decreased – causing the N^{16} γ to decay off (T½ is 7 seconds)

so that by the time that the steam line monitors see the contents of the S/G, the N^{16} γ has decayed away.

Bank Question: 606 ***Answer: D***

1 Pt(s)

Unit 1 was operating at 100% when a steam generator tube rupture occurred in the 1B S/G. Given the following list of valves in the S/G sample and blowdown systems:

- 1NM-267 S/G Sample HDR RAD Monitor Inlet Isolation
- Blowdown Blowoff Automatic Isolation Valves
  - 1BB-119 = from the 1A S/G
  - 1BB-120 = from the 1B S/G
  - 1BB-121 = from the 1C S/G
  - 1BB-122 = from the 1D S/G
- S/G Sample HDR to Conventional Sample System valves
  - 1NM-269 = from the 1A S/G
  - 1NM-270 = from the 1B S/G
  - 1NM-271 = from the 1C S/G
  - 1NM-272 = from the 1D S/G

Which one of the following statements correctly describes the complete set of valves that would automatically close?

A. **1NM-267**

B. **1NM-267, 1BB-120, 1NM-270**

C. **1NM-267**
1NM-269, 1NM-270, 1NM-271, 1NM-272

D. **1NM-267**
1BB-119, 1BB-120, 1BB-121, 1BB-122,
1NM-269, 1NM-270, 1NM-271, 1NM-272

Distracter Analysis:

A. **Incorrect**: insufficient
Plausible: closes sample header

B. **Incorrect**: insufficient – partial list
Plausible: these valves would isolate sample flow from the 1B S/G with the ruptured tube

C. **Incorrect**: insufficient – partial list – does not include blowdown system valves
Plausible: all sample system (NM) valves isolate

D. **Correct answer** – complete list

Bank Question: 608 ***Answer: D***

1 Pt(s) OP/0/A/6350/001C, (250 VDC Auxiliary Power System) contains the following precaution:

> *"The DC bus ties will normally remain open. They are only to be closed during equalization charges of batteries, or on a loss of a battery or battery charger."*

Which one of the following is the basis for this precaution?

A. **Prevents damage to the battery chargers resulting from both battery chargers simultaneously supplying the same bus at different voltage outputs.**

B. **Prevents overloading one battery if the battery terminal voltages are significantly different which would lead to excessive hydrogen evolution and a possible explosive hazard.**

C. **Ensures both battery chargers are operated in parallel to be able to reach the terminal voltage (~271 VDC) required for an equalization charge.**

D. **Ensure DC channels remain independent of each other and that a fault on one bus does not adversely affect the other bus**

Distracter Analysis:

A. **Incorrect**: The reason is DC channel independence
Plausible: If 2 battery chargers were run in parallel, they could fight each other if they had vastly different voltage output characteristics.

B. **Incorrect**: The reason is DC channel independence
Plausible: If the candidate thinks that batteries with different charge states could fight each other if connected in parallel.

C. **Incorrect**: The reason is DC channel independence - during an equalization charge, the battery being charged is charged from the standby charger due to the high termination voltage required to finish the charge.
Plausible: DC ties are closed during an equalization charge to allow one battery to power both buses – to allow the charged battery to achieve termination voltage which is higher than normal voltage and may damage equipment if applied on the bus.

D. **Correct answer**

Bank Question: 609 ***Answer: C***

1 Pt(s)

The Unit 1 SRO was monitoring a release from the waste monitor tank. Which one of the following alarms would terminate this release automatically?

A. 1EMF-31 (Turbine Bld Sump Disch) trip 2

B. 1EMF-44(L) (Cont Vent Drn Tank Out) trip 2

C. 1EMF-49(L) (Liquid Waste Disch) trip 2

D. 1EMF-50(L) (Waste Gas Disch) trip 2

Distracter Analysis:

A. **Incorrect**: does not monitor the WMT release path
Plausible: would terminate a liquid release from the turbine building sump

B. **Incorrect**: does not monitor the WMT release path
Plausible: would terminate a liquid release from the VUCDT using the same automatic valves as the 1EMF-49 (WP-35 and WP-46)

C. **Correct answer:** closes WP-35 and WP-46 to stop the release.

D. **Incorrect**: does not monitor the WMT release path
Plausible: would terminate a release from the WGTD

Bank Question: 610 ***Answer: C***

1 Pt(s) Unit 2 is conducting a core reload and one hundred thirty fuel assemblies have been loaded into the core. The following data has been recorded upon completion of each assembly reload sequence group:

Reload Sequence Group	No. of Assemblies Added	Total No. of Assemblies	Δρ Added by Sequence Group	Source Range Count Rate After Load
9	15	85	4500	300
10	10	95	2500	360
11	5	100	2000	400
12	5	105	3000	425
13	5	110	2500	460
14	10	120	2000	520
15	10	130	2500	600
16	15	145	4500	
17	15	160	3000	
18	15	175	4500	
19	18	193	4000	

Based on the given data, during which reload group (if any) would you predict that the reactor would reach criticality?

A. 16

B. 18

C. 19

D. The reactor will not reach criticality.

Distracter Analysis: Using the thumb rule that if the count rate doubles, the reactor is ½ way to criticality, the following calculation shows:

300-600 counts – count rate doubles between reload sequences 10 and 15

2500 + 2000 + 3000 + 2500 + 2000 + 2500 = 14500 PCM added to cause count rate to double.
4500 + 3000 + 4500 = 12000 PCM < 14500 – not critical on reloads 16-18
12000 + 4000 = 16000 PCM > 14500 – beyond critical on reload 19

A. **Incorrect**: The reactor will go critical on reload sequence 19
Plausible: based on misapplication of the thumb rule (count rate doubles at criticality).

B. **Incorrect**: The reactor will go critical on reload sequence 19
Plausible: based on CR doubling rule using # of assemblies added rather than actual reactivity of the assemblies.

C. **Correct**:

D. **Incorrect:** The reactor will go critical on reload sequence 19
Plausible: if the candidate adds the $\Delta\rho$ from Group 9 to the reactivity from groups 10-15 to determine the amount of reactivity that it took to double count rate, this is the answer. In addition, this is the expected outcome when conducting a core reload.

Bank Question: 611 ***Answer: D***

1 Pt(s)

Unit 2 has just completed a plant shutdown after a record run when a leak was suspected from the relief valve on the waste gas decay tank that had been placed in service at the start of the shutdown. The tank was empty prior to being placed in service for the shutdown. The SRO directs RP to confirm the existence and determine the location of the suspected leak.

Which one of the following statements would be an effective method of locating the leak in the waste gas system?

A. **Radiological Protection could monitor for alpha particle emission from the radioactive decay of entrained tritium gas.**

B. **Radiological Protection could monitor for flammable levels of Hydrogen gas that accumulate in the WGDTs from purging the VCT.**

C. **Radiological Protection could monitor for ammonia (NH_4) gas from the breakdown of ammonium hydroxide (NH_4OH), which is added to the NC system for pH control.**

D. **Radiological Protection could monitor for beta/gamma emission from the radioactive decay of particulate from long-lived fission product gaseous isotopes.**

Distracter Analysis:

A. **Incorrect**: Tritium gas does not emit alpha particles – will not detect Tritium by monitoring for alpha emission.
Plausible: Tritium builds up in the waste gas system from reactor operations and has a relatively long half-life.

B. **Incorrect**: waste gas system recombiners remove Hydrogen during shutdown prior to storage in a WGDT. The Hydrogen gas concentration is reduced below flammable levels prior to storage in the WGDT to assure that it is safe to release to the environment.
Plausible: Hydrogen gas is removed from the VCT, PRT and NCDT by the waste gas system during shutdown.

C. **Incorrect**: Ammonium Hydroxide is not added to the NC system for chemistry control of pH. It is added to the condensate system
Plausible: Ammonia gas is produced in the NC system by the breakdown of Hydrazine (N_2H_4) when temperature is raised above 250 °F during startup. Ammonia gas builds up in the pressurizer and

enters the waste gas system during degas operations. It is not removed in the waste gas decay system.

D. **Correct answer**: The waste gas system would contain fission product gasses.

Bank Question: 612 ***Answer: A***

1 Pt(s)

A large break LOCA occurred on Unit 1. The operators entered ECA-1.1 (Loss of Emergency Coolant Recirculation) for a complete loss of emergency coolant recirculation due to a blockage in the containment sumps, causing large increase in containment temperatures and pressures. Which one of the following parameter changes would indicate that significant core uncovery was occurring?

A. Source range instruments show a rapid increase

B. Power range instruments show a rapid increase

C. 1EMF-51/52 (Containment TRN A/B High Range) shows a rapid increase

D. 1EMF-9 (Rx Bld Incore Inst Rm) radiation ARM shows a rapid increase

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Distracter Analysis: The loss of containment cooling will cause core uncovery. All of the answers will show indications of degrading conditions inside containment.

A. **Correct answer**

B. **Incorrect**: power range instruments are calibrated to detect high levels of neutron flux and to compensate for gamma flux.
Plausible: An increase in power range output would indicate recriticality, not uncovery.

C. **Incorrect**: EMF-51/2 (Containment High Range Radiation Monitor) would rise as fission products are transported into the containment atmosphere – but would not show a rapid increase when core uncovery occurs. The loss of shielding effect (water shielding EMF-51/2) would be very small compared to the other effects – primarily the amount and location of fission products in the containment atmosphere.
Plausible: EMF-51/2 will increase throughout the accident

D. **Incorrect**: The reactor building incore instrument room is essentially the same area as the Seal Table from the SAMGs. The radiation increase is indicative of a core melt and failure of an incore instrument tube
Plausible: Used by SAMGs

Bank Question: 613 ***Answer: B***

1 Pt(s) Unit 2 was operating at 100% when the following indications occurred:

- Pressurizer level began decreasing
- 1A NV Pump ammeter showed running amps decreased
- Normal letdown was in service

If all automatic control system appeared to operate normally, which one of the following conditions would cause the 1A NV pump running amps to decrease to the minimum value?

A. 2NV-238 (Charging Line Flow Control) failed open

B. 2NV-238 (Charging Line Flow Control) failed closed

C. 2NI-241 (Seal Inj Flow Control) failed open

D. 2NV-241 (Seal Inj Flow Control) failed closed

Distracter Analysis:

A. **Incorrect**: pump amps and pressurizer level would initially increase
Plausible: If the candidate does not understand the charging flow path or does not understand the relationship between pump amps and flow.

B. **Correct answer:** This would block the charging flow path and pumps amps would reduce to minimum as all the charging flow was diverted through NV-150 and NV-151 miniflow valves

C. **Incorrect**: this would increase charging flow, which would increase charging pump amps. It would also increase pressurizer level, which would cut back on charging pump speed to offset the flow increase and stabilize the pressurizer level.
Plausible: If the candidate thought that this could divert charging flow or did not understand the relationship between pump amps and flow.

