QUESTION # 001

Unit 1 was operating at 100% power when a reactor trip occurred.
All immediate actions for the reactor trip have been performed, and the crew has transitioned to 1ES-0.1, REACTOR TRIP RECOVERY.

Given the following conditions:

- RCS Tave is 550°F and dropping
- Both RCPs are running
- 11 SG NR level is 78% and rising
- 12 SG NR level is 80% and rising
- Aux feed flow is approximately 500 GPM

What are the crew's required actions concerning the main and auxiliary feedwater systems?

a. Immediately reduce aux feedwater flow AND manually close feedwater containment isolation valves (MV-32023 and MV-32024)

b. Verify closed main feed reg valves AND verify total feed flow to SGs is greater than 200 GPM

c. Maintain total feed flow greater than 200 GPM using the Main FW bypass Valves

d. Stop dumping steam AND reduce total feed flow until both SG NR levels are less than 5%

ANSWER
b.
REFERENCE:
1ES-0.1 step 6b, and 6c
NEW
HIGHER
K/A: 000007A102 Ability to operate and monitor the following as they apply to a reactor trip: MFW System
EXPLANATION
This QUESTION examines the applicant's ability to recognize that 1, a feedwater isolation has occurred, closing the Main FW reg valves and their bypasses and the Main Feed pumps have tripped on HI-HI SG level. Thus there is no motive force for main feed flow; and aux feed flow is excessive for the current plant parameters, leading to an excessive cooldown scenario.

a. Incorrect. Closing the feedwater containment isolation valves is only warranted if the Main FRVs will not close, IAW 1ES-0.1 step 6b.2 RNO, additionally the action is inappropriate since there is no main feed flow. This option is plausible if the applicant fails to recognize that main feed pumps have tripped, and assumes the FRVs have not closed.
b. Correct. 1ES-0.1 step 6b, and 6c require the main feed reg valves demand reduced to zero, valves verified closed, and reduction in feed flow is at least 200 GPM total.
c. Incorrect. Based upon the conditions given in the stem, the MFW pumps have tripped on HI-HI level. The only source of feedwater flow to the SG is from aux feedwater, operating the main FW bypass valves will have no effect on total feed flow. Plausible if there is a misconception as to the arrangement of the FW system piping, or failure to recognize the main FW pumps are tripped.
d. Incorrect. 1ES-0.1 step 3 RNO requires these actions only if RCS Tave cannot be maintained greater than 547°F with RCPs running. Plausible if the applicant fails to recognize the correct temperature associated with this continuous action step.

QUESTION # 002

Given the following conditions:

- An event has occurred on Unit 1.
- Subsequently, the Reactor automatically tripped, and SI initiated.
- Annunciator 47012-0109, PRZR SAFETY/RELIEF VALVE FLOW is LIT.
- 1TI-436, PRZR Safety Line A Temperature reads 220°F and rising.
- 1TI-437, PRZR Safety Line B Temperature reads 120°F and rising.
- Pressurizer level is rising.

Based on the above conditions, what is the approximate rate RCS mass is being discharged?

a. 2983 lbm/min

b. 5417 lbm/min

c. 5750 lbm/min

d. 5967 lbm/min

ANSWER
c.
REFERENCE:
B4A Reactor coolant system, revision 15
PINGP USAR section 4.4, revision 22
Tech Spec Basis, amendment 201
ILT LP P8170L-003, Reactor coolant system, revision 6
NEW
HIGHER
000008 A2.25 Ability to determine and interpret the following as they apply to the Pressurizer Vapor Space Accident: Expected leak rate from open PORV or code safety
EXPLANATION:
The QUESTION stem indicates that a vapor space LOCA via one Pressurizer code safety valve has occurred. The Pressurizer code safeties each have temperature elements in their discharge piping. This discharge piping connects to a common header that leads to the PRT. One element indicating significantly higher and at approximately the saturation temperature for the PRT pressure indicates the lifting of only one safety valve. Each of the values listed represents capacity in lbm/min to prevent direct value recognition. The USFAR described capacity for each PZR safety is 345,000 lbm/hr, which is 5750 lbm/min.

a. Incorrect. This corresponds to the relief capacity of one Pressurizer PORV. 179,000 lbm/hr divided by 60 min/hr= 2983.33 lbm/min. Plausible if the applicant mistakes the PORV capacity for a safety valve, or believes that a PZR PORV has stuck open.
b. Incorrect. Plausible because the technical specifications basis, specifies that the minimum required relief capacity is 325,000 lbm/hr, which corresponds to 5416.67 lbm/min.
c. Correct, see EXPLANATION above.
d. Incorrect. Plausible if the applicant correctly diagnosis's one Pressurizer safety valve is lifting, but calculates the capacity of the safety as double that of a PZR PORV.

QUESTION # 003

An event occurred 30 minutes ago with the following conditions,

- The crew is determining if RCS cooldown and depressurization is required per 1E-1, LOSS OF REACTOR OR SECONDARY COOLANT, Step 20.
- Annunciator 47012-0609, O.P.P.S. IMPROPER has JUST LIT
- RCS pressure is 300 PSIG and stable
- Containment pressure is 4 PSIG and slowly lowering

Under these conditions, the crew will…

a. remain in 1E-1, LOSS OF REACTOR OR SECONDARY COOLANT and prepare for recirculation.

b. transition to 1FR-P.1, RESPONSE TO IMMINENT PRESSURIZED THERMAL SHOCK to arrest the excessive cooldown

c. transition to 1ES-1.1, POST LOCA COOLDOWN AND DEPRESSURIZATION and maintain RHR Pumps running for the subsequent cooldown.

d. transition to 1ES-1.1, POST LOCA COOLDOWN AND DEPRESSURIZATION and stop RHR Pumps to avoid pump damage.

ANSWER

REFERENCE:
1F-0.4, revision 9
ARP 47012-0108
ARP 47012-0609
1E-1, LOSS OF REACTOR OR SECONDARY COOLANT, revision 23
NEW
HIGHER
K/A: 000009 2.4.31 Knowledge of annunciator alarms, indications, or response procedures.
EXPLANATION: An auto Main Steam Line Isolation occurs when 2/3 containment pressures reach 16 psig, therefore the containment is adverse. With containment adverse, and RHR flow indicated the RCPs must be tripped while in 1E-1 when pressure is less than 1575 psig. If RCPs are not tripped when the appropriate thresholds are reached and the opportunity is missed, it is not desirable to trip the RCPs when conditions no longer warrant such action.

The PZR low pressure SI alarm clears approximately above 1827 psig, therefore RCPs will not be tripped. Also at this pressure RHR pumps will be operating at their shutoff head. The O.P.P.S. improper will annunciate in this condition once RCS temperature is less than 310°F. Though the RCS has cooled down more than 100°F in an hour, transition to 1FR-P.1 is only required once temperature is less than 250°F.

QUESTION # 004

Given the following conditions:

- 1 hour ago a LOCA occurred on unit 1
- All systems have operated as designed
- Core Exit T/Cs are reading 300°F and slowly lowering
- RCS pressure is slightly higher than containment pressure
- SG pressures are 675 PSIG and slowly lowering

Which one of the following describes the primary heat removal mechanism occurring, and the correct operator response necessary to enhance/maintain core cooling?

a. Break flow ONLY, maintain at least the minimum necessary ECCS injection flow

b. Break flow ONLY, maintain SG NR level using aux feed, and reduce steam pressure using SG PORVs or Steam Dumps to the condenser if available

c. Break flow AND reflux cooling, maintain SG NR level using aux feed, and reduce steam pressure using SG PORVs or Steam Dumps to the condenser if available

d. Reflux cooling ONLY, once the loop seal clears check for proper RHR flow

ANSWER

a.

REFERENCE:

PINGP USAR appendix K, revision 22

ILT LP P8188L-003; Core Damage and Assessment, revision 2

NEW

HIGHER

K/A: 000011K1.01 Large Break LOCA: Knowledge of the operational implications of the following concepts as they apply to the Large Break LOCA : Natural circulation and cooling, including reflux boiling

EXPLANATION:

The maximum containment pressure under any circumstances during a LBLOCA is expected to be less than 50 PSIG. Under these conditions, the RCS has completely blown down, and ECCS is expected to be cooling the core then spilling out of the break. Two-phase cooling will only occur if the SGs are still a heat sink. In this case the SG saturation temperatures are about 200°F greater than the Core exit temperatures, such that the SGs are actually a heat source. The only operator response to aid in core cooling is to continue to follow the ERGs and maintain core cooling using ECCS.

a. Correct, see EXPLANATION above.
b. Incorrect, plausible if the applicant has a misconception about the conditions given, and procedural guidance as to the basis and timing for reducing SG pressure following a LBLOCA.
c. Incorrect, plausible if the applicant has a misconception of the mechanisms for two-phase cooling. Could be correct if reflux cooling were occurring.
d. Incorrect, plausible if the applicant has a misconception of the mechanisms for two-phase cooling, or core cooling during a LBLOCA vs. SBLOCA.

QUESTION # 005

Given the following conditions:

- Unit 1 was operating at 100%
- Annunciator 47012-0102, 12 RCP LOCKED OUT is NOT lit
- Annunciator 47012-0202, 12 RCP OVERLOAD is NOT lit
- 12 RCP current is 100 amps
- ALL 3 Loop B RC flow meters read 20%

Which one of the following choices correctly describes what RCP malfunction has occurred?

a. The 12 RCP flywheel has separated from the motor shaft

b. The 12 RCP impeller has seized within the pump casing

c. The 12 RCP shaft has sheared at the motor to pump coupling

d. The 12 RCP has an electrical fault in the stator windings with no breaker trip

ANSWER

c.

REFERENCE

C3, Reactor Coolant Pump, revision 36

ILT LP, P8170L-002 Reactor Coolant Pumps, revision 4

PINGP USAR Section 14, revision 29

ARP 47012-0102

ARP 47012-0202

NEW

HIGHER

K/A: 000015/17 K2.10 – Knowledge of the interrelations between the Reactor Coolant Pump Malfunctions (Loss of RC Flow) and the following: RCP indicators and controls

EXPLANATION

The RPS will trip the reactor automatically above P-8 (2/4 PRNI >10%) if flow in either loop is less than 90% on 2 of 3 loop flow detectors. PINGP USAR Figure 14.4-50 shows that volumetric flow in the loop of the affected RCP will reverse 2 seconds and a stable negative flow rate should be about 20%. PINGP uses elbow type loop flow transmitters that have a single high pressure tap on the outside bend of the loop, and 3 inner tap low pressure connections on the inner portion of the loop, therefore, regardless of the direction of the loop flow, a positive scale deflection will be read from all 3 meters in the MCR. Motor current should be minimum as a result of the separation of the motor from the pump.

a. Incorrect, if a flywheel to shaft separation were to occur, loop flow would be higher than 20%, and the reactor would not have tripped. Plausible if the applicant misunderstands the mechanisms involved with flywheel separation.
b. Incorrect, this condition would result in a motor over current condition, which would cause the listed annunciators to be lit. Plausible if the applicant confuses the shaft shear vs. seizure indications.
c. Correct, see EXPLANATION above.
d. Incorrect, plausible if the applicant has a misconception regarding the positive flow indication given.

QUESTION # 006

A loss of reactor coolant has occurred on Unit 1. While starting the 11 RHR Pump in recirculation mode in accordance with 1ES-1.2, Transfer to Recirculation, the 11 RHR Pump tripped because of an incorrect valve line-up. Operators have re-executed 1ES-1.2 and are at Step 13.b. This step directs the operators to start the idle RHR Pump (12 RHR Pump).

- All of the lineup steps have been completed correctly.
- Reactor pressure is less than 250 psig.
- The auxiliary building operator has correctly completed Attachment K.

RWST To RHR Isolation:	MV-32084	MV-32085
SI Test Line To RWST Valves:	MV-32202	MV-32203
RHR To Reactor Vessel Injection Valve:	MV-32064	MV-32065
CC To RHR Heat Exchanger:	MV-32093	MV-32094
Sump B to RHR	MV-32075 AND MV-32077	
	MV-32076 AND MV-32078	

Which of the following is the proper alignment of systems and required conditions that will result in proper recirculation operation when 12 RHR Pump is started?

a. MV-32084 Closed MV32202 Closed MV-32064 Open MV32093 Open
MV-32075 AND MV-32077 Full Open Containment Level 2’2”

b. MV-32085 Closed MV-32203 Closed MV-32065 Open MV-32094 Open
MV-32076 AND MV-32078 Full Open 80% Containment Sump B Level

c. MV-32084 Closed MV-32202 Open MV-32064 Open MV-32093 Closed
MV-32075 AND MV32077 Full Open 70% Containment Sump ‘’B’ Level

d. MV32085 Closed MV-32203 Open MB32065 Open MV32094 Open
MV-32076 AND MV-32078 Full Open Containment Level 2’2”

ANSWER
b.
REFERENCE
1ES1.2 Transfer to Recirculation Rev 21, Step 13
1E-1, Loss of Primary or Secondary Coolant, Rev 23, Step 16
NEW
HIGHER
K/A 000025K2.05 Loss of RHR System: Knowledge of the interrelations between the Loss of Residual Heat Removal System and the following: Reactor building sump
EXPLANATION:

a. the lineup for the 11 RHR pump.
b. the correct ANSWER
c. the lineup for the 11 RHR pump with a sump level < allowable, 32202 incorrect
d. the lineup for the 12 pump with MV-32203 incorrect

Question #: 007

Given the following conditions:

- Unit 2 was operating at full rated power.
- Steam Generator Blowdown (SGB) was aligned to the discharge canal.
- 2-R-19, Steam Generator Blowdown radiation monitor, HIGH alarm actuated.

Which of the following automatic actions occurred, and why are the automatic actions required?

	Actions	**Reason**
a.	21 and 22 SGB motor operated containment isolation valves go CLOSE.	Minimize the effects of inadvertent release of radioactivity to the environment.
b.	21 and 22 SGB flow control valves to the 21 SGB flash tank go CLOSE.	Minimize exhaustion of the cation/anion resin beds in the SGB ion exchangers.
c.	21 and 22 SGB flow control valves to the 21 SGB flash tank go CLOSE.	Minimize the effects of inadvertent release of radioactivity to the environment.
d.	21 and 22 SGB motor operated containment isolation valves go CLOSE.	Minimize exhaustion of the cation/anion resin beds in the SGB ion exchangers.

ANSWER:
c.
Radiation Monitoring System Part-2 Lesson Plan, P8182L-002
B11, Radiation Monitoring System Description, Revision 10
B21B, Liquid Waste System, Revision 12
C47022, Alarm Response Procedure, Revision 37
C47047, Radiation Monitor Alarm Response Procedure, Revision 35
NEW
FUNDAMENTAL
K/A: 000038 K3.03 Steam Gen. Tube Rupture / 3: – Knowledge of the reasons for the following responses as they apply to the SGTR: Automatic actions associated with high radioactivity in S/G sample lines. (RO 3.6)
EXPLANATION:
a. Is incorrect. The 21 and 22 SGB motor operated isolation valves are Containment Isolation valves and only go closed automatically on CI signals.
b. Is incorrect. Although the 21 and 22 flash tank inlet valves will go closed automatically when an R-19 HIGH alarm is received, the resin exhaustion is not the concern.
c. Is correct. This is one of the automatic actions associated with an R-19 HIGH alarm, and is required to prevent/minimize effects of an inadvertent release to the environment.
d. Is incorrect. Both parts are not correct, as stated above.

Question #: 008

Given the following conditions:

- Unit 1 was operating at 50% rated power, EOL.
- Rod Control system was in manual.

Assuming no operator action, describe how reactor power will respond during the initial phase of a major steam line break upstream of the MSIV on 12 SG.

a. Reactor power will rise rapidly until the MSIVs close and then lower and stabilize at the POAH.

b. Reactor power will initially rise and then return to the original value of 50% rated power when the 12 SG is dry.

c. Reactor power will initially rise and then lower and stabilize at approximately 25% rated power.

d. Reactor power will rise rapidly until terminated by a reactor trip generated by the SI initiation signal.

ANSWER:
d.
REFERENCE
Steam Generating Lesson Plan, P8170L-003B
Secondary Accidents Lesson Plan, P8161L-006
NEW
FUNDAMENTAL
K/A: 000040 A1.14 Steam Line Rupture - Excessive Heat Transfer /4: – Ability to operate and/or monitor the following as they apply to the Steam Line Rupture: Nuclear Instrumentation. (RO 4.2)
EXPLANATION:
a. Is incorrect.
b. Is incorrect.
c. Is incorrect.
d. Is correct.

Question #: 009

Given the following conditions:

- Unit 1 was operating at rated power.
- Unit 2 was operating at ~ 6% reactor power, MODE 1, placing the generator on the grid.
- A spurious Safety Injection signal occurred for Train A.
- 21 RCP tripped during the resulting grid perturbation.

Assuming only EOP immediate actions are performed, which Aux Feedwater pumps, if any, on both Units will automatically start and what is the final position of all 3" MOV pump discharge valves?

a. The TD and MD AFW pumps on both Units start and all 3" MOV discharge valves remain in the fully OPEN position.

b. The TD AFW pumps on both Units and the 12 MD AFW pump start, and with the exception of the 21 MD AFW pump discharge valves which go fully SHUT, all 3" MOV discharge valves remain in the fully OPEN position.

c. The MD AFW pumps on both Units and the 11 TD AFW pump start, and with the exception of the 22 TD AFW pump discharge valves which go fully SHUT, all 3" MOV discharge valves remain in the fully OPEN position.

d. The TD and MD AFW pumps on Unit 1 start, the TD and MD AFW on Unit 2 remain in standby (<10% reactor power and therefore no reactor trip), and all 3" MOV discharge valves remain in the fully OPEN position.

ANSWER:
a.
REFERENCE
Auxiliary Feedwater System Lesson Plan, P8180L-007
Secondary Accidents Lesson Plan, P8161L-006
B8, Reactor Protection System, Revision 6
B18C, Engineered Safeguards System, Revision 8
2C1.2, Unit 2 Startup Procedure, Revision 53
NEW
HIGHER
K/A: 000054 A2.04 Loss of Main Feedwater / 4: – Ability to determine and interpret the following as they apply to the Loss of Main Feedwater (MFW): Proper operation of AFW pumps and regulating valves. (RO 4.2)
EXPLANATION:
a. Is correct.
b. Is incorrect.
c. Is incorrect.
d. Is incorrect.

Question #: 010

Given the following conditions:

- Unit 1 was shutdown in Mode 4.
- D1 was out of service for maintenance.
- Unit 2 was at rated power.
- A LOOP occured to both units.
- D5 did not automatically start.
- All other equipment operated as required.

Which EOP entry conditions, for both Units, were met?

	Unit 1	Unit 2
a.	1ECA-0.0	2E-0
b.	1ECA-0.1	2E-0
c.	1E-4	2E-0 and 2ECA-0.0
d.	1ECA-0.0	2E-0 and 2ECA-0.0

ANSWER:
a.
REFERENCE
E-0 Series Procedures Lesson Plan, P8197L-011
NEW
HIGHER
K/A: 000055 2.1.20 Station Blackout / 6: – Ability to interpret and execute procedure steps. (RO 4.6)
EXPLANATION:
a. Is correct.
b. Is incorrect.
c. Is incorrect.
d. Is incorrect.

Question #: 011

Given the following conditions:

- Unit 1 was in Mode 2, startup in progress with Reactor Power at ~5%.
- Unit 2 was operating at rated power.
- A LOOP occurred to Unit 1.

Which of the following is positive indication that natural circulation has been established?

a. RCS cold leg temperature at or near saturation temperature for SG pressure.

b. All heaters energize and the spray valves go.

c. All heaters energize and the spray valves go.

d. All heaters energize and the spray valves go.

ANSWER:
a.
REFERENCE:
Loss of Flow Accidents Lesson Plan, P8161L-009
Non-Safeguards Distribution Lesson Plan P8186L-003
B20.5, 4.16 KV Station Auxiliary System
NEW
FUNDAMENTAL
K/A: 000056 K1.01 Loss of Off-site Power / 6: – Knowledge of the operational implications of the following concepts as they apply to Loss of Offsite Power: Principle of cooling by natural convection. (RO 3.7)
EXPLANATION:
a. Is correct.
b. Is incorrect.
c. Is incorrect.
d. Is incorrect.

QUESTION # 012

With Unit 1 operating normally at 100% power, the following annunciators energize:

- **47018-0501**, SAFEGUARD LOGIC TRAIN A DC FAILURE
- **47024-1003**, D1 EMERGENCY GENERATOR LOSS OF CONTROL VOLTAGE
- **47024-1102**, 11 DC SYSTEM TROUBLE
- **47024-1201**, 11 DC PANEL UNDERVOLTAGE

Which of the following will occur in conjunction with these annunciators?

a. Reactor remains at power but control power for D1 is lost.

b. Reactor remains at power and D1 auto starts but the field will not flash.

c. Reactor trip will occur, D1 will not auto start and auto fuel oil transfer is lost.

d. Reactor trip will occur, D1 auto starts and the field may flash due to residual magnetic flux. Fuel oil transfer still works in automatic.

ANSWER
c.
REFERENCE
C20.9AOP1, LOSS OF UNIT 1 TRAIN “A" DC, Rev 6
K/A 000058K3.01 – Knowledge of the reasons for the following responses as they apply to the Loss of DC Power: Use of dc control power by D/Gs
NEW
HIGHER
EXPLANATION:

a. Reactor trips. Plausible if applicant does not realize a reactor trip occurs.
b. Reactor trips. Plausible if applicant does not realize a reactor trip occurs.
c. correct ANSWER
d. Fuel oil transfer is lost. Plausible if applicant does not realize fuel oil transfer is dc powered.

QUESTION # 013

Given the following conditions:

- Both units are operating at 100% power
- The Cooling Water (CL) system was in its normal alignment
- A large break then occurs in the Train “B” CL header
- Loop A CL pressure is 63 PSIG
- Loop B CL pressure is 25 PSIG
- Loop A CL flow is 16000 GPM
- Loop B CL flow is 34000 GPM
- All equipment has responded as designed

Which of the following is correct combination of Cooling Water System pumps one minute after the break occurred?

a. 11/21/121 Motor Driven CL AND 12/22 Diesel Driven CL pumps are running

b. 11/21 Motor Driven CL AND 12/22 Diesel Driven CL pumps are running, 121 CL Pump has tripped

c. 11/21/121 Motor Driven CL AND 22 Diesel Driven CL pumps are running, the 12 CL pump has tripped

d. 11/21 Motor Driven CL AND 22 Diesel Driven CL pumps are running, the 12 AND 121 CL pumps have tripped

ANSWER
b.
REFERENCE
ILT LP P8176L-003, Cooling Water System, revision 5
PINGP USAR Section 10.4.1, revision 29
B35, Cooling Water System, revision 12
K/A: 000062 A1.07 – Ability to operate and / or monitor the following as they apply to the Loss of Nuclear Service Water (SWS): Flow rates to the components and systems that are serviced by the SWS; interactions among the components
EXPLANATION
Cooling Water pressure less than 80 PSIG will auto start the 121 CL Pump. If pressure drops less than 75 PSIG for greater than 15 seconds, both Diesel driven CL pumps will start, and once their speed is greater than 400 RPM, 121 CL pump should automatically trip.

a. Incorrect. Plausible if the applicant has a misconception of the automatic features of the system.
b. Correct. See EXPLANATION above.
c. Incorrect. Plausible if the applicant believes that is the reason for the difference in header flow.
d. Incorrect. Combination of the correct ANSWER and ANSWER C

QUESTION # 014

Given the following initial conditions and a copy of Figure B34-1:

- 121 & 122 air compressors running in PREFERRED
- 123 air compressor in FIRST STANDBY
- 124 air compressor running in PREFERRED
- 125 air compressor in STANDBY
- Normal valve lineup

A break in a Unit 1 instrument air line occurs which causes the pressure to rapidly decrease to <75 psig. **Which of the following correctly describes the automatic actions which would occur due to this failure?**

a. 123 Compressor starts, U1 instrument air header isolation (MV-32314) closes, 121 air dryer bypass (MV-32362) opens

b. 125 Compressor starts, service air header isolation (MV-32318) opens, 121 air dryer bypass (MV-32362) opens

c. 123 Compressor starts, station air receiver to instrument air supply header (CV-39302 and CV-39301) open, 121 air dryer bypass (MV-32362) opens

d. 125 Compressor starts, instrument air to U1 containment (CV-31740 and CV-31741) close, 121 air dryer bypass (MV-32362) opens

ANSWER
a.
REFERENCE
Instrument and Station Air, LP #: p8178I-005, Rev. #:6
Figure B34-1
BANK
HIGHER
K/A 000065A2.01 – Ability to determine and interpret the following as they apply to the Loss of Instrument Air: Cause and effect of low-pressure instrument air alarm
EXPLANATION:

a. Correct ANSWER.
b. 125 Compressor does not start
c. Station air receiver to instrument air supply header valves remain shut.
d. 125 Compressor does not start.

QUESTION # 015

The plant was operating at 100% power when an inter-system LOCA occurred, pressurizing and rupturing the RHR system in the 11 RHR/CCW Heat Exchanger. Operators have succeeded in isolating the RHR system.

Which of the following could be used to determine if reactor core clad damage has occurred?

a. Containment Radiation levels

b. Excess Cobalt-60 in letdown

c. Containment Hydrogen concentration

d. concentration of Noble Gases in the aux building

ANSWER
a.
REFERENCE
F3 17-1, Core Damage Determination
NEW
HIGHER
K/A W/E042.4.21 – LOCA Outside Containment: Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.
EXPLANATION:

a. The release was not in containment. No change in containment radiation levels.
b. Clad damage would not release additional cobalt, pellet melt would release cobalt.
c. The release was not in containment. Hydrogen generation would not escape to containment.
d. Correct ANSWER.

QUESTION # 016

Unit 1 was operating at 100% power when a large break loss of coolant (LBLOCA) occurred. All ECCS equipment initiated correctly. Shortly after the 11 and 12 RHR Pumps started, a weld seam at the bottom of the RWST split open, causing a rapid loss of RWST water level. Approximately ¼ of the RWST water was successfully injected into the reactor vessel before the tank emptied.

How will the loss of RWST water level affect the operation of the ECCS system?

a. Inadequate boron will be injected into the core to prevent criticality when the core is re-flooded. Core damage is expected.

b. There will be inadequate water in the containment sump B to support recirculation. Core damage may occur.

c. The 11 and 12 RHR Pumps will automatically re-align to Recirculation Sump B at RWST low-low level. No core damage is expected.

d. The 11 and 12 RHR Pumps will automatically trip on low suction pressure. Both can be aligned and started in recirc mode. No core damage is expected.

ANSWER
a.
REFERENCE
B15 Residual Heat Removal System
Tech Spec Bases: B 3.5.4 Refueling Water Storage Tank (RWST)
NEW
FUNDAMENTAL
K/A W/E11K1.1 – Knowledge of the operational implications of the following concepts as they apply to the (Loss of Emergency Coolant Recirculation): Components, capacity, and function of emergency systems
EXPLANATION:
Distractor a: Sufficient Boron to shut down the reactor is supplied by the accumulators and the portion of the RWST injected into the core. Re-criticality should not occur.
Distractor b: Correct ANSWER.
Distractor c: There is no automatic alignment to recirc mode.
Distractor d: There is no automatic trip for low suction pressure on the RHR pumps.

Question #: 017

Given the following conditions:

- Unit 1 and 2 were operating at rated power.
- A dual unit trip occurred due to a LOOP.

Which ESF actuation signal is specifically designed to prevent a loss of secondary heat sink, assuming all safety systems function properly with no operator action

a. The Auxiliary Feedwater Pump Start signal.

b. All heaters energize and the spray valves go.

c. All heaters energize and the spray valves go.

d. All heaters energize and the spray valves go.