D. **Incorrect**: NV-241 closing would increase backpressure on the charging system, which would divert more charging flow through the NCP seals. However, the drop in pressurizer level would act to increase the running speed of the charging pump to compensate for the flow reduction. The overall effect would be to increase pump amps as flow would remain the same but at a higher backpressure.

This would also cause letdown isolation due to the loss of NV flow through the regenerative heat exchanger.
Plausible: If the candidate does not consider the effect of the charging pump speed control circuit.

Bank Question: 616 ***Answer: A***

1 Pt(s)

Unit 1 is operating at full power. Given the following events and conditions on the NCPs:

- An OAC alarm indicates loss of KC flow to the to the NCPs.
- The KC supply outside containment isolation valve (1KC-338) is closed.
- Seal injection flow rate to each NCP is 8 gpm.

What are the likely consequences if the operators do not respond to this alarm?

A. The NCPs should operate without KC indefinitely.

B. The NCP motor bearings will overheat causing motor damage.

C. The NCP stator windings will overheat causing motor damage.

D. The NCPs will experience seal failure within 3-5 minutes.

Distracter Analysis: This failure condition is not explicitly covered in the training materials however the candidates should be familiar with the component failure. When KC flow is stopped to the NCPs, this will stop cooling flow to the motor lube oil coolers for the bearings. This will cause lube oil temperatures to exceed allowable values leading to bearing failure.

A. **Incorrect**: Although the NCP thermal barriers can be operated indefinitely as long as seal injection flow is maintained, the loss of KC flow to the motor coolers will cause motor bearing temperatures to overheat.
Plausible: The NCP seals will operate indefinitely without KC

B. **Correct Answer** : Motor bearings will overheat

C. **Incorrect**: NCP Stator windings are cooled by air coolers
Plausible: if the candidate forgets that the NCP stators are air cooled

D. **Incorrect**: seal failure will not occur as long as seal injection is maintained.
Plausible: If seal injection is lost to a pump along with KC flow to the thermal barrier, this would be true.

Bank Question: 617 ***Answer: C***

1 Pt(s) Unit 2 was operating at 90% power after a start-up from a refueling outage. A PORV is found to be leaking and the associated PORV block valve was shut. The PRT was cooled down to the following PRT conditions:

- PRT Level – 65%
- PRT Pressure – 8 psig
- PRT Temperature – 100°F
- Lower Containment Temperature - 118 °F

What actions are required to restore and maintain normal operating conditions to the PRT for the long term?

A. Vent/purge the PRT to containment.

B. Continue to cool the PRT to 90°F.

C. Vent/purge the PRT to the waste gas system.

D. Lower the PRT level to 50%.

Distracter Analysis: The priority of action to reduce pressure is:

1. Cool the PRT
2. Reduce level
3. Purge to waste gas

A. **Incorrect**: cannot be performed at power as the vent valve is inside containment and is inaccessible at power
Plausible: venting to containment would accomplish the required action

B. **Incorrect**: Lower Containment temp is 118 °F. Cooling the tank further would only delay the time when it would heat back up again and require further cooling.
Plausible: Cooling the PRT is the 1st priority of action to be taken to reduce pressure. Cooling will reduce pressure temporarily but will not allow the PRT to reheat to its normal limit of 114°F without getting a high-pressure condition.

C. **Correct**:

D. **Incorrect:** but will lower level below its normal operating band between 64% and 88%.
Plausible: reducing level will reduce pressure ad is the 2nd priority of action to be taken.

Bank Question: 618 ***Answer: B***

1 Pt(s) Unit 1 is recovering from a loss of offsite power in ES-0.2 (Natural Circulation Cooldown). The operators reach step 17 which states:

> ***IF AT ANY TIME*** *cooldown rate must be raised to greater than 50°F in an hour, THEN* ***GO TO*** *EP/1/A/5000/ES-0.3 (Natural Cooldown with Steam Void in Vessel)*

Given the following plant conditions:

- T-hot = 560 °F
- NC Pressure = 1225 psig
- RVLIS = 100% upper range, 64% lower range
- FWST level = 405 inches
- All plant equipment is operating as designed
- Cooldown rate is 47 °F/hr

Which statement correctly describes the condition of the core and the proper procedure flow path?

REFERENCES PROVIDED:
Steam Tables
Curve Book Curves 1.10B, 1.10C, 1.10D

A. The core is in a superheated condition – transition to ES-0.3 to continue the cooldown

B. The core is in a superheated condition – remain in ES-0.2

C. The core is in a subcooled condition - transition to ES-0.3 to continue the cooldown

D. The core is in a subcooled condition – remain in ES-0.2

Distracter Analysis: ES-0.2 does not provide specific guidance for this transitional step. The EOP bases for this step is to make the transition if:

- There is limited condensate storage
- No CRDM fans are operating.

The entry conditions for ES-03 are:

- If cooldown rate must be raised above setpoint
- If reactor vessel indicates not full and it is determined that depressurization must occur

When evaluating transitions, the operators are **required** to use the curves in the Data Book instead of steam tables. These curves include an instrument error offset of 20 °F. If the operators refer to steam tables to evaluate subcooling, the core conditions are subcooled (ignoring instrument error). If they refer to the curve, it will show that the core is in the saturation region – which means it is below the saturation curve and below the subcooled region – which means it is superheated.

A. **Incorrect**: Transition to ES-0.3 is not required – no indications of void in the core
Plausible: The core is in the "saturation region" of curve 1.10B – which means that it is below the saturation curve and in the superheat region of instrument error is considered. If the candidate thought that RVLIS readings showed that a void was forming in the core, transitioning would be correct.

B. **Correct**:

C. **Incorrect**: The core is in the superheat condition and there is no requirement to transition to ES-0.3.
Plausible: If the candidate used the steam tables to evaluate the core saturation conditions. If the candidate thought that RVLIS readings showed that a void was forming in the core.

D. **Incorrect**: The core is in the superheat condition
Plausible: Remaining in ES-0.2 is correct

Bank Question: 620 ***Answer: C***

1 Pt(s) Unit 1 was operating at 100% power when a loss of VI system air pressure occurred. Which one of the following statements correctly describes the condition of the upper containment airlock seals?

A. The seals will slowly depressurize. They can be manually re-inflated using SA system air pressure.

B. The seals will slowly depressurize. They can be manually re-inflated using the VB system

C. The seals will remain pressurized by an air supply from local air tanks.

D. The seals will remain pressurized by a backup line from the SA system.

Distracter Analysis:

A. **Incorrect**: The seals will not depressurize
Plausible: VI provides air to the seal supply

B. **Incorrect**: The seals will not depressurize
Plausible: VI provides air to the seal supply

C. **Correct**:

D. **Incorrect**: There is no backup connection to the SA system
Plausible: The seals will remain pressurized on a loss of VI.

Bank Question: 621 ***Answer: B***

1 Pt(s)

Unit 2 was responding to a large break LOCA in E-1 (Loss of Reactor or Secondary Coolant). Given the following events and conditions:

- The 4160/600 VAC supply transformer to load center 2ELXD failed
- Motor control center 2EMXD was deenergized

Which one of the following statements correctly describes the actions needed to start containment air return fan 2B?

A. Transfer 2ELXD to transformer 2ELXB

B. Transfer 2ELXD to transformer 2ELXF

C. Transfer 2EMXD to transformer 2ELXF

D. Manually start air return fan 2B

Distracter Analysis: Containment air fan 2B is used to mix the containment atmosphere during a LOCA to enhance removal of Iodine and other fission products by containment spray. Fan 2B is powered from motor control center 2EMXD through load center 2ELXD. The fan is automatically started by a safety injection signal.

A. **Incorrect**: Prohibited by a Kirk Key interlock
Plausible: This would physically repower the fan

B. **Correct**:

C. **Incorrect**: Not physically possible
Plausible: If the transfer could be physically done, it would repower fan 2B

D. **Incorrect**: Will not start – no power to the fan
Plausible: If the candidate does not determine which motor control center powers containment air return fan 2B

Bank Question: 622 ***Answer: D***

1 Pt(s)

Unit 2 was operating at 5% power during a plant startup when a total loss of AC power (station blackout) occurred. Given the following events and conditions:

- The plant was operating within normal limits and bands
- All protection systems operated as designed
- All emergency diesel generators failed to start
- No safety injection occurred
- No operator action was taken

Which one of the following statements correctly describes the response of the reactor trip system?

A. **No automatic reactor trip would occur and the reactor would remain critical.**

B. **The shunt coils in the reactor trip and bypass breakers would energize and a reactor trip would occur.**

C. **The under-voltage and shunt coils in the reactor trip breakers would energize and a reactor trip would occur.**

D. **The CRDMs would deenergize and the rods would drop into the core.**

Distracter Analysis: The loss of the safety bus would cause the rod drive MG sets to lose power and power would be lost to the CRDMs. Below 10% power, the at-power reactor trips are bypassed so no automatic trip would occur. SSPS would remain energized from the 120 VAC Instrument bus but would not get a trip signal.

A. **Incorrect**: CRDM power would be lost from the rod drive MG sets
Plausible: If the candidate thinks that the rod drive MG sets are powered from the 120 VAC instrument bus or a DC bus.

B. **Incorrect**: No trip signal would be generated below 10% power, P-7.
Plausible: SSPS remains energized and could generate a trip signal

C. **Incorrect**: No trip signal would be generated below 10% power, P-7.
Plausible: SSPS remains energized and could generate a trip signal.

D. **Correct**:

Bank Question: 623 ***Answer: A***

1 Pt(s) Unit 2 was operating at 100% power when an alarm was received on the 2B NCP standpipe level. Which one of the following statements correctly describes the cause of the standpipe level alarm?

A. A high standpipe level indicates excessive leakoff through the #2 seal.

B. A high standpipe level indicates reduced leakoff through the #3 seal.

C. A low standpipe level indicates excessive leakoff through the #2 seal.

D. A low standpipe level indicates reduced leakoff through the #3 seal.

Distracter Analysis:

A. **Correct**:

B. **Incorrect**: the standpipe level maintains a backpressure on the #2 seal – reduced leakoff by the #3 seal has no effect on standpipe level as this leakoff goes directly to the RCOT
Plausible: If the candidate thinks that the leakoff from the #3 seal effects standpipe level.

C. **Incorrect**: excessive leakoff from the #2 seal would lead to a higher standpipe level as the standpipe maintains a backpressure on the #2 seal.
Plausible: if the candidate thinks that excessive leakoff would cause standpipe level to drop or thinks that the standpipe is on the #3 seal..

D. **Incorrect**: reduced leakoff from the #3 seal would not lead to a lower standpipe level as the standpipe maintains a backpressure on the #2 seal.
Plausible: If the candidate thinks that flow from the #3 seal goes to the standpipe

Bank Question: 624 ***Answer: D***

1 Pt(s)

Unit 2 was operating at 100% power with train B components in service. If a high strainer differential pressure alarm occurs on the 2B RN pump, what statement describes the RN system alignment upon completion of all automatic actions?

A. **2A RN pump is running from the SNSWP**
2A RN strainer is in service

B. **2A RN pump is running from the low level intake**
2B RN strainer is in backwashing

C. **2B RN pump is running from the SNSWP**
2B RN strainer is in backwashing

D. **2B RN pump is running from the low level intake**
2B RN strainer is backwashing

Distracter Analysis:

A. **Incorrect**: The low D/P alarm causes the 2B strainer to backwash
Plausible: IF the candidate thinks that the low D/P alarm on the B strainer causes the running RN trains to shift to the A train

B. **Incorrect**: the RN pumps do not shift for a high D/P alarm
Plausible: the RN pumps would be manually shifted to the SNSWP for a low suction pressure alarm, not a high D/P alarm

C. **Incorrect**: the suction does not shift to the SNSWP
Plausible: the 2B RN strainer backwashes

D. **Correct**:

Bank Question: 625 ***Answer: B***

1 Pt(s) Unit 1 was operating at 5% power following a reactor startup after a refueling outage. Given the following conditions and events:

- A mixed bed demineralizer that had been isolated at the end of the last fuel cycle was placed in service
- T-ave = 558 °F
- All systems are aligned normally for the existing plant conditions

What will be the effect (if any) on T-ave?

A. T-ave will increase due to the exchange of Lithium ions

B. T-ave will increase due to the exchange of boric acid

C. T-ave will decrease due to the exchange of boric acid

D. T-ave will not change

Distracter Analysis: Placing a mixed bed demineralizer in service will change reactivity by exchanging boric acid (borate ions) for OH ions. The mixed bed demineralizer was exposed to low boric acid concentrations at the end of core life prior to the shutdown so it reached equilibrium saturation conditions of very low concentrations of boric acid. When placed in service at beginning of core life, it will exchange boric acid for OH ions and reduce the boric acid concentration in the NC system.

A. **Incorrect**: T-ave will increase due to exchange of boric acid
Plausible: pH may increase but not temp due to the exchange of Li ions for hydroxyl ions depending on the saturation state of the resin.

B. **Correct Answer**:

C. **Incorrect**: T-ave will increase due to exchange of boric acid
Plausible: If the candidate reverses the effect of this action

D. **Incorrect**: T-ave will increase due to exchange of boric acid
Plausible: This is the correct answer for cation bed resin which exchanges Lithium for OH - but not boric acid

Bank Question: 627 ***Answer: C***

1 Pt(s) Unit 1 was operating at 100% power when panel board 1EKVB was unintentionally deenergized. Which one of the following lists of ESS loads was deenergized.

A. **Process Protection Channel I**
Safeguards Test Cabinet Train A
SSPS Channel I (Trains A&B)
SSPS Train A Output Cabinet
Auxiliary Safeguards Cabinet Train A

B. **Process Protection Channel IV**
Safeguards Test Cabinet Train B
SSPS Channel IV (Trains A&B)
SSPS Train B Output Cabinet
Auxiliary Safeguards Cabinet Train

C. **Process Protection Channel II**
SSPS Channel II (Trains A & B)

D. **Process Protection Channel III**
SSPS Channel III (Trains A & B)

Distracter Analysis:

A. **Incorrect**: powered from EKVA
Plausible: if the candidate does not know the power supply
B. **Incorrect**: powered from EKVD
Plausible: if the candidate does not know the power supply
C. **Correct**:
D. **Incorrect**: powered from EKVC
Plausible: if the candidate does not know the power supply

Bank Question: 628 ***Answer: C***

1 Pt(s)

Unit 2 was responding to a small-break LOCA in E-1 (Loss of Reactor or Secondary Coolant). Given the following conditions:

- Containment pressure = 0.7 psig (at peak pressure for the event)
- 2ETA was deenergized due to a bus fault
- The VI header inside containment was depressurized and isolated due pipe rupture
- The VI system outside containment remained pressurized

Which one of the following statements correctly describes the positions of valves 2RV-79A and 2RV-80B?

REFERENCES PROVIDED
Station Drawing MCFD-1604-03.00 Flow Diagram of RV

A. 2RV-79A is open, 2RV-80B is open

B. 2RV-79A is shut, 2RV-80B is open

C. 2RV-79A is open 2RV-80B is shut

D. 2RV-79A is shut 2RV-80B is shut

Distracter Analysis: These valves are RV containment isolation valves. They are air operated and will close on a high-high containment pressure signal (phase B isolation). They fail closed upon loss of operating air pressure. There are no backup nitrogen accumulators to provide operating pressure (as with the PORVs) even though they are safety-related valves.

In this question, the valves will not auto close because pressure remains below the phase B actuation point (3.0 psig) but they will close due to a loss of operating air pressure.

A. **Incorrect**: the valves are shut due to the loss of VI air pressure
Plausible: if the candidate thinks that they fail open or that they are electrically operated

B. **Incorrect**: the valves are shut due to the loss of VI air pressure
Plausible: if the candidate thinks that they are electrically operated and fail shut

C. **Correct Answer**: 2RV-79A remains open without a phase B signal and 2RV-80B fails closed due to the loss if VI pressure inside containment.

D. **Incorrect**: 2RV-79A remains open because no phase B isolation was generated
Plausible: if the candidate thinks that a phase B signal was generated or if he does not recognize that VI pressure is maintained outside of containment - or if he thinks 2RV-79A is powered from 1ETA and fails closed.

Bank Question: 629 ***Answer: A***

1 Pt(s)

With Unit 1 was operating at 75% power with rods in automatic control when turbine load drops 10%. Which of the following correctly indicates the change in plant parameters when the transient is complete?

A. T-ave decreases approximately 2 to 3°F due to decreased Tref.

B. Tave stays the same due to automatic rod motion.

C. Tave increases approximately 2 to 3 °F due to rod motion.

D. Tave stays the same due to decreased Tref.

Distracter Analysis: T-ave is ramped from 557 °F at 0% power to 585 °F at 100% power. A 10% drop in load causes a 10% reduction in T-ave within the operating band - so T-ave would drop by 10% of 28 °F or ~ 3°F.

A. **Correct**:

B. **Incorrect**: Tave decreases 10%.
Plausible: If you think rod control compensates for load change by maintaining T-ave

C. **Incorrect**: Tave decreases 10%.
Plausible. If you think Tref change reflects load increase.

D. **Incorrect:** Tave decreases 10%.
Plausible: If you think Tref decreases and compensates for load decrease.

Bank Question: 630 ***Answer: A***

1 Pt(s) Which of the following statements correctly describes the major effect of a failure of a large number of lower ice condenser doors to open for an unisolable main steam line break with an associated SGTR (tube rupture on the faulted S/G) accident inside containment.

A. Containment peak pressure would be higher and would be achieved sooner in the event.

B. Containment peak pressure would be higher but would be achieved later in the event.

C. Containment sump water inventory would not be adequate after shift to recirculation mode.

D. Containment sump water inventory would not be adequate to maintain long-term subcriticality during the cooldown.

Distracter Analysis: This question is a variation on a bank question – which is normally asked from the perspective of a LBLOCA. However, the analysis is also true for a faulted ruptured S/G inside containment. The difference in these events is that the MSL break is a HELBIC and the SGTR is a small break LOCA – so this would cause the water in the containment sump to include the secondary water, which is unborated. In addition, the amount of water from the S/G would be less than the water from a LBLOCA.

A. **Correct**:

B. **Incorrect**: containment pressure would peak sooner
Plausible: if the candidate thought that the ice condenser melt was delayed due to the door not opening – causing the release of the cold water to occur later

C. **Incorrect**: sufficient water is added from the FWST during the injection phase
Plausible: the ice water would not be immediately available as melting would be delayed. In addition, the amount of water that enters the sump from the S/G would be less than the water from a LBLOCA. A SGTR is essentially a SBLOCA and water would not be released from the core at a rapid rate to add to the sump before swap over was required.

D. **Incorrect**: the FWST would provide sufficient borated water to maintain shutdown throughout the process

Plausible: release of the borated water from the ice melt would be delayed and thus would not be available until after melting. In addition, the water from the S/G is unborated.

Bank Question: 631 ***Answer: D***

1 Pt(s) The Unit 2 NV system cold leg flow path balance test procedure throttles high pressure injection flow between a minimum value to limit ____(1)____ and a maximum value to limit ____(2)____.

	(1)	(2)
A.	**Pump overheating**	**pipe erosion**
B.	**Break flow**	**pipe erosion**
C.	**Pump overheating**	**pump runout**
D.	**Break flow**	**pump runout**

Distracter Analysis:

A. **Incorrect**: pump overheating is not a problem at the higher flow rates and pipe erosion is not a limiting problem for this system
Plausible: low flow rates can cause pump overheating and high flow rates can cause pipe erosion

B. **Incorrect**: safe-end erosion is not a limiting factor
Plausible: break spillage is the basis for the minimum throttle limit

C. **Incorrect**: pump overheating is not a problem at the higher flow rates and pipe erosion is not a limiting problem for this system
Plausible: pump runout is the basis for the higher throttle setting

D. **Correct**:

Bank Question: 632 ***Answer: D***

1 Pt(s) Unit 1 was responding to a station blackout in ECA-0.0 (Loss of all AC Power). What pressurizer heaters are available to control reactor pressure?

A. Group A backup heaters can be controlled from the SSF.

B. Group B backup heaters can be controlled from the ASP.

C. Group C backup heaters can be controlled from the ASP.

D. Group D backup heaters can be controlled from the SSF.

Distracter Analysis: Only about 10% of Group D heaters will be available under these conditions

A. **Incorrect**: Group A has no power and is controlled from the ASP.
Plausible: If confuses group A with D and keys on SSF as the power source.

B. **Incorrect**: Group B has no power.
Plausible: If associates group B with its correct local control station.

C. **Incorrect**: Group C has no power and is controlled from the MCB.
Plausible: If doesn't know power supplies and chooses based on logical extension of normal pressure control.

D. **Correct**:

Bank Question: 633 ***Answer: B***

1 Pt(s)

Unit 2 was operating at 100% power when the pressurizer spray valve failed open. Given the following conditions:

- PZR Channel Select is in 1-2 position
- PZR Pressure Control in AUTO
- NCS Pressure is 2245 psig

Which one of the following describes the response of the PZR pressure control system to these conditions?