ANSWER:
a.
REFERENCE
Engineered Safeguards Systems Lesson Plan, P8180L-006
B28B, Auxiliary Feedwater System, Revision 9
NEW
FUNDAMENTAL
K/A: W/E05 K2.1 Inadequate Heat Transfer - Loss of Secondary Heat Sink / 4: – Knowledge of the interrelations between the (Loss of Secondary Heat Sink) and the following: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features. (RO 3.7)
EXPLANATION:
a. Is correct.
b. Is incorrect.
c. Is incorrect.
d. Is incorrect.

QUESTION # 018

Given:

- Both Units were operating at 100% power.
- An oil leak developed on 2RX transformer and it was taken out of service.
- All actions in C20.3 AOP 4 for removing the transformer were completed.
- Unit #1 then tripped.

What actions are required to be taken in response to this event to meet bus loading limits?

a. No actions are required to be taken.

b. Unit #2 is required to reduce power and remove a MFW pump from service.

c. Unit #2 can remain at 100% power and Unit #1 must be placed on natural circulation.

d. Unit #2 can remain at 100% power and Unit #1 can run 1 RCP and 1 MFW pump.

ANSWER
c.
REFERENCE
2C.20.5 U2 4.16 KV System, Section 5.9, Transfer of 4.16KV Buses 21 and 22 to 1R Transformer from 2RX Transformer, Caution
BANK
FUNDAMENTAL
K/A 000077K3.02 Generator Voltage and Electric Grid Disturbances - Knowledge of the reasons for the following responses as they apply to Generator Voltage and Electric Grid Disturbances: Actions contained in abnormal operating procedure for voltage and grid disturbances
EXPLANATION:

a. Action to reduce loading must be taken.
b. Unit 2 will be unaffected with the U1 Scram.
c. Correct ANSWER.
d. U1 must trip all RCPs and MFW pumps to meet bus loading requirements.

QUESTION # 019

Given the following conditions:

- Unit 1 is in Mode 3 following a Reactor trip 5 minutes ago.
- THREE RCCAs failed to insert during the reactor trip.
- The MAKE-UP MODE SELECTOR control switch is in MANUAL.
- Blender outlet valve, CV-31200 is OPEN.
- BA Inlet to Blender control valve, CV-31199 is OPEN.
- HC-110, BA TO BLENDER FLOW CONT, is in MANUAL and currently set to 32%.
- The BORIC ACID MAKEUP CONTROL switch has just been placed in START.

Assuming Unit 1 will remain in Mode 3, the MAXIMUM allowable boric acid flow controller setting is ______ , and actual boric acid flow will _____ over the next hour.

a. 75%, Rise

b. 80%, Lower

c. 90%, Rise

d. 100%, Lower

ANSWER
d.
REFERENCE
C12.5 AOP 1, Emergency Boration of the Reactor Coolant System, revision 3
ILT LP P8172L-001A, CVCS, revision 5
B12A, Chemical and Volume Control, revision 11
1FR-S.1, Response to Nuclear Power Generation/ATWS, revision 12
NEW
HIGHER
K/A: 000024 A1.03 – Ability to operate and / or monitor the following as they apply to Emergency Boration: Boric acid controller
EXPLANATION
Plant procedures require Boration flow to be limited to 75% of charging flow to prevent adverse effects of boric acid fouling RCP seal injection components. This is required knowledge of the operators. In this case, 100% boric acid flow or 15 GPM should be accommodated since charging flow is not expected to drop less than 20 GPM. Overall VCT level will rise as a result of the addition of the boric acid, causing VCT pressure to also rise. Over the course of the Boration (which will take over an hour under these conditions), VCT level and pressure will remain higher than their pre-trip values, and with the boric acid flow controller setting unchanged, actual boric acid flow will lower as a result of the increased VCT pressure.

a. Incorrect. Plausible if the applicant fails to properly analyze the CVCS system response and/or confuses maximum boric acid flow controller setting with the 75% flow limitation.
b. Incorrect. Plausible if the applicant miscalculates the boric acid controller setting.
c. Incorrect. Plausible if the applicant fails to properly analyze the CVCS system response and/or does not recall the limitation on boric acid flow.
d. Correct. See EXPLANATION above.

QUESTION # 020

Given the following conditions:

- Unit 1 has been operating at 100% power for several months
- 1FI121, CHG FLOW TO REGEN HX indicates 0 GPM
- 1LI427, PRZR LEVEL indicates 20% and rising
- 1FI132, LETDOWN HX OUTLET FLOW indicates 0 GPM

Which one of the following malfunctions would have resulted in the above conditions?

a. The charging pump controller has failed

b. The controlling Pressurizer Level channel has failed

c. The control power fuse for the Letdown Isolation valve, CV-31226, has blown

d. Instrument air was lost to the Regenerative Heat Exchanger Charging Line Outlet valve CV-31328, causing Letdown to isolate

ANSWER

b.

REFERENCE

ILT LP P8170L-006, Pressurizer Level Control System, revision 4

ILT LP P8172L-001A, CVCS, revision 5

NEW

HIGHER

K/A: 000028A2.06 – Ability to determine and interpret the following as they apply to the Pressurizer Level Control Malfunctions: Letdown flow indicator

EXPLANATION

Normally Pressurizer level is 33% at full power. When the controlling Pressurizer level channel fails high, charging pump speed will go to minimum (16 GPM), the actual Pressurizer level will lower until letdown automatically isolates at 14.8%. Pressurizer level will then rise due to charging through the RCP seals.

a. Incorrect. Plausible if the applicant determines that is the cause of the low Pressurizer level and charging flow.
b. Correct. See EXPLANATION above.
c. Incorrect. Plausible if the applicant believes this is the cause of the letdown being isolated, however, Pressurizer level would be greater than 33%
d. Incorrect. Plausible if the applicant has a misconception of the interlocks associated with the letdown and charging valves.

Question #: 021 (BANK QUESTION: P8184L-002 #62)

Given the following conditions during a reactor startup on Unit 2:

- Unit 2 reactor startup was in progress.
- N35 read 2 x 10^{-10} amps; N36 read 3 x 10^{-10} amps.
- Permissive P-6 was actuated, but SR trips had NOT been blocked.
- The RO had just completed verifying proper SR/IR overlap.
- SR channel N31 has just failed low

Which of the following statements describes Unit 2's compliance with Technical Specifications and the required actions?

a. Violation of a Technical Specification LCO. Trip the reactor and implement E-0.

b. Compliance with Technical Specifications. Suspend any positive reactivity changes until N31 is repaired.

c. Violation of a Technical Specification LCO. Fully insert control rods to maintain the reactor subcritical.

d. Compliance with Technical Specifications. Block the SR trips and continue the reactor startup.

ANSWER:
d.
REFERENCE
Nuclear Instrumentation System (NIS) Lesson Plan, P8184L-002
B9A, Nuclear Instrumentation System, Revision 9
Unit 2 Technical Specification 3.3.1
BANK
HIGHER
K/A: 000032 2.1.7 Loss of Source Range NI / 7: – Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation. (RO 4.4)
EXPLANATION:
a. Is incorrect. Unit is in compliance with TS (both SRs not required with power in the intermediate range) and a Reactor trip and entering 2E-0 in not required.
b. Is incorrect. Unit is not in violation of TS and suspending positive reactivity additions to repair the SR channel is not required.
c. Is incorrect. Unit is currently in compliance with TS (both SRs not required with power in the intermediate range) and this action would return power to source range where both SRs are required.
d. Is correct. Unit is in compliance with TS: SRs not required above P-6 setpoint per mode note in Table 3.3.1-1.

Question #: 022 (BANK QUESTION: P8184L-002 #8-MODIFIED)

Given the following conditions:

- Unit 1 is operating at rated power.
- Annunciator C47013-0502, N35 INTERMEDIATE RANGE LOSS OF COMP VOLT alarms.

Which of the following describes the response of the SR Detectors if a Unit 1 reactor trip occurs with no operator actions?

a. BOTH N31 and N32 Source Range Detectors will automatically energize earlier than expected after a reactor trip.

b. ONLY N31 Source Range Detector will automatically energize at a HIGHER power level on decreasing power.

c. BOTH N31 and N32 Source Range Detectors will FAIL to automatically energize on a reactor trip.

d. ONLY N32 Source Range Detector will FAIL to automatically energize on decreasing power.

ANSWER:
c.
REFERENCE
Nuclear Instrumentation System (NIS) Lesson Plan, P8184L-002
B9A, Nuclear Instrumentation System, Revision 9
C47013, Alarm Response Procedure, Revision 35
BANK
FUNDAMENTAL
K/A: 000033 K1.01 Loss of Intermediate Range NI / 7: – Knowledge of the operational implications of the following concepts as they apply to Loss of Intermediate Range Nuclear Instrumentation: Effects of voltage changes on performance.
EXPLANATION:

a. Is incorrect. N35 will always indicate > $10E^{-10}$ amps and therefore neither SR will energize.
b. Is incorrect. N35 will always indicate > $10E^{-10}$ amps and therefore neither SR will energize
c. Is correct. SR Detector HV will not energize to either detector unless 2/2 IRNI indicated reading are < 10^{-10} amps.
d. Is incorrect. N35 will always indicate > 10^{-10} amps and therefore neither SR will energize

Question #: 023 (BANK QUESTION: P8178L-001B #6)

A fuel handling accident occurred in the Spent Fuel Pool. R-25 and R-31 both went into alarm.

Which of the following describes the proper response of the Spent Fuel Pool ventilation fans to this event?

Both 121 & 122 Spent Fuel Special & In-Service Purge Exhaust fans will be running, and . . .

a. the Spent Fuel Normal Make-up and Exhaust fans will be stopped with NO green light indication.

b. the Spent Fuel Normal Make-up and Exhaust fans will be stopped with green light indication.

c. the Spent Fuel Normal Make-up fan will be running and the Exhaust fan will be stopped with green light indication.

d. the Spent Fuel Normal Make-up fan will be running and the Exhaust fan will be stopped with NO green light indication.

ANSWER:
b.
REFERENCE
Radiation Monitoring System Lesson Plan, P8182L-002
B11, Radiation Monitoring System Description, Revision 10
BANK
HIGHER
K/A: 000036 K2.02 Fuel Handling Accident / 8: – Knowledge of the interrelations between the Fuel Handling Incidents and the following: Radiation monitoring equipment (portable and installed).
EXPLANATION:
a. Is incorrect. The green light indications will be on because R-25 is in alarm.
b. Is correct. This is the normal system response to both radiation monitors in alarm.
c. Is incorrect. The make-up fan will stop with green light indication.
d. Is incorrect. The make-up fan will stop and both green light indications will be on.

QUESTION # 024

During a Steam Generator Tube Rupture, feedwater flow is terminated to the affected steam generator after level has returned into the narrow range (> 5% NR). Which one of the following is NOT a reason for terminating feedwater flow?

Feedwater flow is terminated at 5% N/R level to . . .

a. prevent an unnecessary transitions to FR-H.1.

b. minimize the possibility of steam generator overfill.

c. promote thermal stratification and prevent SG tubes from coming in contact with the ruptured SG steam space.

d. reduce the radioactive release rate by "filtering the release" through a layer of water prior to it escaping to the atmosphere.

ANSWER
d.
REFERENCE
P8197, E-3 Review, Step D.1.f, p32
NEW
FUNDAMENTAL
K/A 000037K3.07 Steam Generator Tube Leak – Knowledge of the reasons for the following responses as they apply to the Steam Generator Tube Leak: Actions contained in EOP for S/G tube leak.
EXPLANATION:

a. S/G overfill is minimized
b. It could prevent an unnecessary transition to FR-H.1.
c. It promotes thermal stratification and prevents S/G tubes from coming in contact with the ruptured S/G steam space.
d. Correct answer. Scrubbing in this content is a mis-conception.

QUESTION # 025

A release of liquid waste to the standpipe is in progress. **If the process monitor R-18, liquid waste disposal, control room alarm actuates, what automatic action(s) occur?**

a. Valve CV-31256, waste liquid release line isolation, closes.

b. Valve CV-31841, waste liquid release line keylock isolation, closes.

c. Valve CV-31465, combined U1 and U2 SGB keylock isolation closes.

d. Valves CV-31519 and CV-31607, Unit 1 and Unit 2 SGB isolations close.

ANSWER
a.
REFERENCE
Lesson Plan R3901C, Rev 05, page 18 of 25, Learning Objective 3.2
NEW
FUNDAMENTAL
B21B LIQUID WASTE SYSTEM, Section 3.12, Liquid Effluent Release Line
K/A 000059A2.04 Accidental Liquid RadWaste Release – Ability to determine and interpret the following as they apply to the Accidental Liquid Radwaste Release: The valve lineup for a release of radioactive liquid
EXPLANATION:

a. correct answer
b. does not shut on high radiation
c. does not shut on high radiation, wrong flow path
d. does not shut on high radiation, wrong flow path

QUESTION # 026

Given the following conditions:

- A release of 121 and 125 Waste Gas Decay Tanks is in progress
- The radioactivity content of these tanks is 1000 times higher than expected due to errors in the sample analysis
- 2R-30 has reached the alarm setpoint; the operators are verifying automatic actions per the ARP

Which of the following actions will occur automatically to stop the gaseous radwaste release?

a. 121 and 122 Sample Room exhaust fans stop

b. Laundry, Locker, and Filter Room ventilation exhaust fans stop

c. Low Activity Gas Decay Tanks Plant Vent Valve (CV-31271) closes

d. 122 Aux Building Special Ventilation starts

ANSWER:
c.
REFERENCE:
Facility Exam Bank
FUNDAMENTAL
K/A: 000060 Accidental Gaseous Radwaste Rel. 2.1.45 – Ability to identify and interpret diverse indications to validate the response of another indication.
Explanation:

a. Incorrect, Sample Room exhaust fans don't stop automatically.
b. Incorrect, Laundry, Locker, and Filter Room ventilation exhaust fans stop automatically, but don't affect the release.
c. Correct, per reference.
d. Incorrect, starts but does not terminate release.