A. **PZR pressure does not decrease because the spray valves will not open below 2260 psig**

B. **PZR pressure decreases to 1945 psig where the RPS reactor trips.**

C. **PZR pressure decreases to 2210 psig, where the backup heaters take control of pressure.**

D. **PZR pressure decreases to 2185 psig where the spray line block valves close.**

Distracter Analysis:

A. **Incorrect**: the spray valves open and pressure decreases.
Plausible: based on a misunderstanding of how the controller works.

B. **Correct**:

C. **Incorrect**: Backup heaters come on but do not control pressure.
Plausible: based on misconception of an IPE system "degas" mode.

D. **Incorrect**: There are no block valves.
Plausible: based on possible confusion between spray and PORV systems.

Bank Question: 634 ***Answer: A***

1 Pt(s) Which one of the following statements correctly describes the response of the turbine driven CA pump if turbine speed exceeds 4500 rpm?

A. A mechanical flyweight assembly unlatches a trip hook on the turbine stop valve

B. A mechanical flyweight assembly unlatches a trip hook on the turbine governor valve

C. The Woodward governor will generate a signal that trips the turbine stop valve tripping latch assembly

D. The Woodward governor will generate a trip signal that trips the turbine governor valve

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Distracter Analysis:

A. Correct Answer:
B. Incorrect: turbine stop valve trips
Plausible: the mechanical flyweight assembly causes the trip
C. Incorrect: the Woodward governor does not cause the trip
Plausible: the turbine stop valve is the valve that actually trips
D. Incorrect: the Woodward governor does not cause the trip
Plausible: this is the mechanism by which the turbine speed is limited

Bank Question: 635 ***Answer: B***

1 Pt(s)

Unit 1 was conducting refueling in Mode 6. RP requested the control room operator to independently verify the adjustment of the trip 2 setpoint for 1EMF-16 (Containment Refueling Bridge) area radiation monitor.

If the trip 2 setpoint was required to be set at ½ decade above the background and background radiation levels were 3.0 mR/hr, what is the correct value for the trip 2 setpoint?

A. 6 mR/hr

B. 9 mR/hr

C. 12 mR/hr

D. 15 mR/hr

Distracter Analysis: ½ decade is 3.16x background above background that is commonly considered 3x background for setting trip 2 setpoints.

A. **Incorrect**: too low - the correct value is 3x3 = 9 mR/hr
Plausible: if the candidate thinks that ½ decade = twice background levels

B. **Correct Answer**: 3x3 mR/hr = 9 mR/hr

C. **Incorrect**: too high - 9 mR/hr is the correct answer
Plausible: if the candidate adds the background level to the calculation – i.e. 3x3+3 = 12 mR/hr

D. **Incorrect**: too high – 9 mR/hr is correct
Plausible: if the candidate thinks that ½ decade is 5 x background (as one decade is ½ of 10 X background) 5x3 = 15

Bank Question: 637 ***Answer: D***

1 Pt(s) Unit 1 was conducting a reactor startup. Given the following conditions:

- All shutdown rod banks have been fully withdrawn at 222 steps
- Control bank "A" rods are being withdrawn at 80 steps.
- The RPI Urgent Failure Annunciator alarms.

Which of the following conditions would cause this alarm?

A. **A control bank "A" rod is misaligned from its bank position by 8 steps.**

B. **Data "A" failure has occurred on one or more rods in shutdown bank "A".**

C. **The rod control bank overlap unit has detected an improper rod step sequence.**

D. **A rod in shutdown bank "C" has dropped into the bottom of the reactor.**

Distracter Analysis:

A. **Incorrect**: requires a position deviation > 12 steps - and will not withdraw control bank C at this point - overlap unit restricts
Plausible: if candidate does not know the criteria for deviation alarm

B. **Incorrect**: Will not cause an urgent failure alarm
Plausible: Will cause a non-urgent failure alarm

C. **Incorrect**: will not cause an urgent failure
Plausible: could confuse with a Non-Urgent Failure alarm.

D. **Correct Answer**:

1 Pt(s) Unit 1 was operating at 60% power. Given the following events and conditions:

- Pressurizer pressure decreased to 1940 psig.
- The SSPS train A low PZR pressure trip logic relay failed to actuate.

What effect would this failure have on the function of the reactor protection system?

A. **The reactor would not trip because the Train A logic relay would not remove power from the UV coil for RTA.**

B. **The reactor would not trip because the Train B logic relay would not remove power from the UV coil for RTA.**

C. **The reactor would trip because the Train B logic relay would remove power from the UV coil for RTB.**

D. **The reactor would trip because the Train B logic relay would remove power from the UV coil for RTA.**

Bank Question: 638 ***Answer: C***

1 Pt(s) Unit 1 was operating at 60% power. Given the following events and conditions:

- Pressurizer pressure decreased to 1940 psig.
- The SSPS train A low PZR pressure trip logic relay failed to actuate.

What effect would this failure have on the function of the reactor protection system?

A. The reactor would not trip because the Train A logic relay would not remove power from the UV coil for RTA.

B. The reactor would not trip because the Train B logic relay would not remove power from the UV coil for RTA.

C. The reactor would trip because the Train B logic relay would remove power from the UV coil for RTB.

D. The reactor would trip because the Train B logic relay would remove power from the UV coil for RTA.

Distracter Analysis:

A. **Incorrect**: The reactor will trip.
Plausible: based on misunderstanding of RPS redundancy. SSPS Train A had the failure so it makes sense that this could potentially fail to cause the trip

B. **Incorrect**: The reactor will trip.
Plausible: based on misunderstanding of RPS redundancy.

C. **Correct**:

D. **Incorrect:** The Train B logic does not affect RTA.
Plausible: based on misunderstanding of RPS redundancy. If the candidate thinks that SSPS Train B opens RTA.

Bank Question: 638 ***Answer: C***

1 Pt(s) Unit 1 was operating at 60% power. If the SSPS train A low PZR pressure trip logic relay fails to respond to a valid sensor trip signal, what effect would this failure have on the function of the reactor protection system?

A. The reactor would not trip because the Train A logic relay would not remove power from the UV coil for RTA.

B. The reactor would not trip because the Train B logic relay would not remove power from the UV coil for RTA.

C. The reactor would trip because the Train B logic relay would remove power from the UV coil for RTB.

D. The reactor would trip because the Train B logic relay would remove power from the UV coil for RTA.

Distracter Analysis:

A. **Incorrect**: The reactor will trip.
Plausible: based on misunderstanding of RPS redundancy. SSPS Train A had the failure so it makes sense that this could potentially fail to cause the trip

B. **Incorrect**: The reactor will trip.
Plausible: based on misunderstanding of RPS redundancy.

C. **Correct**:

D. **Incorrect:** The Train B logic does not affect RTA.
Plausible: based on misunderstanding of RPS redundancy. If the candidate thinks that SSPS Train B opens RTA.

Bank Question: 639 ***Answer: D***

1 Pt(s)

Unit 1 is operating at 35% power with all systems in normal alignment for plant conditions. Given the following plant conditions:

- The steam pressure input for Steam Flow for "B" Steam Generator fails HIGH (1300psig).

If the operators take no action, which one of the following describes the steam generator level control system initial response to the steam pressure failure until the level error signal counteracts the SM/CF flow mismatch?

A. Indicated steam flow decreases, due to the decrease in density compensation, and S/G Level will decrease.

B. Indicated steam flow decreases, due to the increase in density compensation, and S/G Level will increase.

C. Indicated steam flow increases, due to the decrease in density compensation, and S/G Level will decrease.

D. Indicated steam flow increases, due to the increase in density compensation, and S/G Level will increase.

Distracter Analysis:

A. **Incorrect**: indicated steam flow increases and S/G level increases
Plausible: If the candidate reverses the effect of steam pressure on density compensation.

B. **Incorrect**: indicated steam flow increases and S/G level increases
Plausible: apparent steam density does correctly increase if steam pressure increases – if the candidate reverses the effect of density compensation on indicated steam flow

C. **Incorrect**: density compensation increases and S/G level increases
Plausible: if the candidate reverses the effect of steam pressure on apparent steam density – and reverses the effect of a density decrease on steam flow – then the answer follows as steam flow increase does cause S/G level to increase

D. **Correct**:

Bank Question: 640 ***Answer: A***

1 Pt(s) Unit 2 is responding in E-1 (Loss of Reactor or Secondary Coolant) to a LOCA inside containment. Given the following conditions:

- Phase B containment isolation actuated
- Containment pressure remained above 3 psig
- The FWST level decreased to 20 inches.

Which of the following best describes the steps necessary to prevent damaging the NS pumps?

A. Reset NS, stop the NS pumps.

B. Reset CPCS, stop the NS pumps.

C. Reset containment phase B isolation, stop the NS pumps.

D. Override CPCS, stop the NS pumps.

Distracter Analysis:

A. **Correct**:

B. **Incorrect**: There is no CPCS reset.
Plausible: based on confusion between NS and CPCS actuation logic.

C. **Incorrect**: phase B will not reset - > 3 psig
Plausible: based on confusion between phase B and NS actuation logic – can reset NS.

D. **Incorrect**: Overriding CPCS will not reset phase B or NS.
Plausible: based on confusion with CPCS failure actions.

Bank Question: 642 ***Answer: D***

1 Pt(s)

Unit 1 was operating at 100% power when a design basis earthquake caused a loss of all AC power (station blackout). Given the following events and conditions:

- The suction line to the 1A KF pump sheared during the earthquake
- Spent fuel pool (SFP) makeup was aligned from the FWST to compensate for any loss of SFP level as required.
- FWST level was at 300 inches.
- The operators entered ECA-0.0 (Loss of All AC Power)

Which of the following events would cause spent fuel pool level to continue to decrease?

A. The ruptured suction line on the 1A KF pump caused the water to be siphoned out of the spent fuel pool.

B. The loss of containment and spent fuel pool ventilation fans caused a change in the differential pressure between the spent fuel pool and the reactor cavity.

C. The FWST gravity makeup line to the spent fuel pool was not properly isolated and water has been siphoned back into the FWST.

D. The standby makeup pump was in operation.

Distracter Analysis:

A. **Incorrect**: The suction line inlet is very close to the surface of the spent fuel pool and a rupture in the line would not cause an appreciable drop in level
Plausible: If the suction line inlet was not designed to prevent this event, it would drain the pool

B. **Incorrect**: The reactor cavity isolation devices are closed.
Plausible: based on misunderstanding plant-operating conditions. The spent fuel pool level will change if there is a change in differential pressure between containment and the refueling building – has occurred in the past.