QUESTION # 027

Both units were operating normally at 100% power when R-1, Control Room Area Radiation Monitor began alarming. **Which of the following automatic actions will occur?**

a. There are no automatic actions.

b. The Outside Air Supply Damper CLOSES for U1

c. The Outside Air Supply Dampers will CLOSE for both U1 and U2

d. The PAC Filter Outside Air Supply Damper will CLOSE

ANSWER
a.
REFERENCE
P8182L-002, Radiation Monitoring System, Section VI.A.2
NEW
FUNDAMENTAL
K/A 000061A1.01 ARM System Alarms – Ability to operate and / or monitor the following as they apply to the Area Radiation Monitoring (ARM)System Alarms: Automatic actuation
EXPLANATION:

a. ARMs have no automatic actions. b., c., d., are incorrect.

QUESTION # 028

Given the following conditions:

- Unit 1 is CRITICAL and in Mode 2 following a refueling outage
- Unit 2 is operating at 100% Power, with normal full power system alignment

The following sequence of events occurs:

- An operator attempts to start the 13 Heater Drain Pump
- The 13 Heater Drain Pump immediately trips
- Annunciator 47006-0101, 1R RESERVE AUX XFMR SUDDEN PRESS TRIP lights
- BOTH Unit 1 EDGs start and supply power to safeguards buses 15 and 16

With the above conditions, the Unit 1 Reactor…

a. must be manually tripped, and both RCPs are running

b. has automatically tripped because no RCPs are running

c. must be manually tripped because no RCPs are running

d. May remain in Mode 2 because one RCP is running

ANSWER
c.
REFERENCE
ILT LP P8170L-002; Reactor Coolant Pumps, revision 4
ILT LP P8186L-003; 4160 & 480V Non-Safeguards Distribution, revision 3
ILT LP P8184L-004; Reactor Protection, revision 8
ARP 47006-0101, revision 24
1C1.2; Unit 1 Startup Procedure, revision 52
NEW
HIGHER
K/A: 003K2.01 — Knowledge of bus power supplies to the following: RCPs.
EXPLANATION
The QUESTION stem indicates that reactor power level is below the point that the automatic reactor trips associated with a loss of flow, are currently blocked. 11 and 12 RCP power supplies are bus 11 and 12, respectively. A sudden pressure trip of the 1R transformer trips and locks out feed breakers for bus 11 and 12 from the 1R transformer, and at the current power level with the main generator in off-line, the 1M transformer is denergized leaving no power for either RCP. A manual trip is required per plant procedures.

a. Incorrect. No RCPs are running.
b. Incorrect. The automatic trips are blocked.
c. Correct. See EXPLANATION above.
d. Incorrect. See EXPLANATION above.

QUESTION # 029

Given the following:

- You are the Unit 1 Reactor Operator
- Both Units are operating at 100% Power
- All Unit 1 parameters are within their normal bands for this power level
- The plant has just received notification from TSO that System Condition RED exists
- The crew implemented AB-7, STATION AUX LOAD REDUCTION, and C20.3 AOP 12, GRID VOLTAGE OR FREQUENCY DISTURBANCES.

- One hour later several Control Room Annunciators alarmed and you observe the following indications on Unit 1:

▪ Unit 1 Main Generator Load	600MW and rising
▪ 1R X and Y winding voltages	3300VAC
▪ 1M X and Y winding voltages	3300VAC
▪ RCS Flow	95%

Which of the following is the correct next action?

a. Manually trip the Unit 1 Reactor

b. Continue to closely monitor parameters

c. Manually start D1 and D2 EDGs

d. Reduce load to prevent overpowering the Unit

ANSWER
a.
REFERENCE
AB-7; Station Aux Load Reduction, revision 4
C20.3 AOP 12; Grid Voltage of Frequency Disturbances, revision 5
ILT LP P8186L-003; 4160 & 480V Non-Safeguards Distribution, revision 3
ILT LP P8176L-008; Safeguards 4160 & 480V Electrical Distrobution, revision 6
NEW
HIGHER
K/A: 003A3.04 — Ability to monitor automatic operation of the RCPS, including: RCS flow.
EXPLANATION: The underfrequency trip of the RCP breakers occurs when one relay on both RCP busses sense frequency less than 58.2 HZ. The RCS flow indication given in the stem corresponds to less this setpoint, and the RCPs should have automatically tripped. Bus voltage had not reached the undervoltage trip setpoint (78% of nominal bus voltage for >5 seconds)

a. Correct. Automatic action has not occurred, but should have.
b. Incorrect. Plausible if the applicant fails to correlate RCS flow to bus frequency
c. Incorrect. Plausible if the applicant believes either the degraded of undervoltage thresholds have been met for the EDGs.
d. Incorrect. Plausible if the applicant believes this is the only parameter that has or will exceed a limit.

QUESTION # 030

Given the following conditions:

- Unit 1 is operating at 100% steady state
- Control Rods are in AUTO with CBD at 220 steps
- The Unit has been online for 500 days
- On the previous shift, operators made up to the RWST to raise level
- HC-110, BA TO BLENDER FLOW CONT, is in AUTO and set to current RCS boron concentration
- HC-111, RX M-U WTR TO BLENDER FLOW CONT, is in AUTO set to 45%
- Unknown to operators, the rack mounted controller HCF-111, failed to minimum output upon the realignment of the system for auto make-up to the VCT
- CS-46300, MAKEUP MODE SELECTOR switch is in AUTO
- CS-46457, BORIC ACID MAKEUP switch was placed to START
- All plant parameters are currently stable
- An auto make-up to the VCT has just occurred

Assuming no operator actions occur and ALL systems respond as designed:

1) **What is the effect on RCS TAVG?**
2) **What alarm will be expected in the Control Room?**

a. 1) Lowers
2) 47015-0404, REACTOR MAKEUP FLOW CONTROLLER DEVIATION

b. 1) No effect
2) 47015-0404, REACTOR MAKEUP FLOW CONTROLLER DEVIATION

c. 1) Lowers
2) 47013-0107, BANK D ROD WITHDRAWAL HI LIMIT

d. 1) No effect
2) 47013-0107, BANK D ROD WITHDRAWAL HI LIMIT

ANSWER
c.
REFERENCE
C12.5; Boron Concentration Control, revision 27
ARP 47013-0107, revision 44
ARP 47015-0404, revision 35
B12A; Chemical and Volume Control, revision 11
ILT LP P8172L-001a, CVCS, revision 5
NEW
HIGHER
K/A: 004K6.13 — Knowledge of the effect of a loss or malfunction on the following CVCS components: Purpose and function of the Boration/dilution batch controller
EXPLANATION
With the HCF-111 rack mounted controller output failed low and an auto makeup to the VCT demanded CV-31206 will not open, but CV-31200 will open and CV-31200 will throttle boric acid flow to the outlet of the VCT until VCT level reaches 28%. Roughly 180 gallons of boric acid will be added to the system (the VCT is approximately 16.5 Gallons/%) which will result in a significant drop in RCS Tave, especially at this time in the fuel cycle. No RMCS flow deviation alarms are expected, because RMU water flow demand is 0 GPM. The Rod Control system will step out CBD rods, until the withdrawal high limit is reached, with little or no effect on RCS Tave.

a. Incorrect. Plausible if the applicant expects CV-31200 to close, but some boric acid reaches the RCS.
b. Incorrect. Plausible if the applicant expects CV-31200 to close.
c. Correct. See EXPLANATION above.
d. Incorrect. Plausible if the applicant has a misconception as to the effects or amount of boric acid added, and the reactivity worth of CBD.

QUESTION # 031

Given the following conditions:

- Unit 1 is in Mode 2 following a refueling outage
- 1C1.2, UNIT 1 STARTUP PROCEDURE, is in progress
- Two minutes ago reactor power was stabilized at $1X10^{-8}$ amps by IRNIs to collect critical data
- All parameters are in their normal operating bands for this condition

While collecting data, the OATC notices that reactor power is rising.

Assuming MTC is NEGATIVE, which one of the following has caused the observed reactor response, and what parameter(s) could the crew monitor to verify the cause?

a. 1PT-484, MS Header Pressure transmitter has failed low, monitor MCB light indications for the steam dumps

b. A small SG tube leak has developed in one SG, monitor Steam line radiation levels

c. A leak has developed in the Seal Water Heat Exchanger tubes, monitor CC surge tank level

d. The Letdown Heat Exchanger TCV M/A station has failed to 0% output, monitor the Letdown HX outlet temperature and pressure

ANSWER
c.
REFERENCE
ILT LP P8172L-002; Component Cooling, revision 5
ILT LP P8174L-002; Steam Dump, revision 4
ILT LP P8172L-001a, CVCS, revision 5
C1B; Appendix – Reactor Startup, revision 18
1C1.2; Unit 1 Startup Procedure, revision 52
NEW
HIGHER
K/A: 004A1.10 — Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CVCS controls including: Reactor Power.
EXPLANATION: Normal VCT pressure is approximately 30 PSIG. CC System pressure is greater than 65 PSIG. If a leak were to develop in the Seal Water HX tubes, non-borated CC water would flow into the VCT, causing an RCS dilution, which would raise reactor power.

a. Incorrect. This would result in RCS temperature rising, because the MS Dumps would close. Plausible if the applicant misinterprets the failure of a MS transmitter or the operation of the MS dumps.
b. Incorrect. Charging flow would rise slightly, but as long as the RMCS controllers are set correctly, no change in RCS temperature should be observed. Plausible if the applicant confuses this choice with a small steam leak.
c. Correct. See EXPLANATION above.
d. Incorrect. The Letdown heat exchanger TCV failing closed would cause letdown temperature to rise, releasing boron from the demineralizers, and cause RCS Tave to lower.

QUESTION # 032

Given the following conditions:

- Unit 1 is currently in a refueling outage, with Core Alterations in progress
- RCS temperature is 126°F
- The ‘A’ RHR Train is in Phase II cooling
- The ‘B’ RHR Train is in Standby
- Instrument Bus IV (Yellow) was lost earlier in the shift
- The crew has implemented 1C15 AOP 3, RHR OPERATION WITHOUT CONTROL ROOM INSTRUMENTATION OF FLOW CONTROL
- The Unit 1 SS has directed you to maintain RCS temperature between 115°F and 125°F

RCS temperature can be lowered by manually ____(1)____ demand on ____(2)____, while total RHR flow is maintained less than 2000 GPM by ____(3)____ demand on ____(4)____.

	(1)	(2)	(3)	(4)
a.	Raising	1HC-624	Lowering	1HC-626A
b.	Raising	1HC-626A	Raising	1HC-624
c.	Lowering	1HC-624	Lowering	1HC-626A
d.	Lowering	1HC-626A	Raising	1HC-624

ANSWER
c.
REFERENCE
1C15; Residual Heat Removal system Unit 1, revision 36
1C15 AOP 3; RHR operation without control room instrumentation or flow control, revision 6
ILT LP P8180L-003; Residual Heat Removal System, revision 7
NEW
HIGHER
K/A: 005A4.02 — Ability to manually operate and/or monitor in the control room: Heat exchanger bypass flow control.
EXPLANATION: With the 11 RHR Pump and HX in service, RCS temperature is controlled by throttling CV-31235 using 1HC-624 manually. With the Yellow Instrument Bus lost, there is no power to the rack mounted 1FC-626A, which normally maintains total system flow automatically. Flow can be controlled manually from the CR using 1HC-626A, which is a direct acting controller.

a. Incorrect. Raising demand lowers flow through the RHR HX, thereby raising RCS temperature. Also closing CV-31237, would lower total RHR flow. Plausible if the applicant confuses the mechanisms involved, or is unaware of the reverse acting controller.
b. Incorrect. Plausible if the applicant is confuses which valve controls total flow vs. temperature.
c. Correct. Lowering demand on 1HC-624, which is a reverse acting controller, will throttle CV-31235 in the open direction, raising the cooling flow to the RCS. Since flow through the 11 RHR HX rises, total system flow will be maintained by throttling the RHR HX bypass valve, CV-31237, more closed.
d. Incorrect. This is a plausible combination of A and B.

QUESTION # 033

Given the following:

- Control room operators are responding to a RED path condition on the Core Cooling status tree
- While attempting to restore ECCS flow to the RCS, conditions degrade to the point that SGs must be depressurized

The SGs must be depressurized RAPIDLY to...

a. Prevent pressurized thermal shock of the reactor vessel

b. Prevent SG from becoming a heat source to the RCS

c. Raise ECCS flow

d. Establish conditions for starting RCPs

ANSWER
c.
REFERENCE
1FR-C.1; Response to Inadequate Core Cooling, revision 10
NEW
HIGHER
K/A: 006K3.01 — Knowledge of the effect that a loss or malfunction of the ECCS will have on the following: RCS.
EXPLANATION
Under the conditions given, a malfunction of the ECCS has resulted in an inadequate core cooling condition. Absent of restoring high pressure injection, the RCS must be depressurized to allow for the accumulators and RHR pumps to inject into the RCS. This is accomplished by rapidly depressurizing the SGs at the maximum rate and, therefore, raises ECCS flow.

a. Incorrect. PTS is not a concern in an ICC condition. Plausible if the applicant believes this is a priority over establishing some form of core cooling.
b. Incorrect. Though it may be true that the SGs are acting as a heat sink, an ICC condition exists, and the main priority is to restore some type of ECCS flow. Plausible if the applicant fails to prioritize restoration of ECCS flow correctly.
c. Correct. See EXPLANATION above.
d. Incorrect. Plausible if the applicant believes that all ECCS is lost, and misinterprets RCP starting conditions.

QUESTION # 034

Given the following conditions:

- A Large break of the Loop B Cold Leg has occurred on Unit 1
- A Loss of off-site power has occurred concurrently with the LBLOCA
- ‘A’ SI Accumulator failed to inject
- Both Unit 1 EDGs started and are powering their safeguards busses
- All other safeguards equipment has functioned as designed

Concerning the time period from the beginning of the event to when the SI pumps began to inject, which of ONE (1) statement below is true?

a. No core cooling is assumed to occur from the ECCS during this time

b. ECCS water will refill the reactor vessel to the nozzles

c. ECCS water will fill the volume outside the core barrel, the bottom plenum and about ½ the core.

d. The core can be assumed to remain covered during this time

ANSWER
a.
REFERENCE
PINGP Technical Specification Bases, revision 211
PINGP Technical Specifications, revision 201
ILT LP P8180L-005; Emergency Core Cooling System, revision 3
NEW
FUNDAMENTAL
K/A: 006K6.02 — Knowledge of the effect of a loss or malfunction on the following will have on the ECCS: Core flood tanks (accumulators)
EXPLANATION: Choice A is correct. The entire contents of one accumulator are assumed to completely pass through the break during a LBLOCA. The stem provides that one accumulator has also failed, thus during the period in QUESTION, no flow from the ECCS is occurring.

a. Correct.
b. Incorrect.
c. Incorrect. Plausible if the applicant is unfamiliar with the assumptions of the ECCS during a LBLOCA
d. Incorrect.

QUESTION # 035

Given the following conditions:

- Unit 1 is operating at 100% power
- Annunciator 47012-0406, PRZR RELIEF TANK HI TEMP/LVL/PRESS OR LO LVL has just been received.
- Current PRT conditions are as follows:
  - Level 80% and rising
  - Pressure 11 PSIG and rising
  - Temperature 100°F and rising

Which one of the following identifies the source of PRT in-leakage AND is a true statement concerning the operation of the system?

a. The RCP seal return relief;
The PRT will not pump to the HUT when the PRT drain valve is opened

b. Reactor Vessel flange leakage;
The PRT cannot be vented to the WG header to reduce pressure

c. CVCS orifice isolation relief valve;
The PRT can be vented to the WG header to reduce pressure

d. RHR suction relief due to valve leakage;
The PRT will pump to the HUT when the PRT drain valve opened

ANSWER
d.
REFERENCE
ILT LP P8170L-003; Reactor Coolant System, revision 6
ILT LP P8180L-003; Residual Heat Removal System, revision 7
ILT LP P8182L-001a; Radioactive Waste Liquid, revision 4
ARP 47012-0406, revision 46
1C4; Reactor Coolant System, revision 13
NEW
HIGHER
K/A: 007A3.01 — Ability to monitor automatic operation of the PRTS, including: Components which discharge to the PRT.
EXPLANATION: When the PRT drain AOV reaches its full open position (manually operated from the Control Room), the 11 RCDT pump will automatically start, and remains running until the PRT drain is closed.

a. Incorrect. Though this is a possible source, the 11 RCDT pump will start.
b. Incorrect. Reactor Vessel flange leakage is an input to the RCDT, not the PRT.
c. Incorrect. This is a possible source to the PRT, however the PRT vent AOV to the WG header automatically closes if PRT pressure is greater than 10 PSIG and cannot be opened from the Control Room.
d. Correct. RHR suction relief discharge is directed to the PRT, and is designed to accommodate possible valve leakage at power.

QUESTION # 036

Given the following conditions:

- Both Units are at full power, normal alignment
- Bus 25 Suddenly locks out

Which one of the following correctly describes the effect of the loss of Bus 25?

a. The 121 CR ventilation Cleanup fan will not start on an SI signal

b. The 22 CC pump auto started on low discharge pressure

c. The 121 CL Pump automatically started on low CL pressure

d. The 1st standby IA Compressor started on low IA pressure

ANSWER
b.
REFERENCE
ILT LP P8180L-009I; Control Room Ventilation & Safeguards Chilled Water Systems, revision 3
ILT LP P8172L-002; Component Cooling, revision 5
ILT LP P8176L-003; Cooling Water System, revision 5
ILT LP P8186L-008; Safeguards 4160V & 480V Electrical Distribution, revision 6
ILT LP P8178L-005; Instrument and Station Air, revision 6
NEW
FUNDAMENTAL
K/A: 008K2.02 Component Cooling Water: Knowledge of bus power supplies to the following: CCW pump, including emergency backup.
EXPLANATION

a. Incorrect. Plausible if the applicant believes that the CR cleanup fans are being supplied power from unit 2. Both Cleanup fans are normally powered from unit 1 safeguards busses via transfer switches.
b. Correct. The 22 CC pump will auto start on low discharge pressure of 65 PSIG.
c. Incorrect. Plausible if the applicant fails to recall that though the 121 CL pump receives power from bus 25 or 26, it is normally aligned to bus 25.
d. Incorrect. Plausible if the applicant believes the 121 Instrument air compressor is lost.

Question # 037

Given the following:

- Unit 1 is Mode 3
- RCS is at normal operating temperature and pressure
- Containment closeout inspections are in-progress
- One Pressurizer safety fails open

What action should be taken next?

a. Evacuate containment to control personal exposure.

b. Secure both RCPs and initiate RCS cooldown using the steam dumps to effect repairs.

c. Direct personnel in containment to open RC-22-1, Aux PRT vent to prevent PRT rupture disc failure.

d. Direct personnel in containment to close the Pressurizer safety manual isolation valve to attempt to reseat the safety valve.

ANSWER:
a.
REFERENCE:
F2; Radiation Safety, revision 33
Reactor Coolant System Lesson Plan
NEW
FUNDAMENTAL
K/A: 010 2.3.12 Pressurizer Pressure Control: – Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc. (RO 3.2)
EXPLANATION:

a. Correct. The primary concern should be safety.
b. Incorrect. Though this may be performed at some point, it is not be the next action to be taken. Additionally, in Mode 3, PINGP maintains temperature using the MS PORVs
c. Incorrect. This will not prevent rupture disc failure, and is not desirable.
d. Incorrect. There is no such valve.

Question #: 038 (BANK QUESTION: P8184L-003 #20)

Given the following initial conditions:

- Unit 1 was operating at 18% rated power.
- Pressurizer pressure blue channel (431) failed.
- All associated protective bistables have been tripped.
- Pressurizer pressure control has been returned to automatic.

An instrument failure then results in a Unit 1 reactor trip.

Which of the following instrument failures caused the Unit 1 reactor trip?

a. PT-485, Turbine 1st Stage Pressure, failed HIGH.

b. T_{hot}, red Channel, failed HIGH.

c. T_{cold}, red Channel, failed HIGH.

d. N-35, IR Nuclear Instrument, failed HIGH.

ANSWER:
b.
REFERENCE:
Reactor Protection Lesson Plan, P8184L-004
B8, Reactor Protection System, Revision 6
1C51.1, Red Bus Instrument Failure Guide, Revision 18
1C51.2, White Bus Instrument Failure Guide, Revision 19
1C51.3, Blue Bus Instrument Failure Guide, Revision 21
BANK
HIGHER
K/A: 012 A3.06 Reactor Protection: – Ability to monitor automatic operation of the RPS, including: Trip logic. (RO 3.7)
EXPLANATION:
a. Is incorrect. PT-485 failing high will not impact the reactor trip inputs.
b. Is correct. When all associated protective bistables are tripped for the failed PZR pressure blue channel, the blue channel OTdT, PZR HI Press, and PZR LO Press status lights are illuminated, indicating that 1 of the required 2-out-of-4 coincidence inputs are present for three of the reactor trips functions. When the red channel T_{hot} instrument fails high, the red channel OTdt and OPdT bistables are tripped and this completes the 2-out-of-4 coincidence for the OTdT reactor trip logic and Unit 1 reactor trips.
c. Is incorrect. T_{cold} failing high does not generate an OTdt and OPdt reactor trip input.
d. Is incorrect. There is no response for this failure since reactor power is above 10% and the IRNI trip is blocked

.

Question #: 039

Given the following:

- Unit 1 is tripped following a normal reactor shutdown.
- Both Main Feed pumps have just been shutdown.
- The 12 MD AFW pump is running.

Which one of the following signals will cause an automatic start of the 11 TDAFW pump?

a. AMSAC/DSS.

b. Train B SI signal.

c. 12 SG less than 13% NR level.

d. Loss of power to either bus 11 or 12.

ANSWER:
c.
REFERENCE:
Auxiliary Feedwater System lesson plan
NEW
FUNDAMENTAL
K/A: 013 K4.04 Engineered Safety Features Actuation: – Knowledge of ESFAS design feature(s) and/or interlock(s) which provide for the following: Auxiliary feed actuation signal.
EXPLANATION:
a. Is incorrect.
b. Is incorrect.
c. Is Correct.
d. Is incorrect.

Question #: 040

Given the following:

- Unit 1 tripped due to an MSLB.
- 11, 12, and 14 CFCUs were running in SLOW
- Unit 1 SS has directed the performance of 1E-0, Attachment L: SI Alignment Verification.
- The “SI ACTIVE” light, “13 CNTNMT FAN COIL RUNNING” is NOT lit.

1E-0, Attachment L step 1c states, “’SI ACTIVE’ lights LIT for plant conditions”, and if not “Manually align components as necessary. Note any exceptions”

Based on the plant conditions given above, what components, if any, must be manually aligned?

a. Place the 13 FCU discharge to containment dome/auto/gap CS to the DOME position, and verify proper damper alignment.

b. If the MSLB is outside of containment, the SI Active light for the 13 CFCU is not required to be lit and no further alignment is required.

c. Place the train A Cooling Water/Chilled Water valves control switch in the ISOLATE position and check the green and blue lights on.

d. Place any closed Cooling Water MVs associated with the 13 CFCU in the OPEN position, and verify the slow speed breaker is closed.

ANSWER:
d.
REFERENCE:
ILT LP P8180L009H; Safeguards Ventilation: Containment Air Handling System, revision 3
1E-0; Reactor Trip or Safety Injection, revision 28
NEW
FUNDAMENTAL
K/A: 013 A2.02 Engineered Safety Features Actuation: – Ability to (a) predict the impacts of the following malfunction or operations on the ESFAS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Excess steam demand. (RO 4.3)
EXPLANATION:
1E-0, Attachment L requires an operator to verify that SI Active lights are lit for plant conditions. Upon an SI signal, the containment air handling system is designed that the 3 Cooling Water MVs associated with each CFCU fully open, and CFCUs either start in/or shift to slow speed. When these conditions are met for each CFCU, its SI active light will illuminate. CFCUs are required to be operated in this manner for any accident that would actuate SI. The applicant must determine that all 4 CFCUs should be running in their accident mode at this time, and that the SI active light not lit is an abnormality. 1E-0 directs the operator to “manually align components as necessary”. Therefore, the facility expects its operators to understand what is necessary, to correct this situation without explicitly specific direction (i.e. from memory).

a. Is incorrect.
b. Is incorrect.
c. Is incorrect.
d. Is correct.

Question #: 041

Which of the following Engineered Safety Features Actuation System signals is NOT AUTOMATICALLY generated upon any Safety Injection signal AND IS designed to prevent containment from exceeding its design limits?

a. Containment Spray.

b. Containment Isolation.

c. Main Steam Line Isolation.

d. Containment Ventilation Isolation.

ANSWER:
a.
REFERENCE
ESFAS Lesson Plan
NEW
FUNDAMENTAL
K/A: 022 A1.01 Containment Cooling: – Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CCS controls including: Containment temperature. (RO 3.6)
EXPLANATION:
a. Is correct.
b. Is incorrect.
c. Is incorrect.
d. Is incorrect.

Question #: 042 (BANK QUESTION: P8197L-012 #151)

Given the following conditions:

- Unit 1 and 2 were operating at rated power.
- A Design Basis LOCA occured on Unit 1.
- Operators are responding per 1FR-Z.1, Response to High Containment Pressure.

Based on the operating behavior characteristics of the facility, Containment Pressure should remain below the design limit of __________, IF ___________________________.

The Containment Fan Coil Units must be running in SLOW to the DOME to ____________.

a. 23 psig, IF all 4 Containment Fan Coil Units are running.
prevent the discharge of the fans from interfering with the operation of the Containment Spray System.

b. 23 psig, IF 2 Containment Fan Coil Units are running.
prevent stalling under the high density atmospheric conditions in the Containment.

c. 46 psig, IF 1 Containment Spray Pump AND 1 Containment Fan Coil Unit is running.
prevent the discharge of the fans from interfering with the operation of the Containment Spray System.

d. 46 psig, IF 1 Containment Spray Pump AND 2 Containment Fan Coil Units are running.
prevent stalling under the high density atmospheric conditions in the Containment.

ANSWER:
d.
REFERENCE:
Containment Spray System Lesson Plan, P8180L-002
Containment Air Handling System Lesson Plan, P8180L-009H
B18D, Containment Spray System, Revision 8
B19, Containment Systems, Revision 10
BANK
FUNDAMENTAL
K/A: 026 A1.01 Containment Spray: – Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CSS controls including: Containment pressure. (RO 3.9)
EXPLANATION:
a. Is incorrect. Plausible if student confuses the CS setpoint with the design basis pressure in Containment.
b. Is incorrect. Plausible if student believes that 1 CFCU only similar to 1 spray pump is necessary to control containment pressure.
c. Is incorrect. Plausible if student does not understand the design basis of the CFCU's.
d. Is correct.

Question #: 043

Given the following conditions:

- Unit 1 startup was in progress following maintenance outage MOL.
- The Reactor was critical at the POAH.
- Heat removal was transferred to the condenser steam dump.
- 12 MD AFW Pump was running, supplying both steam generators.
- 11 TD AFW Pump was stopped in ‘SD AUTO’.
- SP 1361, Exercising Feedwater Isolation and Feedwater Check Valve Test, was in progress; Main Feedwater was not in service.
- RCS temperature was 547°F and stable.

PT-484, Main Steam Header Pressure failed high

Assuming NO actions were taken by the operating crew and all equipment responded as designed, which one of the following describes the reactor plant conditions after 15 minutes?

a. Reactor not tripped with reactor power stable at the POAH.

b. Reactor tripped with reactor power stable in the Source Range.

c. Reactor not tripped with reactor power stable at approximately 7.5%.

d. Reactor not tripped with reactor power slowly lowering in the Intermediate Range.

ANSWER:
d.
REFERENCE:
Steam Dump System Lesson Plan, P8174L-002
1C1.2, Unit 1 Startup Procedure, Revision 52
C1B, Appendix - Reactor Startup, Revision 18
NEW
HIGHER
K/A: 039 K5.08 Main and Reheat Steam: – Knowledge of the operational implications of the following concepts as they apply to the MRSS: Effect of steam removal on reactivity. (RO 3.6)
EXPLANATION:
a. Is incorrect. Plausible if correct reactivity response is realized but RCP heat input is not considered.
b. Is incorrect. Plausible if incorrectly assume condenser dump will remain open and reactor will trip on low SG levels since only the MD AFW pump will be running.
c. Is incorrect. Plausible if incorrectly assume that condenser steam dump will remain open and the MD AFW pump will maintain SG levels.
d. Is correct. All steam dumps ramp OPEN, with Steam Dump System in Steam Pressure Mode, causing Tave to lower until the LO-LO Tave Interlock is received and all steam dumps go SHUT and remain SHUT. As Tave now rises and continues to rise due to RCP heat it inputs negative reactivity and lowers power back below the POAH. The steady increase in negative reactivity will drive power through the IR and into the SR without a Reactor trip ever occurring.