C. **Incorrect**: FWST makeup does not reverse siphon until below level is below 100 inches.

Plausible: based on misunderstanding the SFP/FWST design criteria.

D. **Correct**: The standby makeup pump takes suction from the spent fuel pool

Bank Question: 643 ***Answer: D***

1 Pt(s) Unit 1 was operating at 100% power when a steam generator tube rupture occurred in the 1B S/G. If the operators respond properly in E-3 (Steam Generator Tube Rupture) and isolate the 1B S/G, which of the following conditions are indicative of successful isolation prior to commencing the initial cooldown of the NC system?

A. **S/G level decreases as S/G water flows back through the break into the NC system.**
S/G pressure decreases as steam generator pressure equalizes with NC system pressure.

B. **S/G level decreases as S/G water flows back through the break into the NC system.**
S/G pressure increases as steam generator pressure equalizes with NC system pressure.

C. **S/G level increases as NC system coolant water flows through the break into the S/G.**
S/G pressure decreases as steam generator pressure equalizes with NC system pressure.

D. **S/G level increases as NC system coolant water flows through the break into the S/G.**
S/G pressure increases as steam generator pressure equalizes with NC system pressure.

Distracter Analysis:

A. **Incorrect**: S/G level increases and S/G pressure increases
Plausible: provided for psychometric balance

B. **Incorrect**: S/G level increases
Plausible: S/G pressure increases correctly

C. **Incorrect**: S/G pressure increases
Plausible: S/G level increases correctly

D. **Correct Answer**:

Bank Question: 644 ***Answer: A***

1 Pt(s)

Unit 2 has tripped due to instrument technician error during a surveillance test. The moisture separator reheaters (MSRs) did not reset. Assuming no operator action, what effect would this failure have on the plant response to this transient?

A. The NCS would be overcooled because the main steam supply to the MSRs would not isolate.

B. Safety injection will actuate on low SG pressure because MSR steam supply valve 2SM-15 fails to close.

C. Safety injection will actuate on low SG pressure because the main steam supply to the MSRs would not isolate.

D. The NCS would be overcooled because MSR steam supply valve 2SM-15 fails to close.

Distracter Analysis:

A. **Correct**:

B. **Incorrect**: Unit 2 does not have low SG pressure safety injection.
Plausible: The cooldown will continue until MSIV closure on low SG pressure. A plausible answer for Unit 1.

C. **Incorrect**: Unit 2 does not have low SG pressure safety injection.
Plausible: The cooldown will continue until MSIV closure on low SG pressure. A plausible answer for Unit 1 which does have low SG pressure safety injection.

D. **Incorrect**: 2SM-15 closure is a manual action if the MSRs do not reset.
Plausible: based on a misunderstanding of the MSR system configuration.

Bank Question: 648 ***Answer: B***

1 Pt(s) Which one of the following correctly describes the normal loading of the 125VDC vital battery chargers?

A. **(1) Battery on "charge", (1) 125VDC DC distribution center, (1) 125VDC DC panel board, (1) 120VAC AC static inverter.**

B. **(1) Battery on "float", (1) 125VDC DC distribution center, (2) 125VDC DC panel boards, (2) 120VAC AC static inverters.**

C. **(1) Battery on "float", (1) 125VDC DC distribution center, (1) 125VDC DC panel board, (1) 120VAC AC static inverter.**

D. **(1) Battery on "charge", (2) 125VDC DC distribution centers, (2) 125VDC DC panel boards, (2) 120VAC AC static inverters.**

Distracter Analysis:

A. **Incorrect**: There are 2 panel boards and inverters, one per unit.
Plausible: based on misunderstanding of float, and a one unit only perspective.

B. **Correct**:

C. **Incorrect**: There are 2 panel boards and inverters, one per unit.
Plausible: based on one unit only perspective.

D. **Incorrect**: There is only one distribution center normally aligned.
Plausible: based on a misunderstanding of float, and battery capacity versus normal alignment.

Bank Question: 651 ***Answer: D***

1 Pt(s) Which of the following actions occur when an emergency diesel generator reaches 95% of rated speed during an emergency start?

A. Generator field flashed and voltage and frequency automatically controlled.

B. Low lube oil pressure trip reinstated and starting air secured.

C. Generator field flashed, and starting air secured.

D. Low lube oil pressure trip reinstated and voltage and frequency automatically controlled.

Distracter Analysis:

A. **Incorrect**: Field flash occurs at 40%.
Plausible: based on a logical progression for auto start logic.

B. **Incorrect**: Starting air is secured at 40%.
Plausible: based on misunderstanding of starting air reset setpoint.

C. **Incorrect**: Field flash and starting air secured occur at 40%.
Plausible: based on misunderstanding of correct setpoint.

D. **Correct**:

Bank Question: 653 ***Answer: C***

1 Pt(s)

During preparation for a waste gas release, the pre-release surveillance test of 1EMF-50(L) (Waste Gas Disch (Lo Range)) revealed that there was no response to the source check. IAE reported that the scintillation detector had failed. What action is necessary to begin releasing the waste gas decay tank?

A. Take manual grab samples of the waste gas decay tank prior to any gaseous waste release.

B. Source check the GM detector for 1EMF-50(H) (Waste Gas Disch (Hi Range)) prior to any gaseous waste release.

C. Verify 1EMF-36(L) (Unit Vent Gas (Lo Range)) is in service prior to any gaseous waste release.

D. Repair 1EMF-50(L) prior to any gaseous waste release.

Distracter Analysis:

A. **Incorrect**: Either EMF-50 or 36 must be in service or unit vent samples taken.
Plausible: based on misunderstanding of the sample location.

B. **Incorrect**: 1EMF-50(H) does not provide automatic protection for the waste gas release path - the correct EMF is 1EMF-50(L) - which is NOT a GM detector
Plausible: based on 1 of 2 detectors failed.

C. **Correct**:

D. **Incorrect**: EMF-36 can substitute for EMF-50.
Plausible: based on not knowing any others are correct.

Bank Question: 654 ***Answer: A***

1 Pt(s) Which one of the following interlocks is designed to prevent a water hammer in the RC piping if the RC pumps trip?

A. The vacuum breaker valves automatically open if all RC pumps trip.

B. There is a 45 second time delay before pump discharge valves close to allow coast down.

C. The pump discharge valves remain open until flow decreases below a preset value.

D. The pump discharge valves close over 120 seconds while the pump is coasting down.

Distracter Analysis:

A. **Correct**:
B. **Incorrect**: There is no time delay.
Plausible: based on a logical alternative.
C. **Incorrect**: The valves close regardless of RC flow rate.
Plausible: based on a logical alternative.
D. **Incorrect**: Not designed to preclude water hammer.
Plausible: based on a logical alternative.

Bank Question: 655 ***Answer: B***

1 Pt(s)

On May 19th, the NLO was directed by the unit supervisor to perform a sequence of steps using a working copy of a procedure in progress that had previously been correctly validated against the controlled copy on May 1st. Which one of the following statements correctly describes the required actions of the NLO?

A. Perform just the designated steps as directed using the existing working copy.

B. Re-validate the working copy of the procedure and perform just the designated steps from the existing working copy.

C. Obtain a new working copy of the procedure and perform just the designated steps from the new working copy.

D. Obtain a new working copy of the procedure and inform the shift supervisor that all procedure steps must be performed or validated from the first step in the procedure.

Distracter Analysis:

A. **Incorrect**: the working copy cannot be used unless it has been validated every 14 days
Plausible: if the candidate thinks that the validation requirement exceeds 14 days

B. **Correct Answer**:

C. **Incorrect**: There is no requirement to obtain a new working copy if the validation has exceeded 14 days – in addition this would now have working 2 copies of the procedure with completed steps initialed on each – would be hard to keep track of the configuration control
Plausible: if the candidate thinks that once the working copy has exceeded its validation requirement, it must be replaced

D. **Incorrect**: There is no requirement to obtain a new working copy if the validation has exceeded 14 days - or to revalidate all steps in the procedure
Plausible: This answer is overly conservative – but some candidates might select the most conservative answer if they do not know the requirement.

Bank Question: 657 ***Answer: D***

1 Pt(s)

During a surveillance test of NI system valves, 2NI-9A (NC COLD LEG INJ FROM NV) did not respond to a safety injection signal. The SOM requests that the OSM issue a special order pre-assigning a dedicated operator by name on each shift to open 2NI-9A should a safety injection signal occur before the cause of this condition is corrected.

Which one of the following statements best describes the requirements for issuance of this special order?

A. The OSM is not authorized to issue special orders, only the SOM can issue a special order.

B. The special order cannot be authorized for any situation when a procedure change is required.

C. The special order cannot be issued until after an operability evaluation has been completed.

D. The special order cannot be issued until after the procedure change has been issued.

Distracter Analysis:

A. **Incorrect**: OSMs are authorized to issue special orders at McGuire.
Plausible: based on the standard industry practice having Ops Mgr issue all special orders.

B. **Incorrect**: Special orders can be issued after a procedure change.
Plausible: based on guidance prohibiting issuance until a procedure change is implemented.

C. **Incorrect:** Special orders can be issued after an operability evaluation.
Plausible: based on operability evaluations coming from NSD-203.

D. **Correct**:

Bank Question: 659 ***Answer: B***

1 Pt(s)

Unit 1 is at 1% power, starting up from a plant trip due to multiple power range nuclear instrument failures. Unit 2 is shutting down (30% power) to Mode 3, to investigate the potential common mode failure mechanism. The Unit 2 power range nuclear instrument channel N41 has been tagged out in preparation for the investigation.

Which of the following best describes the TSAIL entry for power range nuclear instrument inoperability during this maintenance for Unit 2?

A. No TSAIL entry is required because N41 will not be required to be operable in Mode 3.

B. A TSAIL entry is required because N41 is inoperable in Mode 1.

C. No TSAIL entry is required because N41 will be within the action statement time limits.

D. A TSAIL entry is required for tracking only

Distracter Analysis:

A. **Incorrect**: A TSAIL entry is required.
Plausible: based on knowledge that no entry would be required in Mode 3.

B. **Correct**: tagging N41 out of service makes N41 inoperable regardless of the outcome of the investigation into the common mode failure.

C. **Incorrect**: A TSAIL entry is required.
Plausible: based on misunderstanding of TSAIL entry requirements.

D. **Incorrect**: A TSAIL entry is required due to N41 inoperability in Mode 1.
Plausible: based on the requirement for a "tracking only" entry.

Bank Question: 660 ***Answer: D***

1 Pt(s)

Unit 2 was operating at 100% power. There is a packing leak on a VI system containment isolation valve inside lower containment. You are reviewing an RWP that controls the work permit to inspect and repair the VI valve.

What are the minimum dosimetry requirements for this job?