Question #: 044 (BANK QUESTION: P8174L-003 #28 - MODIFIED)

While Unit 1 was operating at 70% power, a rupture of the main feedwater system occurs inside containment upstream of the check valve.

Which of the following initiates a main feedwater pump trip in response to this rupture, and what procedure is used to mitigate this event?

a. Initiated by a low steam generator pressure safeguards actuation. Perform actions of 1E-0 and go to 1E-2, FAULTED STEAM GENERATOR ISOLATION.

b. Initiated by a low-low steam generator level reactor trip. Perform actions of 1E-0 and go to 1ES-0.1, REACTOR TRIP RECOVERY, Step 1.

c. Initiated by a SGWLC system high steam flow/feed flow mismatch. Perform actions of 1C1.4 AOP 1, RAPID POWER REDUCTION UNIT 1.

d. Initiated by high containment pressure safeguards actuation. Perform actions of 1E-0 and go to 1ES-0.2, SI TERMINATION.

ANSWER:
d.
REFERENCE:
Condensate and Feedwater Lesson Plan, P8174L-003
B28B, Condensate and Feedwater System, Revision 7
1C1.4 AOP1, Rapid Power Reduction Unit 1, Revision 10
1E-0, Reactor Trip or Safety Injection, Revision 7
MODIFIED
HIGHER
K/A: 059 A2.05 Main Feedwater: – Ability to (a) predict the impacts of the following malfunction or operations on the MFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Rupture in MFW suction or discharge line. (RO 3.1)
EXPLANATION:

a. Is incorrect. Plausible if incorrectly assume that a low SG pressure generates an SI signal. Location of leak delays pressure lowering due to the leak; MFW pumps trip on High containment pressure SI signal.
b. Is incorrect. Plausible if incorrectly assume that reactor trips on LO-LO SG level and an SI signal is not generated and and not required. Location of leak delays lowering level reactor trip until after MFW pumps trip from SI signal.
c. Is incorrect. Plausible if incorrectly assume a mismatch will generate a MFW pump trip and this AOP does list an impending loss of a MFW pump as a condition requiring a rapid power reduction.
d. Is correct. High containment pressure generates an SI signal which trips the MFW pumps.

QUESTION # 045

Which one of the following is NOT a possible flowpath to the 11 SG?

a. 21 CST to 21 MDAFW pump to 11 SG

b. 22 CST to 11 TDAFW pump to 11 SG

c. 11 CST to 22 TDAFW pump to 11 SG

d. 11 CST to 12 MDAFW pump to 11 SG

ANSWER
c.
REFERENCE
B28, Auxiliary Feedwater, Figure B28B-1.
P8180L-007, Auxiliary Feedwater System, page 15-17, Rev. 4, EO-2.
BANK
FUNDAMENTAL
K/A 061K1.01: Knowledge of the physical connections and/or cause-effect relationships between the AFW and the following: S/G system.
EXPLANATION:

QUESTION # 046

The capacity of each auxiliary feedwater pump is 200 gpm to the steam generators at a discharge pressure of 1300 psia.

What is the MINIMUM auxiliary feedwater flow needed to compensate for decay heat following a reactor scram caused by a loss of main feedwater?

a. 200 gpm at 1300 psia

b. 180 gpm at 1100 psia

c. 160 gpm at 1300 psia

d. 140 gpm at 1100 psia

ANSWER
c.
REFERENCE
USAR Section 11.9 page 2
Lesson Plan P8180L-007
NEW
FUNDAMENTAL
K/A RO: 061K5.02: Knowledge of the operational implications of the following concepts as they apply to the AFW: Decay heat sources and magnitude.
EXPLANATION:

a. This is the design of the system.
b. This is more than that required for decay heat.
c. To prevent thermal cycling and removing the decay heat requires about 160 gpm per unit.
d. Insufficient AFW flow.

QUESTION # 047

The fire brigade is composed of the ____________ as the Fire Brigade Chief and total of _____ fire brigade members, but NOT including the ____ members of the minimum shift crew for safe shutdown of the reactors.

a. Unit 1 Turbine Building APEO; 5; 6

b. Unaffected unit shift supervisor; 5; 6

c. Unit 1 Turbine Building APEO; 7; 5

d. Unaffected unit shift supervisor; 7; 5

ANSWER
a.
REFERENCE
P9150L-004
BANK
MEMORY
K/A 062 2.4.25: AC Electrical Distribution Knowledge of fire protection procedures.
EXPLANATION:
Bank Question

QUESTION # 048

Unit 1 was operating at 100% power when the following annunciators energized:

- **47024-1102**, 11 DC SYSTEM TROUBLE
- **47024-1201**, 11 DC PANEL UNDERVOLTAGE
- **47024-1003**, D1 EMERGENCY GENERATOR LOSS OF CONTROL VOLTAGE
- **47018-0501**, SAFEGUARD LOGIC TRAIN A DC FAILURE

Of the choices given below, with no operator action, which of them occurs as a result of the event that caused these annunciators?

1. Automatic start of 11 TDAFW pump.
2. Turbine Trip with Automatic Generator Breaker Trip
3. Non-safeguards buses 11, 12, 13, and 14 will de-energize.
4. Automatic transfer of DC control power to 4KV busses 11 and 12 to the standby source, DC panel 21.

a. 1 and 3 ONLY

b. 2 and 4 ONLY

c. 2 and 3 ONLY

d. 1 and 4 ONLY

ANSWER
d.
REFERENCE
LOSS OF UNIT 1 TRAIN “A" DC, 1C20.9 AOP1, Rev 6
NEW
HIGHER
K/A 063 K3.02: DC Electrical Distribution Knowledge of the effect that a loss or malfunction of the DC electrical system will have on the following: Components using DC control power.
EXPLANATION:

a. The non-safeguards buses do not auto de-energize Distractor a. is incorrect.
b. An automatic generator breaker trip does not occur. Distractor b. is incorrect.
c. An automatic generator breaker trip does not occur. Distractor c. is incorrect.
d. Distractor d. is correct.

QUESTION # 049

Given the following conditions:

- Unit 1 is at 100% power.
- SP 1093, D1 Diesel Generator Monthly Slow Start Test, is in progress.
- D2 is in its normal standby mode.
- While unloading D1, the "Motoring Current" relay actuates.
- Prior to any operator action, a Loss of Offsite Power occurs CONCURRENTLY with an automatic Safety Injection actuation.

With NO operator action, what is the status of D1? D1 will . . .

a. stop AND not restart.

b. continue to run unloaded.

c. stop, restart, and run unloaded.

d. continue to run AND will re-power Bus 15.

ANSWER
a.
REFERENCE
8186L-004-09
BANK
FUNDAMENTAL
K/A 064A2.16: Ability to (a) predict the impacts of the following malfunction or operations on the ED/G system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of offsite power during full-load testing of ED/G.
EXPLANATION:

a. Bank – indicated correct answer.
b. D1 will not continue to run.
c. D1 will not restart.
d. D1 will not continue to run.

QUESTION # 050

You are in an area for four hours where the airborne radioactivity concentration is 0.25 DAC of a deterministically limiting radionuclide.

What is your estimated internal dose?

a. 0.25 mrem

b. 2.5 mrem

c. 25 mrem

d. 250 mrem

ANSWER
c.
REFERENCE
NRC GET Exam
BANK
HIGHER
K/A 073K5.03: Process Radiation Monitoring Knowledge of the operational implications as they apply to the PRM system: Relationship between radiation intensity and exposure limits.
EXPLANATION:

QUESTION # 051

Given the following conditions:

- Unit 1 is at 100% power.
- A small tube leak in 11 Steam Generator has existed for 180 days.
- Steam Generator Blowdown is aligned to the river for chemistry concerns.
- 1R-19, Steam Generator Blowdown radiation monitor goes into HIGH alarm.

What control room indications will indicate that the proper automatic actions have occurred?

a. 11 SGB ONLY isolation motor valves indicate CLOSED.
   11 SGB ONLY flow on recorder 42040 indicates ZERO.

b. 11 AND 12 SGB flow on recorder 42040 indicates ZERO.

c. 11 AND 12 SGB isolation motor valves indicate CLOSED.
   11 AND 12 SGB flows on recorder 42040 indicate ZERO.

d. S/G blowdown is now routed through 11 SGBD HX.

ANSWER
b.
REFERENCE
P8182L-003-83
MODIFIED
FUNDAMENTAL
K/A 073A4.01: Ability to manually operate and/or monitor in the control room: Effluent release.
EXPLANATION:

a. No closed indication on the isolation motor valves.
b. correct
c. No closed indication on the isolation motor valves.
d. Blowdown is isolated, not re-routed.

QUESTION # 052

Given the following conditions:

- Unit 1 has tripped and SI has actuated
- The slave relays for the SI actuation of the Cooling Water system Train B have failed to operate.
- All other systems respond normally.

What is the status of 12 Component Cooling Water Heat Exchanger if NO operator action is taken?

a. There is NO cooling water flow through the heat exchanger

b. The outlet FCV strokes open but open travel is limited by the travel stop, and the valve does not modulate based on temperature

c. The outlet FCV opens to control temperature and is NOT limited by the travel stop in opening

d. The outlet FCV opens to control temperature but maximum open position is limited by the travel stop

ANSWER
d.
REFERENCE
P8176L-003-22
BANK
HIGHER
K/A 076 K3.01: Knowledge of the effect that a loss or malfunction of the SWS will have on the following: Closed cooling water.
EXPLANATION:
BANK Question

QUESTION # 053

Given the following:

- Both Units were operating at 100% rated thermal power.
- A pressurizer pressure instrument failed low on Unit 2.
- Two I&C technicians were sent to troubleshoot the failed instrument, but got on the wrong instrument and initiated a low pressurizer pressure signal from that instrument.
- As a result, SI Train A and Train B initiated on Unit 2.

MV-32036 and MV-32037 - CL Header Isolation Valves
MV-32159 and MV-32144 - CL Header Crossover Valves

What is the status of the Cooling Water System after the SI signal?

a. MV-32036 and MV-32037 - shut
MV-32034 and MV-32035 - shut
The 121 CL pump starts, then shuts down when Diesel Driven pumps reach 400 rpm
Both Diesel Driven CL pumps start and continue to run

b. NV-32036 and MV-32037 - shut
MV-32034 and MV-32035 - open
The 121 CL pump starts if CL header pressure declines to 80 psig
Both Diesel Driven CL pumps start if CL header pressure declines to 75 psig

c. MV-32036 and MV-32037 - open
MV-32034 and MV-32035 - shut
The 121 CL pump starts and aligns to Unit 2
The Diesel Driven CL pumps start and align to their respective Units

d. MV-32036 and MV-32037- shut
MV-32034 and MV-32035 - open
The 121 CL pump starts and aligns to Unit 2
Both Diesel Drive CL pumps start and align to their respective Units

ANSWER
c.
REFERENCE
Initial License Training Lesson Plan P8176L-003
NEW
HIGHER
K/A 076K4.06: Knowledge of CWS design feature(s) and/or interlock(s) which provide the following: Cooling water train separation.
EXPLANATION:
MV-32034 and MV-32035 receive close signals making distractors b. and d. incorrect. The 121 CL pump aligns to the affected unit and continues to run, making distractor a. incorrect. Distractor c. is the correct answer.

QUESTION# 054

Given the following initial conditions:

- 121 & 122 IA compressors running in PREFERRED
- 123 IA compressor in FIRST STANDBY
- 124 SA compressor running in PREFERRED
- 125 SA compressor in STANDBY
- Normal valve lineup

A break in the Unit 2 Auxiliary Building Instrument Air line occurs which causes the Instrument Air header pressure to rapidly decrease to <75 psig.

Which of the following correctly describes the automatic actions which would occur due to this failure? 123 IA Compressor starts at ____ psig and MV-32315 (STA AIR HDR ISOL MV B) closes to separate U-1 and U-2 Instrument Air headers at _____ psig IA header pressure.

a. 95; 80

b. 95; 85

c. 90; 85

d. 90; 80

ANSWER:
d.
REFERENCE
P8178L-005 – 019
BANK
FUNDAMENTAL
K/A 078A4.01 Ability to manually operate and/or monitor in the control room: Cross-tie valves with IAS.

Question #: 055

Given the following conditions:

- SI automatically actuated on Unit 1.
- Containment pressure is 5 PSIG and slowly lowering.
- Various containment isolation valves did not close automatically.
- Both CI manual switches were momentarily placed in the ACTUATE position.
- The crew is now terminating SI.

Which of the following below describes the MINIMUM steps necessary to allow opening of the containment isolation valves?

a. Depress the CI reset pushbuttons ONLY.

b. Depress the SI reset pushbuttons, THEN the CI Pushbuttons.

c. Reduce containment pressure to less than 3.5 PSIG, THEN depress the CI reset pushbuttons.

d. Reduce containment pressure to less than 3.5 PSIG, THEN depress the SI reset pushbuttons, and THEN depress the CI reset pushbuttons.

ANSWER:
a.
REFERENCE
ESFAS lesson plan, revision 4
1ES-0.2; SI Termination, revision 23
NEW
FUNDAMENTAL
K/A: 103 K1.08 Containment: – Knowledge of the physical connections and/or cause-effect relationships between the containment system and the following systems: SIS, including action of safety injection reset. (RO 3.6)
EXPLANATION:
According to the logic diagrams supplied in the ESFAS lesson plan
a. Is correct.
b. Is incorrect.
c. Is incorrect.
d. Is incorrect.

QUESTION # 056

Given the following conditions:

- Unit 1 has just started up, following a refueling outage
- Unit 1 Main turbine load is 60 MWe
- T_{ave} is 547°F
- PRNI A = 10%
- PRNI B = 9%
- PRNI C = 8%
- PRNI D = 9%
- The Rod Control Selector switch is in AUTO
- Steam Dumps are in STM PRESS mode
- The Steam Dump controller is set to 1050 PSIG

If the Main Generator trips…

a. Rods will only step in at 8 steps/minute

b. Rods will step in, then out at 8 steps/minute

c. Rods will not move

d. The Reactor Trip breakers will open automatically

ANSWER
a.
REFERENCE
ILT LP P8184L-004; Reactor Protection, revision 8
ILT LP P8184L-005; Rod Control and Rod Position Indication, revision 4
1C1.2; Unit 1 Startup Procedure, revision 52
NEW
HIGHER
K/A: 001K4.20 – Knowledge of CRDS design feature(s) and/or interlock(s) which provide for the following: The permissives and interlocks associated with increase from zero power
EXPLANATION: On Unit 1, Permissives P-7, 8 and 9 are all associated with PRNI power level. Permissive, P-2 requires turbine impulse pressure, 1PT-485, to be greater than 15% equivalent steam pressure to allow automatic Control Rod withdrawal. P-9 unblocks the turbine trip reactor trip when 2 of 4 PRNI are >10%

a. Correct. With the MS dumps set to 1050 PSIG, Tave will rise to 551°F when the main turbine trips. With impulse pressure at 0%, control rods will initially step in at 8 steps/minute due to a 4°F mismatch.
b. Incorrect. Low Power interlock, P-2, will not allow rods to automatically step out unless impulse pressure is greater than 15%.
c. Incorrect. Plausible if the applicant believes that all rod motion is blocked due to this situation.
d. Incorrect. Turbine trip/Reactor trip, P-9, is blocked until power is greater than 10% on 2 of 4 PRNI channels.
   Notes: during a S/U the dumps are set to 1005# in STM PRESS, and shifted to TAVG around 15%

QUESTION # 057

Given the following conditions:

- Unit 1 has experienced a Loss of Offsite Power with a SGTR in the 11 SG
- The Crew has implemented 1E-3, STEAM GENERATOR TUBE RUPTURE and is in the progress of cooling down the RCS at the maximum obtainable rate
- The STA has just announced that the INTEGRITY Critical Safety Function Status Tree is ORANGE due to the A RCS LOOP cold leg temperature dropping rapidly.

Which one of the following describes the reason the A RCS LOOP cold leg temperature has dropped rapidly?

The A RCS Loop flow has…

a. Increased because the cooldown has increased natural circulation and is moving cold 11 SG U-tube water past the cold leg RTD.

b. Restarted. Natural circulation has set up, causing a sudden rise then rapid drop in temperature as the stagnant water from the hot leg is flushed through the loop.

c. Reversed due to the increase in SI flow, causing the cold water from B RCS loop to pass over the cold leg RTD in the A RCS loop.

d. Stopped, allowing the cold SI flow to pass over the cold leg RTD.

ANSWER
d.
REFERENCE
1E-3, Steam Generator Tube Rupture, revision 22
NEW
HIGHER
K/A: 002K5.13 – Knowledge of the operational implications of the following concepts as they apply to the RCS: Causes of circulation
EXPLANATION
While the RCS is being cooled down on natural circulation during a SGTR event, reverse flow through the ruptured loop during the cooldown or when the Pressurizer PORV is opened to depressurize the RCS is possible and could cause the SI flow path in the ruptured loop to change. This change in the SI flow path could result in an indicated cold leg temperature (due to the location of the cold leg RTD) that decreases to the point that the symptoms for 1FR-P.1 would occur. This false indication would only be seen in the ruptured loop since it is essentially stagnant while the other loop is circulating by natural circulation.

a. Incorrect. Flow in the A RCS loop will not increase. Plausible if the applicant believes that natural circulation is occurring in the loop with the ruptured SG, and is being enhanced by steaming the SG.
b. Incorrect. Flow will remain relatively stagnant. Plausible if the applicant believes that conditions for nautral circulation has now been established by commencement of the cooldown, or misunderstand which SG is being cooled down.
c. Incorrect. Though flow may somewhat reverse in the loop with the RSG, cold water from the B loop cannot directly pass over to the A cold leg. Plausible if the applicant recognizes that flow can reverse, but misinterprets the RCS flow path.
d. Correct. See EXPLANATION above.

QUESTION # 058

Given the following conditions:

- A Small break LOCA has occurred on Unit 1
- Both RCPs have been tripped
- Subcooling is 0°F
- ALL Pressurizer level indication has been lost

Based on the indications above, which one of the following statements is true?

a. The crew will be unable to determine if SI must be reinitiated if terminated

b. The crew will be unable to determine if the conditions necessary for natural circulation exist

c. The crew will be unable to quantify the actual RCS leak rate

d. The crew will be unable to terminate SI, when required

ANSWER
c.
REFERENCE
PINGP Technical Specification Bases, revision 211
1E-1; Loss of Reactor or Secondary Coolant, revision 23
NEW
HIGHER
K/A: 011K6.05 – Knowledge of the effect of a loss or malfunction on the following will have on the PZR LCS: Function of PZR level gauges as post accident monitors
EXPLANATION
The TS Bases states that with regard to post accident instrumentation, Pressurizer level is used to determine whether to terminate SI, if still in progress; or to reinitiate SI if it has been stopped, and is used to verify the unit conditions necessary to establish natural circulation in the RCS.

a. Incorrect. Subcooling is also used to determine if SI must be reinitiated. Plausible if the applicant does not understand generally when SI should be terminated.
b. Incorrect. Plausible if the applicant does not know all indications necessary to determine if natural circulation can exist.
c. Incorrect. Plausible in the applicant is unaware of other methods available to an operator to determine actual the RCS leak rate
d. Correct. Pressurizer level is used to determine SI termination criteria in most EOPs, and should be used in this case.

QUESTION # 059

Given the following conditions:

- Unit 1 is at full power
- Last shift, a power supply failed in PRNI N44
- 1C51.4, YELLOW BUS (114) INSTRUMENTS FAILURE GUIDE has been implemented
- All appropriate actions have been taken for the failed NI and I & C is troubleshooting
- Multiple Annunciators begin alarming in the Control Room
- Unit 1 Turbine Building APEO reports a fire in Instrument Bus 111 (White)

Which one of the following statements correctly describes the response of the NIs?

a. N31, N35, and N41 are lost, N32 will not automatically energize below P-6, but can be manually unblocked

b. N31 and N32 will not automatically energize below P-6, and cannot be unblocked

c. N31 will automatically energize below P-6, N32, N36, and N42 are de-energized

d. N32 can be unblocked below P-6, N42 and N44 front panel lights are de-energized

ANSWER
b.
REFERENCE
ILT LP P8184L-002; Nuclear Instrumentation System, revision 7
ILT LP P8186L-015; Safeguard Distribution 120 VAC Instrumentation, revision 2
1C51.2; Instrument Failure Guide, revision 19
1C20.8 AOP 1; Abnormal Operation, Instrument AC Inverters, revision 10
NEW
HIGHER
K/A: 015K2.01 – Knowledge of bus power supplies to the following: NIS channels, components, and interconnections
EXPLANATION
Actions taken following a failure of N44 include the tripping of bistables for that channel, and is accomplished by pulling control power and instrument power fuses for that channel. Instrument Bus 111 supplies power to N32, N36, and N42. When power is lost to Instrument Bus 111, the reactor will trip and permissive P-10 will not allow either SRNI N31 or N32 to automatically energize as they normally would once reactor power has decayed below the permissive P-6 setpoint. Additionally, P-6 cannot be manually unblocked as long as P-10 will not allow.

a. Incorrect. N31, N35, and N41 are powered from Instrument Bus 112. Plausible if the applicant correlates Instrument Bus 111 with the first NI stack. Also N32 has no power, and neither SRNI could be manually unblocked with P-10 in its current state.
b. Correct. See EXPLANATION above.
c. Incorrect. See EXPLANATION above. Plausible if the applicant fails to understand the state of P-10 in this configuration.
d. Incorrect. See EXPLANATION above. Plausible if the applicant recognizes the state of P-10, but has a misconception related to the ability to override certain permissives.

QUESTION # 060

The Unit 1 reactor is operating at 100% in a normal alignment and configuration when the following events occur. All other equipment functions as designed.

- 1A T_{HOT} Instrument Loop failed HIGH
- Unit 1 reactor tripped
- Unit 1 turbine tripped but only one stop valve closed
- PT-484 fails **as is** immediately prior to the Turbine trip

Assuming no operator action, what will RCS temperature be?

a. 540°F

b. 547°F

c. 550°F

d. 552°F

ANSWER
a.
REFERENCE
Steam Dump Lesson Plan
NEW
HIGHER
K/A 041 Steam Dump A1.02 – Ability to predict and/or monitor changes in parameters associated with operating the SDS controls including: Steam Pressure
EXPLANATION
a. Correct with 1A instrument loop failed high actual RCS temp will drop until low temp interlock closes steam dumps at 540 F
b. When system works normally RCS temp should be held at 547°F
c. Temp corresponds to S/G PORV lift setpoint, plausible if applicant thinks Steam Dump System functions in such a manner where system would be controlled by S/G PORV
d. Temp corresponds to S/G Safety lift setpoint, plausible if applicant thinks Steam Dump System functions in such a manner where system would be controlled by S/G Safeties.

QUESTION # 061

The Unit 1 reactor was operating at 100% in a normal full power alignment when Bus 14 experienced a ground fault resulting in a bus lockout.

How does 1) the plants secondary system respond to the loss of Bus 14 and, 2) how will the crew respond to this situation?

a. 1) Condensate Pump #11 will be running as will Feedwater Pump #11
The Crew will implement 1C11.4 AOP1, Rapid Power Reduction

b. 1) Condensate Pump #11 will be running as will Feedwater Pump #12
The Crew will implement 1E-0, Reactor Trip or Safety Injection

c. 1) Condensate Pumps #11and #13 will be running as will Feedwater Pumps #11 and #12
The Crew will implement 1C11.4 AOP1, Rapid Power Reduction

d. 1) Condensate Pumps #12 and #13 will be running as will Feedwater Pump #11 and #12
2) The Crew will implement 1E-0, Reactor Trip or Safety Injection

ANSWER
b.
REFERENCE
B28A Condensate and Feedwater System, 1C28.3 Unit 1 Condensate System
NEW
HIGHER
K/A Ability to (a) predict the impacts of the following malfunctions or operations on the Condensate System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of Condensate Pumps.
EXPLANATION
a. Incorrect, when 2 out of 3 condensate pumps trip when both feedwater pumps were running the #11 feedwater pump trips on Unit 1.
b. Correct condensate pump #11 is fed from bus 13, condensate pumps #12 & #13 are fed from bus 14. When 2 out of 3 condensate pumps trip when both feedwater pumps were running the #11 feedwater pump trips on Unit 1.
c. Incorrect, applicants may confuse power supplies for condensate pumps and may confuse which feedwater pump trips when 2 out of 3 condensate pumps trip.
d. Incorrect, applicants may confuse power supplies for condensate pumps and may confuse which feedwater pump trips when 2 out of 3 condensate pumps trip.

QUESTION # 062

The following plant conditions exist at 0930.

- RCDT Level increases to 52%
- The #11 RCDT Pump starts
- Containment pressure is 0.6 psig
- Pressurizer Level is 59% and stable

At 0931 the following plant conditions are observed.

- Pressurizer Level is **decreasing** at 2%/min
- Pressurizer Pressure is **decreasing** and is at 1790 psig
- Containment Pressure is 0.6 psig and stable
- RCDT Level is 83% and **increasing**

Based on the above stated information what is the status of the RCDT pump / pumps?

a. Neither RCDT Pump is running

b. #11 RCDT Pump is running with the #12 RCDT Pump in Standby

c. #12 RCDT Pump is running with the #11 RCDT Pump is Standby

d. Both the #11 and #12 RCDT Pumps are running

ANSWER

a.

REFERENCE

P8182L-001a, Radioactive Waste Liquid Lesson Plan

NEW

HIGHER

K/A 068 Liquid Radwaste A3.02 – Ability to monitor automatic operation of the Liquid Radwaste System including: Automatic isolation

EXPLANATION

a. Correct – Based on Pressurizer Pressure an SI occurred which caused a Containment Isolation Signal, this closed the RCDT outlet isolation valves and cause the RCDT pump to trip.
b. Incorrect – Plausible if the applicant does not realize the Cnmt. Isol. has occurred and that the #12 RCDT pump doesn't start until 85% RCDT level
c. Incorrect – Plausible if the applicant does not realize the Cnmt. Isol. has occurred and mixes up which RCDT pump starts first.
d. Incorrect – Plausible if the applicant doesn't realize the Cnmt. Isol. has occurred because if it hadn't both RCDT pumps would be running.

Question # 063

Given the following:

- The Aux Building APEO is in the Control Room (CR).
- The on-line WGDT pressure is 110 psig and rising.

Which of the following indications in the CR would alert operators that WGDT swap-over has occurred?

a. 47015-0603, RAD WASTE BLDG LOCAL ALARM.

b. 47015-0503, LIQUID WASTE DISPOSAL LOCAL ALARM.

c. 47015-0101, WASTE DISPOSAL BORON RECYCLE LOCAL ALARM.

d. 47012-0406, PRZR RELIEF TANK HI TEMP/LVL/PRESS OR LO LVL.

ANSWER:
c.
REFERENCE:
Various 47015, and 47012 ARPs
ILT LP P8182L-001C, Radioactive Waste Gas, revision 4
NEW
HIGHER
K/A: 071 Waste Gas Disposal A4.09 – Ability to manually operate and/or monitor in the control room: Waste gas release rad monitors.
EXPLANATION: The WGDT pressure control valve will isolate the on-line WGDT at 110 psig, and aligning flow to the standby WGDT, (if selected). A local alarm will be received at the waste disposal panel, and if not acknowledged after 30 seconds the waste disposal boron recycle local alarm will be received in the CR.

a. Incorrect. Plausible similar alarm.
b. Incorrect. Plausible similar alarm.
c. Correct. See explanation above.
d. Incorrect. Possible misconception.