A. Thermoluminescent and neutron dosimeters

B. Electronic alarming and neutron dosimeters

C. Electronic alarming and pocket ion chamber dosimeters

D. Thermoluminescent and electronic alarming dosimeters

Distracter Analysis: All personnel entering an RCA must have a real time reading dosimeter to alert them of dangerous radiation levels and a permanent record dosimeter to record their exposure for their legal record. In addition, neutron exposure is measured by either measuring neutron radiation levels in the general area and multiplying by actual stay time, or by measuring the ratio of neutron dose to gamma dose and then ratioing the TLD dose measurement.

A. **Incorrect**: Must have a PIC or EAD.
Plausible: if the candidate focuses on the requirement to monitor neutron exposure – inside containment of an operating reactor.

B. **Incorrect**: Must have a TLD – no legal record of gamma dose.
Plausible: if the candidates think that EADs or PICs are permanent record dosimeters.

C. **Incorrect**: No neutron dosimetry and no legal record of dose received.
Plausible: if the candidate is not aware of requirement to have a TLD

D. **Correct**: Neutron dose can be measured by ratioing the gamma dose to the neutron dose and using the TLD readout.

Bank Question: 661 ***Answer: A***

1 Pt(s) Units 1 and 2 are at 100% power. Given the following conditions:

- Unit 2 has experienced 2 fuel pin failures.
- The mechanical seal has failed on NI pump 2B.
- The NI-2B pump room general area is 200 mrem/hr.
- In order to reach the NI-2B pump room the workers must transit through 6 rem/hr high radiation area for 1 minute and return.
- Worker A has an accumulated annual dose of 400 mrem, respectively.

How long can worker A participate in the seal repair on NI Pump 2B without exceeding the alert flag exposure limit for external exposure?

A. No longer than 5 hours

B. No longer than 5.5 hours

C. No longer than 6 hours

D. No longer than 7 hours

Distracter Analysis:
The candidate should determine that the alert flag exposure limit is 80% of 2000 mrem admin limit = 1600 mrem

Transient exposure is 200 mrem (6000mrem/hr x 2/60hr). (During transit to and from job).
400 mrem + 200 mrem = 600 mrem
1600 mrem – 600 mrem = 1000 mrem allowable before reaching alert flag exposure admin limit

1000 mrem / 200 mrem/hr = 5 hours

A. **Correct**:
B. **Incorrect**: The answer is 5 hours
Plausible: based on calculating a one-way transit dose.
C. **Incorrect**: The answer is 5 hours.
Plausible: based on no transit dose.
D. **Incorrect**: The answer is 5 hours.
Plausible: based on using admin limit (2000) versus alert flag.

Bank Question: 662 ***Answer: A***

1 Pt(s) Which one of the following statements correctly describes the use of Severe Accident Management Guides (SAMGs) by the control room operators?

A. Control room operators are the implementers for the SAMGs, which are used in place of ERGs.

B. Control room operators are the implementers and SAMGs are used simultaneously with ERGs.

C. Control room operators are evaluators for the SAMGs, which are used in place of ERGs.

D. Control room operators are evaluators and SAMGs are used simultaneously with ERGs.

Distracter Analysis:

A. **Correct Answer**:

B. **Incorrect**: SAMGs replace ERGs when implemented
Plausible: partially correct – the control room operators are the implementers

C. **Incorrect**: control room operators are implementers
Plausible: partially correct – the SAMGs are used in place of the ERGs

D. **Incorrect**: control room operators are implementers - SAMGs replace the ERGs
Plausible: psychometric balance

Bank Question: 663 ***Answer: C***

1 Pt(s)

With the plant at 10% power, an Instrument Technician was allowed to adjust the limit switches on NS32A (containment spray pump A discharge containment isolation valve) without a tag-out. He cycled the valve using the manual hand wheel to set up the limit switches. Upon completion of the work, NS32A was manually closed using the manual hand wheel.

Which one of the following statements is correct concerning this maintenance activity?

A. **"A" NS train was operable during the maintenance but NS32A must be cycled electrically.**

B. **"A" NS train was operable during the maintenance because NS-32A would have opened on demand.**

C. **"A" NS train was inoperable during the maintenance and NS-32A must be cycled electrically.**

D. **"A" NS train was inoperable during the maintenance, but is now operable.**

Distracter Analysis:

A. **Incorrect**: NS train was inoperable
Plausible: based on failure to know the effect of manual closure of the valve.

B. **Incorrect**: NS train was inoperable
Plausible: based on failure to isolate the valve during maintenance.

C. **Correct**:

D. **Incorrect**: Not operable until NS-32A is cycled.
Plausible: based on failure to know the effect of manual closure of the valve.

Bank Question: 664 ***Answer: D***

1 Pt(s)

Unit 1 was conducting control rod drop tests during a plant startup at 4% reactor power when a complete loss of RN occurred. Given the following events and conditions:

- Control room operators enter AP/20 (Loss of RN)
- Several NCP motor stator windings exceed 311 °F
- The operators manually trip the reactor but the trip breakers fail to open
- Reactor power is 5%
- Pressurizer pressure = 1930 psig

Which one of the following statements correctly describes the proper procedural flow path for these conditions?

A. **Remain in AP/20, trip all NCPs and commence a reactor shutdown.**

B. **Implement FR-S.1 (Response to Nuclear Power Generation/ATWS) concurrently with AP/20.**

C. **Terminate AP/20, enter E-0 (Reactor Trip or Safety Injection) and immediately transition to FR-S.1.**

D. **Enter E-0 and immediately transition to FR-S.1 while continuing on in AP/20 as time and conditions permit.**

Distracter Analysis:

A. **Incorrect**: AP/20 directs the operator to enter E-0 in step 2
Plausible: E-0 entry is not required if < P-11 and performing control rod drop tests below 5% power per OMP 4-3

B. **Incorrect**: Do not enter FRPs unless ERGs have been implemented
Plausible: Meet the criteria for RED PATH on subcriticality CSF

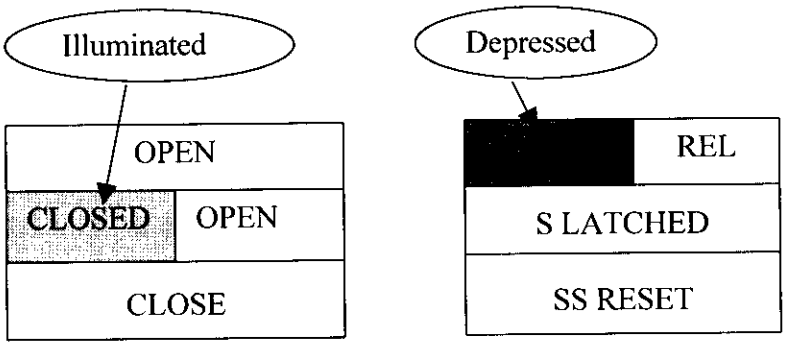
C. **Incorrect**: should not terminate AP/20 as still have a loss of RN.
Plausible: the priority is to make the reactor subcritical

D. **Correct Answer**:

Bank Question: 665 ***Answer: A***

1 Pt(s) Unit 1 was shutdown in Mode 4 preparing to enter Mode 5 during a forced outage to repair a problem on 1NS-18A (A NS Pump Suct From Cont Sump). Given the following events and conditions:

- 1A and 1B ND pumps were in operation
- 1NS-18A was tagged shut
- The following indications existed on the 1NI-185A (RB Sump to Train A ND & NS) controls



Which one of the following actions could cause a loss of shutdown cooling by draining the ND system to the containment sump?

A. Manually depress the "OPEN" pushbutton for 1NI-185A

B. Manually push the "REL" pushbutton for 1NI-185A

C. Manually push the "SS RESET" pushbutton on 1NI-185A

D. The "S LATCHED" light illuminates for 1NI-185A

Distracter Analysis: The normal ND valve interlock between 1ND-19A and 1NI-185A prevents opening 1NI-185A when 1ND-19A is open. This interlock can be bypassed when the BYPASS pushbutton is depressed as long as 1NS-18A is closed.

A. **Correct Answer**: Allows flow path between ND system and containment sump – ND drains to containment

B. **Incorrect**: This will only release the bypass pushbutton

Plausible: if the candidate does not know the function of this control

C. **Incorrect**: This action restores the 1NI-185A interlock with 1ND-19A and resets the S latch
Plausible: if the candidate does not know the ND valve interlock controls

D. **Incorrect**: S latch allows auto opening of 1NI-185A on FWST low level and allows manual opening of 1NI-185A – but the valve is not manually opened in this distracter
Plausible: If a safety injection signal were to occur, 1NI-185A would auto open when FWST level drops to the low level alarm – but 1ND-19A would then auto close

Bank Question: 666 ***Answer: D***

1 Pt(s) Unit 1 was operating at 100% power when a total loss of offsite power occurred. Given the following events and conditions:

- The diesel generators started and loaded as designed
- The operators completed E-0 (Reactor Trip Response)
- The operators reached step 11 of ES-0.1 (Natural Circulation Cooldown) which requires the cooldown of the NC system

Which one of the following components are necessary to prevent the formation of a void in the reactor vessel while cooling down the plant?

A. NI pumps

B. Head vent

C. VL/VU fans

D. CRDM fans

Distracter Analysis:

A. **Incorrect**: NI pumps are not required - SI is blocked
Plausible: If the candidate thinks that SI is required to prevent void in reactor vessel

B. **Incorrect**: not required for cooldown
Plausible: head vent would relieve a void in the reactor vessel but would not prevent a void

C. **Incorrect**: not required for cooldown
Plausible: these are another set of fans in containment that provide cooling to various components

D. **Correct answer**

Bank Question: 672 ***Answer: C***

1 Pt(s) Which one of the following conditions would cause 1EMF-51A (Containment TRN A (Hi Range)) to increase.

A. An increase in alpha radiation from a tritium leak

B. A cloud of radioactive gas that emits beta radiation

C. An increase in gamma flux from a failed fuel event

D. An increase in neutron radiation from a criticality event

Distracter Analysis:

A. **Incorrect**: does not respond to alpha radiation - nor does tritium emit and alpha particle
Plausible: a type of radiological hazard - provided for psychometric balance

B. **Incorrect**: does not respond to beta radiation
Plausible: some detectors respond to beta such as scintillation detectors

C. **Correct answer**

D. **Incorrect**: does not respond to neutron radiation
Plausible: would seem appropriate to measure neutron radiation for criticality events

Bank Question: 676 ***Answer: B***

1 Pt(s) Unit 1 was operating in Mode 4, shutdown cooling. Given the following conditions and events:

- A mixed bed demineralizer was being pre-treated in preparation for placing it in service.
- The operators had initiated automatic makeup to compensate for a reduction in pressurizer level.
- Both trains of ND were in service.

Which one of the following statements correctly describes a condition which would cause an inadvertent dilution of the NCS?

A. **NCS Boron depletion had occurred over the life of the core in the NC system and initiation of automatic makeup caused a dilution event.**

B. **1NVSS5450 (BA FLOW CNTRL) for 1NV-267A had been incorrectly set to 5.0 when it should have been set at 6.7.**

C. **INSS5460 (BA BLEND DISCH CNTRL) for 1NV-252A had been incorrectly set to 4.0 when it was required to be set at 5.5.**

D. **A cation bed demineralizer that had previously been in service at the beginning of the fuel cycle and was not pre-treated was inadvertently placed in service in place of the pre-treated mixed bed demineralizer.**

Distracter Analysis:

A. **Incorrect**: Boron depletion causes the amount of B^{10} in the NC system to be reduced due to neutron flux interactions. This reduces the effective cross-section compared to the same concentration of Boric acid that has not been exposed to neutron flux in the NC system. When blended makeup into the NCS occurs, the boric acid contains a higher proportion of B^{10} for the same concentration of boric acid. This effectively results in adding makeup water that has higher effective concentration of B^{10}.
Plausible: If the candidate reverses the effect of boron depletion i.e. thinks it causes positive reactivity to be added - not negative reactivity. This phenomenon was described in PIP 1-M99-2394 (included in lesson plan PS-NV).

B. **Correct Answer**: This problem would cause a decrease in the addition rate of boric acid to the blender which would decrease the concentration of the boric acid that was added to the NC system. This would cause a dilution of the NC system boric acid concentration..

C. **Incorrect**: A reduction in the set point for the controller for total blended flow would cause 1NV-252A to throttle makeup flow to the blender which would effectively increase the concentration of boric acid added to the NC system - which would not cause the opposite of a dilution event.
Plausible: if the candidate reverses the effect of the setpoint of 1NV-267A.

D. **Incorrect**: adding an untreated cation demineralizer that had last been in service at the start of the cycle would not cause a dilution event. Cation resin does not achieve equilibrium with boric acid (as does anion resin in mixed bed demineralizers which exchanges borate ions for OH ions) and no change would occur to boric acid concentration. In addition, the cation demineralizer was in service at the beginning of the fuel cycle when NCS boric acid concentration was highest - so any residual water in the demineralizer would be at a higher concentration of boric acid - not lower - causing a boration not dilution.
Plausible: if the candidate reverses the effect of adding the demineralizer.

INITIAL SUBMITTAL

MCGUIRE EXAM 2000-301
50-369/2000-301 AND 50-370/2000-301

MAY 8 - 12, MAY 19,
MAY 22 - 25, 2000

INITIAL SUBMITTAL
SRO WRITTEN EXAMINATION

ANSWER KEY VERSION

PWR SRO Examination Outline *ES-401-4*

Facility: McGuire		Date of Exam: 5/19/00										Exam Level: SRO		
		K/A Category Points											Point	
Tier	Group	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G	Total	Target
1 Emergency & Abnormal Plant Evolutions	1	3	5	5				5	2			4	24	24
	2	1	2	3				3	4			3	16	16
	3	1	0	0				0	1			1	3	3
	Tier Totals	5	7	8				8	7			8	43	43
2 Plant Systems	1	2	1	0	2	1	2	3	2	2	2	2	19	19
	2	2	2	2	3	1	1	2	2	0	1	1	17	17
	3	0	0	1	0	1	0	0	2	0	0	0	4	4
	Tier Totals	4	3	3	5	3	3	5	6	2	3	3	40	40
3	Generic Knowledge and Abilities				Cat 1		Cat 2		Cat 3		Cat 4			
					4		5		4		4		17	17

ES-401 — PWR SRO Examination Outline — Form ES-401-3

Emergency and Abnormal Plant Evolutions - Tier 1/Group 1

E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G	K/A Topic(s)	Imp.	Points	Bank Question
000001 Continuous Rod Withdrawal / I				1.05			Ability to operate and/or monitor ...reactor trip switches	4.3/4.2	1	308
000003 Dropped Control Rod /I			3.07				Knowledge of the reasons for the following responses ...Tech spec limits for Tave	3.8*/3.9*	1	599
000005 Inoperable/Stuck Control Rod /I						4.48	Ability to interpret control room indications to verify the status and operation of system and how operator actions and directives affect plant system conditions	3.5/3.8	1	401
000011 Large Break LOCA /III	1.01						Knowledge of the operational implications of the following concepts ...natural circulation and cooling including reflux boiling	4.1/4.4	1	601
W/E04 LOCA Outside Containment / III				1.3			Ability to operate and/or monitor the following...desired operating results during abnormal and emergency situations	3.8/4.0	1	507
W/EO2 SI Termination / III					2.2		Ability to determine and interpret the following...adherence to appropriate procedures and operation within limitations in the facilities license and amendments	3.5/4.0	1	602
000015/17 RCP Malfunction / IV					2.09		Ability to determine and interpret...when to secure RCPs on high stator temperatures	3.4/3.5	1	242
W/E09 Natural Circ. /IV	1.1						Knowledge of the operational implications of the following concepts as they apply to the...components, capacity and function of emergency systems	3.0/3.4	1	666
000024 Emergency Boration /I	1.02						Knowledge of the operational implications of the following concepts ...relationship between boron addition and reactor power	3.6/3.9	1	591
000026 Loss of Component Cooling Water / VIII						4.24	Knowledge of loss of cooling water procedures	3.3/3.7	1	592
000029 Anticipated Transient w/o Scram / I		2.06					Knowledge of the interrelationships between...breakers, relays and disconnects	2.9*/3.1*	1	241
000040 Steam Line Rupture - Excessive Heat Transfer / IV				1.05			Ability to operate and/or monitor ...manual and automatic RPS trip initiation	4.5/4.5	1	593
W/E08 RCS Overcooling - PTS / IV		2.1					Knowledge of the interrelationships between...components and functions of control and safety systems including instrumentation, signals, interlocks, failure modes, and automatic and manual features	3.4/3.7	1	372
000051 Loss of Condenser Vacuum / IV			3.01				Knowledge of the reasons for the following responses ...loss of steam dump capability upon loss of condenser vacuum	2.8*/3.1*	1	594
000055 Station Blackout / VI			3.02				Knowledge of the reasons for the following responses ...actions contained in EOP for loss of onsite and offsite power	4.3/4.6	1	595
000057 Loss of Vital Ac Elec. Inst. Bus. / VI				1.06			Ability to operate and/or monitor ...manual control of components for which automatic control is lost	3.5/3.5	1	82
000059 Accidental Liquid Radwaste Rel. / IX		2.01					Knowledge of the interrelationships between...radioactive liquid monitors	2.7/2.8	1	609
000062 Loss of Nuclear Service Water / IV						4.22	Knowledge of bases for prioritizing safety functions during abnormal /emergency operations	3.0/4.0	1	664
000067 Plant Fire On-site / IX			3.04				Knowledge of the reasons for the following responses ...actions contained in EOP for plant fire onsite	3.3/4.1	1	596
000068 Control Room Evac. / VIII			3.07				Knowledge of the reasons for the following responses ...maintenance of SG levels using AFW control valves	4.0/4.3	1	501
000069 (W/E14) Loss of CTMT Integrity / V		2.03					Knowledge of the interrelationships between...personnel access hatch and emergency access hatch	2.8*/2.9	1	597
000074 (W/E06&E07) Inad. Core Cooling / IV				1.02			Ability to operate and/or monitor ...RCS cooldown rate	3.9/4.2	1	598
000076 High Reactor Coolant Activity / IX						4.11	Knowledge of abnormal condition procedures	3.4/3.8	1	307
W/EO2 SI Termination / III		2.2					Knowledge of the interrelationships between...facility's heat removal systems, including primary coolant, emergency coolant, decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility	3.5/3.9	1	390
K/A Category Totals:	3	5	5	5	2	4	Group Point Total:	24	24	24

ES-401 — PWR SRO Examination Outline — Form ES-401-3

Emergency and Abnormal Plant Evolutions - Tier 1/Group 2

E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G	K/A Topic(s)	Imp.	Points	Bank Question
000007 Reactor Trip - Stabilization - Recovery / I				1.08			Ability to operate and/or monitor ...AFW system	4.4/4.3	1	504
000008 Pressurizer Vapor Space Accident / III	1.01						Knowledge of the operational implications of the following concepts ...thermodynamics and flow characteristics of open or leaking valves	3.2/3.7	1	311
000009 Small Break LOCA / III				1.13			Ability to operate and/or monitor ...ESFAS	4.4/4.4	1	430
W/E03 LOCA Cooldown - Depress. / IV										
W/E11 Loss of Emergency Coolant Recirc / IV			3.4				Knowledge for the reasons for the following responses...RO or SRO function within the control room team as appropriate to the assigned position in such as way that procedures are adhered to and the limitations in the facilities license and amendments are not violated	3.6/3.8	1	51.00
000022 Loss of Reactor Coolant Makeup / II					2.04		Ability to determine and interpret...how long PZR level can be maintained within limits	2.9/3.8	1	603
000025 Loss of RHR System / IV		2.05					Knowledge of the interrelationships between...reactor building sump	2.6/2.6	1	665
000027 Pressurizer Pressure Control System Malfunction / III					2.10		Ability to determine and interpret...PZR heater energized/de-energized condition	3.3/3.6	1	298
000032 Loss of Source Range NI / VII						2.34	Knowledge of the process for determining the inertnal and external effects of core reactivity	2.8/3.2*	1	512
000033 Loss of Intermediate Range NI / VII				1.02			Ability to operate and/or monitor ...level trip bypass	3.0*/3.1*	1	604
000037 Steam Generator Tube Leak / III						1.7	Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior and instrument interpretation	3.7/4.4	1	605
000038 Steam Generator Tube Rupture / III			3.03				Knowledge of the reasons for the following responses ... Automatic actions associated with high radioactivity in S/G sample lines	3.6*/4.0*	1	606
000054 Loss of Main Feedwater / IV			3.04				Knowledge of the reasons for the following responses ... actions contained in EOPs	4.4/4.6	1	191
W/E05 Inadequate Heat Transfer - Loss of Secondary Heat Sink / IV					2.2		Ability to determine and interpret the following...adherence to appropriate procedures and operation within limitations in the facilities license and amendments	3.7/4.3	1	471
000058 Loss of DC Power / VI						2.22	Knowledge of limiting conditions for operation and safety limits	3.4/4/1	1	608
000060 Accidental Gaseous Radwaste Rel. / IX					2.02		Ability to determine and interpret...the possible location of a radioactive gas leak with the assistance of PEO, health physics and chemistry personnel	3.1/4.0	1	611
000061 ARM System Alarms / VII										
W/E16 High Containment Radiation / IX		2.20					Knowledge of the interrelationships between...facility's heat removal systems, including primary coolant, emergency coolant, decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility	2.6/3.0	1	612
000065 Loss of Instrument Air / VIII										
K/A Category Totals:	1	2	3	3	4	3	Group Point Total:	16	16	16

ES-401 — PWR SRO Examination Outline — Form ES-401-3

Emergency and Abnormal Plant Evolutions - Tier 1/Group 3

E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G	K/A Topic(s)	Imp.	Points	Bank Question
000028 Pressurizer Level Malfunction / II					2.04		Ability to determine and interpret...ammeters and running indicators for CVCS charging pumps	2.6/3.1	1	613
000036 Fuel Handling Accident / VIII						2.30	Knowledge of RO duties in the control room during fuel handling accident such as alarms from fuel handling area, communication with fuel storage facility, system operated from the control in support of fueling operations and supporting instrumentation	3.5/3.3	1	447
000056 Loss of Off-site Power / VI	1.03						Knowledge of the operational implications of the following concepts ...definition of subcooling; use of the steam table to determine it	3.1*/3.4*	1	618
W/E 13 Steam Generator Over-pressure / IV										
W/E 15 Containment Flooding / V										
K/A Category Totals:	1	0	0	0	1	1	Group Point Total:	3	3	3

ES-401 — PWR SRO Examination Outline — Plant Systems - Tier 2 Group 1 — Form ES-401-3

System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G	K/A Topic(s)	Imp.	Points	Bank Question
001 Control Rod Drive								2.09				Ability to predict the impacts of the following malfunction or operation...and based on those predictions, use procedures to correct, control or mitigate the consequences of...station blackout	3.8*/4.0	1	622
003 Reactor Coolant Pump							1.10					Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the...RCP standpipe levels	2.5/2.7	1	623
004 Chemical Volume Control							1.09					Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the...RCS pressure and temperature	3.6/3.8	1	625
013 Engineered Safety Features Actuation		2.01										Knowledge of bus power supplies to...ESFAS/safeguards equipment control	3.6*/3.8	1	627
014 Rod Position Indication											4.5	Ability to interpret control room indicators to verify the status and operation of the system, and understand how operator actions and directives affect plant and system conditions	3.5/3.8	1	637
015 Nuclear Instrumentation						6.03						Knowledge of the effect that a loss or malfunction will have on ...component interconnections	2.6/3.0	1	152.00
017 In-core Temperature Monitor					5.02							Knowledge of the following operational implications...saturation and subcooling of water	3.7/4.0	1	404.00
022 Containment Cooling									3.01			Ability to monitor automatic operation of the...initiation of safeguards mode of operation	4.1/4.3	1	628.00
025 Ice Condenser						6.01						Knowledge of the effect that a loss or malfunction will have on ...upper and lower doors	3.4*/3.6*	1	630
026 Containment Spray										4.05		Ability to manually operate and/or monitor in the control room...containment spray reset switches	3.5/3.5	1	640
055 Condensate	1.03											Knowledge of physical connections and/or cause and effect relationships...MFW	2.6*/2.6	1	531
059 Main Feedwater								2.11				Ability to predict the impacts of the following malfunction or operation...and based on those predictions, use procedures to correct, control or mitigate the consequences of...failure of feedwater control system	3.0*/3.3*	1	538
061 Auxiliary/Emergency Feedwater				4.07								Knowledge of design feature(s) and/or interlock(s) which provide for...turbine trip including overspeed	3.1*/3.3*	1	634
063 DC Electrical	1.02											Knowledge of physical connections and/or cause and effect relationships...AC electrical system	2.7/3.2	1	648
068 Liquid Rad Waste									3.02			Ability to monitor automatic operation of the...automatic isolation	3.6/3.6	1	407.00
071 Waste Gas Disposal											4.46	Ability to verify that alarms are consistent with plant conditions	3.5/3.6	1	479
072 Area Radiation Monitoring							1.01					Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the...radiation levels	3.4/3.6	1	672
072 Area Radiation Monitoring										4.02		Ability to manually operate and/or monitor in the control room...major components	2.5*/2.5	1	635
013 Engineered Safety Features Actuation				4.12								Knowledge of design feature(s) and/or interlock(s) which provide for...safety injection block	3.7/3.9	1	412
K/A Category Totals:	2	1	0	2	1	2	3	2	2	2	2	Group Point Total:	19	19	19

ES-401 — PWR SRO Examination Outline — Plant Systems - Tier 2 Group 2 — Form ES-401-3

System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G	K/A Topic(s)	Imp.	Points	Bank Question
002 Reactor Coolant							1.09					Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the...RCS T-ave	3.7/3.8	1	629
006 Emergency Core Cooling				4.12								Knowledge of design feature(s) and/or interlock(s) which provide for...HPI flow throttling	4.1*/4.3*	1	631
010 Pressurizer Pressure Control		2.01										Knowledge of bus power supplies to...PZR heaters	3.0/3.4	1	632
011 Pressurizer Level Control								2.06				Ability to predict the impacts of the following malfunction or operation...and based on those predictions, use procedures to correct, control or mitigate the consequences of...inadvertent pressurizer spray actuation	3.7/3.9	1	633
012 Reactor Protection						6.03						Knowledge of the effect that a loss or malfunction ...will have on ...trip logic circuits	3.1/3.5	1	638
016 Non-nuclear Instrumentation	1.12											Knowledge of physical connections and/or cause and effect relationships...S/G	3.5*/3.5*	1	639
027 Containment Iodine Removal		2.01										Knowledge of bus power supplies to...fans	3.1*/3.4*	1	621
028 Hydrogen Recombiner and Purge Control															
029 Containment Purge															
033 Spent Fuel Pool Cooling							1.01					Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the...spent fuel pool water level	2.7/3.3	1	642
034 Fuel Handling Equipment											4.48	Ability to interpret control room indications to verify the status and operation of system, and understand how operator actions and directives affect plant and system conditions	3.5/3.8	1	610
035 Steam Generator										4.06		Ability to manually operate and/or monitor in the control room...S/G isolation on steam leak or tube rupture/leak	4.5/4.6	1	643
039 Main and Reheat Steam			3.05									Knowledge of the effect that a loss or malfunction ...will have on ...RCS	3.6/3.7	1	644
055 Condenser Air Removal															
062 AC Electrical Distribution			3.01									Knowledge of the effect that a loss or malfunction ...will have on ...major system loads	3.5/3.9	1	243
064 Emergency Diesel Generator				4.05								Knowledge of design feature(s) and/or interlock(s) which provide for...incomplete-start relay	2.6/3.2	1	651
073 Process Radiation Monitoring								2.02				Ability to predict the impacts of the following malfunction or operation...and based on those predictions, use procedures to correct, control or mitigate the consequences of...detector failure	2.7/3.2	1	653
076 Circulating Water															
079 Station Air				4.01								Knowledge of design feature(s) and/or interlock(s) which provide for...cross-connect with IAS	2.9/3.2	1	451
086 Fire Protection					5.04							Knowledge of the following operational implications...hazards to personnel as a result of fire type or methods of suppression	2.9/3.5*	1	60
103 Containment	1.05											Knowledge of physical connections and/or cause and effect relationships...personnel access hatch and emergency access hatch	2.8*/3.0*	1	620
K/A Category Totals:	2	2	2	3	1	1	2	2	0	1	1	Group Point Total:	17	17	17

ES-401 — PWR SRO Examination Outline — Plant Systems - Tier 2 Group 3 — Form ES-401-3

System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G	K/A Topic(s)	Imp.	Points	Bank Question
005 Residual Heat Removal					5.09							Knowledge of the following operational implications...dilution and boration considerations	3.2/3.4	1	676
007 Pressurizer Relief/Quench Tank								2.02				Ability to predict the impacts of the following malfunction or operation...and based on those predictions, use procedures to correct, control or mitigate the consequences of...abnormal pressure in the PRT	2.6/3.2	1	617
008 Component Cooling Water			3.03									Knowledge of the effect that a loss or malfunction ...will have on ...RCP	4.1/4.2	1	616
041 Steam Dump/Turbine Bypass Control															
045 Main Turbine Generator															
076 Service Water								2.01				Ability to predict the impacts of the following malfunction or operation...and based on those predictions, use procedures to correct, control or mitigate the consequences of...loss of SWS	3.5*/3.7*	1	624
078 Instrument Air															
K/A Category Totals:	0	0	1	0	1	0	0	2	0	0	0	Group Point Total:	4	4	4

ES-401	Generic Knowledge and Abilities Outline (Tier 3)	Form ES-401-5

Facility: McGuire		Date of Exam: 5/19/00	Exam Level: SRO		Bank
Category	K/A #	Topic	Imp.	Points	Question
Conduct of Operations	2.1.21	Ability to obtain and verify controlled procedure copy	3.1/3.2	1	655
	2.1.27	Knowledge of system purpose or function	2.8/2.9	1	217
	2.1.32	Ability to explain and apply all system limits and precautions	3.4/3.8	1	97
	2.1.15	Ability to manage short term information such as night and standing orders	2.3/3.0	1	657
	Total		4	4	
Equipment Control	2.2.23	Ability to track limiting conditions for operations	2.6/3.8	1	659
	2.2.28	Knowledge of new and spent fuel movement procedures	2.6/3.5	1	216
	2.2.22	Knowledge of limiting conditions for operations and safety limits	3.4/4.1	1	207
	2.2.2	Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels	4.0/3.5	1	465
	2.2.21	Knowledge of pre and post maintenance operability requirements	2.3/3.5	1	663
	Total		5	5	
Radiation Control	2.3.1	Knowledge of 10 CFR 20 and related facility radiation control requirements	2.6/3.0	1	125
	2.3.4	Knowledge of radiation exposure limits and contamination control, including permissible levels in excess of those authorized	2.5/3.1	1	353
	2.3.10	Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure	2.9/3.3	1	661
	2.3.5	Knowledge of use and function of personnel monitoring equipment	2.3/2.5	1	660
	Total		4	4	
Emergency Procedures and Plan	2.4.8	Knowledge of how the event-based emergency/abnormal operating procedures are used in conjunction with symptom-based EOPs	3.0/3.7	1	338
	2.4.39	Knowledge of the RO's responsibilities in emergency plan implementation	3.3/3.1	1	111
	2.4.47	Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material	3.4/3.7	1	120
	2.4.13	Knowledge of crew roles and responsibilities during EOP ... use	3.3/3.9	1	662
	Total		4	4	
		Tier 3 Point Total	17	17	17