Question # 064

Given the following conditions:

- Unidentified leakage from SP 1001AA, “Daily Reactor Coolant System Leakage Test” has been rising.
- 1R-7 has just alarmed.

Based on the conditions given, what procedure should be implemented?

a. 1E-0, “Reactor Trip or Safety Injection”.

b. 1C4 AOP 1, “Reactor Coolant Leakage”.

c. 1C4 AOP 2, “Steam Generator Tube Leak”.

d. 1C14 AOP 2, “Leakage Into the Component Cooling System”.

ANSWER:
b.
REFERENCE:
1C4 AOP 1; Reactor Coolant Leak, revision 13
8182L-002; RD list
NEW
HIGHER
K/A: 072 Area Radiation Monitoring 2.4.11 – Knowledge of abnormal condition procedures.
EXPLANATION:
1R-7 is the Seal Table ARM. Entry conditions for 1C4 AOP 1 include several radiation monitors, including 1R-7, and the daily leak rate surveillance. ROs are required to recognize entry conditions for AOPs.

a. Incorrect. Plausible misinterpretation of the conditions given in the stem. Nothing in the stem indicated such action is warranted.
b. Correct. See explanation above.
c. Incorrect. Plausible misinterpretation of the conditions given in the stem. A SGTL will not result in a Seal table ARM alarm.
d. Incorrect. Plausible misinterpretation of the conditions given in the stem. An RCS to CC leak will not produce an alarm on 1R-7.

Question # 065

The Emergency Intake Bay is backwashed by using water from the discharge of the...

a. Cooling Water pumps.

b. Deicing Water pumps.

c. Cooling Tower pump 121 or 122.

d. Circulating Water pumps 12 and 21.

ANSWER:
d.
REFERENCE:
ILT LP P8176L-002a, Circulating Water and Cooling Towers, revision 4
BANK P8176L-002a, #11
FUNDAMENTAL
K/A: 075 Circulating Water K1.01 – Knowledge of the physical connections and/or cause effect relationships between the circulating water system and the following systems: Emergency/essential SWS
EXPLANATION:
The discharge of the CW pumps 12 & 21 can be valved to the emergency intake bay forcing flow back to the river.

a. Inorrect
b. Incorrect
c. Incorrect
d. Correct. See explanation above.

Question #: 066

In accordance with SWI O-10, "Operations Manual Usage", while performing abnormal or emergency procedures, the Reactor Operator may perform __(1)__ from memory, until: ___(2)___.

	(1)	(2)
a.	Immediate operator actions	The SS has obtained the appropriate procedure, and is ready to use the procedure, beginning with step 1.
b.	Continuous Action Steps	The SS has found, and read aloud the particular Continuous Action Steps.
c.	Notes and/or cautions	The SS has reached, and read aloud the particular note and/or caution.
d.	Continuous Action Steps	The SS has obtained the appropriate procedure, and is ready to use the procedure, beginning with step 1.

ANSWER:
a/
REFERENCE:
SWI O-10; Operations Manual Usage, revision 50
NEW
FUNDAMENTAL
K/A: 2.1.2 Conduct of Operations: – Knowledge of operator responsibilities during all modes of plant operation. (RO 4.1)
EXPLANATION:

a. Is correct. According the SWI O-10, the Control Room operators shall start to perform immediate action steps from memory until the earlier of: a) Completion of all immediate action steps of appropriate procedure; OR b) The SS (or other procedure reader) has obtained the
b. appropriate procedure, and is ready to use the procedure, in hand, beginning with Step 1.
c. Is incorrect. Continuous Action Steps are not performed from memory until "found" but rather once they become applicable. (i.e. when first encountered until superseded, or stated inapplicable)
d. Is incorrect. Notes and/or cautions are not performed from memory prior to encountering them in a procedure.
e. Is incorrect. Similar to distractor b.

Question #: 067

Given the following for Unit 1

- Rod G-11 (CB-D) is at 172 steps.
- All other control bank D rods are at 146 steps.
- All control bank D have been determined to be OPERABLE.

Which of the following describes the method for realigning rod G-11 to control bank D per 1C5 AOP5, "Misaligned Rod, Stuck Rod, and/or RPI Failure or Drift"?

a. Withdraw control bank D to 160 steps while maintaining rod G-11 stationary.

b. Alternately insert rod G-11 and withdraw the bank until alignment is achieved.

c. Insert rod G-11 to 146 steps while maintaining all other rod in the control bank stationary.

d. Fully insert all four control bank D rods into the core, then withdraw them to their initial position.

ANSWER:
c.
REFERENCE
Facility LXR test bank questions
BANK
FUNDAMENTAL
2.1.23 Conduct of Operations: – Ability to perform specific system and integrated plant procedures during all modes of plant operation. (RO 4.3)
EXPLANATION:
This is a bank question. The stem and distracters were modified for clarity, then reformatted. These changes were not significant.

a. Is incorrect.
b. Is incorrect.
c. Is correct.
d. Is incorrect.

QUESTION# 068

Which of the following, under the conservative decision making process, would require an IMMEDIATE action?

a. A power range nuclear instrument fails low when the plant is raising power.

b. Both source range instruments de-energize while power is below P-6.

c. ‘1B’ Steam generator narrow range level is constant at 50% while power level is decreasing during a shutdown.

d. An I&C technician enters the control room an smells of alcohol.

ANSWER
b.
REFERENCE
FP-OP-COO-01, Revision 10, Attachment 5, Conservative Decision Making
NEW
HIGHER
K/A 2.1.39 Knowledge of conservative decision making practices.
EXPLANATION
a. Incorrect - Enter the applicable technical specification
b. Correct – Immediate scram is required per technical specifications and conservative decision making process.
c. Incorrect – Enter the applicable off-normal procedure
d. Incorrect – Immediate action required, but not covered under the station's conservative decision making program.

QUESTION # 069

Given the following conditions:

- Unit 1 is at 100% power
- At 0735, the Unit 1 RO initiated SP 1001B, UNIT 1 CONTROL ROOM LOG - MODES 1 AND 2 and began taking period 2 readings, starting with the NI readings
- ALL period 2 Tech Spec required NI parameters were within the acceptance criteria of SP 1001B
- At 0830 I & C began a routine surveillance on Unit 1 PRNI N43, which removes it from service
- At 1230 the I & C technicians completed their surveillance on N43, and returned the NI to service
- At 1415 the I & C supervisor noted a Tech Spec required parameter was outside the acceptance criteria of the N43 surveillance and verbally notified the Unit 1 SS of the condition
- At 1545 the Unit 1 RO completed SP 1001B

Based on the above conditions the Tech Spec Required Action completion time clock for the PRNI N43 inoperability started at…

a. 0735

b. 0830

c. 1415

d. 1545

ANSWER
b.
REFERENCE
PINGP Technical Specifications, revision 201
PINGP Technical Specification Bases, revision 211
NEW
HIGHER
K/A: 2.2.23 Ability to track Technical Specification limiting conditions for operations.
EXPLANATION
The tech spec surveillance was removed from service at 0830, starting the period of inoperability. This condition remains until the review of surveillance data is completed. From Tech Spec 3.0.2, SR 3.0.1, and Bases for TS 3.0.2, and SR 3.0.1 time of LCO entry is at time of discovery of failure to meet LCO, and failure to meet the TS constitutes inoperability, in this case.

a. Incorrect. Plausible if the applicant believes that the duration of inoperability begins at the last known time of operability.
b. Correct. See EXPLANATION above.
c. Incorrect. Plausible if the applicant believes that operability exists prior to the surveillance receiving the proper reviews and, therefore discovery time would be the time of notification.
d. Incorrect. Plausible if the applicant believes the LCO is met until the end of the duration of the CR surveillance.

QUESTION # 070

Given the following conditions:

- Both Units are operating at full power, normal alignment
- Annunciator 47016-0204, “11 RWST Level Low” has alarmed 15 times in the past minute
- RWST levels are 92% and stable on both 1LI-920 and 1LI-921
- The annunciator has now been silenced, but not acknowledged

Which of the following is required concerning the process to track this condition?

a. Include the words “Potential OWA” in the AR One-Line Description

b. Ensure that the SS or SM adds this issue to the “Top 10 Equipment List”

c. Record the number of times the annunciator re-flashes on the Unit 1 Narrative Log

d. Write the AR number on a annunciator deficiency sticker and place it on the annunciator tile

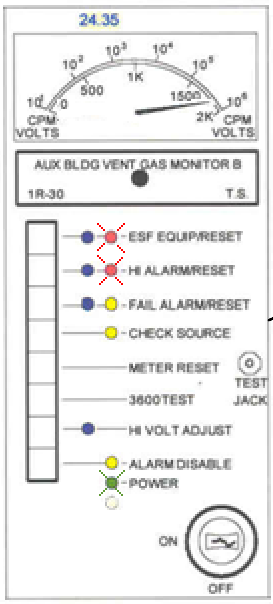
Answer
a.
References:
5AWI 3.10.8; Equipment Problem Resolution Process, revision 13
FP-OB-COO-001; Conduct of Operations, revision 10
SWI O-2; Shift Organization, Operation & Turnover, revision 73
ARP 47019-0403, revision 31
ARP 47016-0204, revision 39
NEW
FUNDAMENTAL
K/A: 2.2.43 Knowledge of the process used to track inoperable alarms.
Explanation:

a. Correct. 5AWI 3.10.8, states that all plant operators are responsible for identifying potential operator work-arounds in this fashion.
b. Incorrect. A separate individual has this assignment. A plausible misconception of the process.
c. Incorrect. No such requirement exists.
d. Incorrect. Plausible misconception, as individuals may have observed this practice, however, not a requirement.

QUESTION # 071

Given the following conditions:

- Both Units are at full power
- A Waste Gas release is in progress
- ANNUNCIATOR 47022-0108; AUX BLDG VENT GAS High Radiation Level MONITOR B, has just come in
- You observe the following indications on the 21 RM Cabinet:



- RED ESF EQUIP/RESET LED is ON
- RED HI ALARM/RESET LED is ON
- GREEN POWER LED is ON
- NO other LEDs are LIT

Based on the indications above, which one of the following statements is true?

a. The 121 Aux Bldg Special Vent exhaust fan has started, and a filtered and monitored release is in progress

b. WG Release Pressure Control Valve, CV-31271 has closed, stopping the release

c. If 1R-37 also reads a low value, the check source for 1R-30 may be mechanically stuck

d. Depressing the meter reset pushbutton will cause the RAD MONITOR DOWNSCALE FAILURE PANEL ALARM annunciator

ANSWER
d.
REFERENCE
ARP 47048 1R-30, revision 35
ARP 47022-0108, revision 37
ARP 47022-0208, revision 46
ILT LP P8182L-002; Radiation Monitoring System, revision 8
B11; RADIATION MONITORING SYSTEM, revision 10
NEW
HIGHER
K/A: 2.3.15 Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.
EXPLANATION

a. Incorrect. 1R-30 starts the 122 ABSV exhaust fan. Plausible if the applicant confuses which train is started by each rad monitor.
b. Incorrect. This would be correct for the Unit 2 Aux Building gas monitors.
c. Incorrect. The Aux Building gas monitors are the only NMC monitors that are not equipped with a check source.
d. Correct. This is true for all NMC monitors.

QUESTION # 072

Given the following conditions:

- Unit 2 is in Mode 2 following a refueling outage
- A dilution to criticality is in progress
- A small leak has developed in the Unit 2 RCS
- The Unit 2 RO has just declared the reactor critical
- Flux mapping equipment is energized and warming up

If the Unit 2 SS wants to send an extra SRO into containment to investigate, what additional requirements must be met to allow the containment entry on Unit 2?

a. The extra SRO must be accompanied by at least one other Operations Department individual

b. The flux mapping equipment must be secured and placarded

c. All reactivity changes must cease, or the reactor must be tripped

d. The extra SRO must be accompanied by the NRC resident inspector

ANSWER
c.
REFERENCE
F2; Radiation Safety, revision 33
C10; Incore Instrumentation; revision 22
NEW
HIGHER
K/A: 2.3.13 Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.
EXPLANATION

a. Incorrect. In Mode 2, entries must be made by more than one person, and a qualified Radiation Protection Specialist should accompany the party. There is no requirement that the individual must be from the operations department. Individuals are allowed to enter alone in all other modes.
b. Incorrect. Though this may be desirable, no such requirement exists. The requirement is that no incore detector movement is in progress.
c. Correct. According to F2; Radiation Safety; entry with reactivity changes occurring is not allowed while critical, or during startup unless authorization is obtained from the RPM
d. Incorrect. There is no such requirement. Plausible if the applicant does not understand the facility's containment entry requirements.

Question #: 073

Given the following conditions:

- You are using the Two Step Monitor at Access Control following a tour of the Aux Bldg
- The Two Step Monitor alarms
- You exit and see indication of contamination on your right shoe
- You perform a personnel frisk using a hand held frisker
- No contamination is detected during performance of the frisk
- You leave Access Control and proceed with end of shift watch turnover

As you are exiting the site following watch relief, the Portal Monitor at the Security Access Facility displays a valid contamination alarm.

Which one of the following is the proper response to the alarm?

a. Immediately contact the Radiation Protection Group because of the earlier alarm at Access Control. Do not leave the Security Access Facility or the Site.

b. Immediately return directly to Access Control and use the Two Step Monitor to resolve the Portal Monitor alarm. If no alarm received, you may return and exit the Site.

c. Perform one additional Portal Monitor recount and if there is no valid contamination alarm received, you may exit the Security Access Facility and leave the Site.

d. Perform two additional Portal Monitor recounts and if there is no valid contamination alarm received, you may exit the Security Access Facility and leave the Site.

ANSWER:
d.
REFERENCE
F2, Radiation Safety, Revision 33
NEW
HIGHER
K/A: 2.3.5 Radiation Control: – Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc. (RO 2.9)
EXPLANATION:
a. Is incorrect. Plausible if candidate Portal Monitor alarm constitutes two of three valid alarms.
b. Is incorrect. Plausible if candidate believes that the Two Step Monitor is a more accurate device and necessary to use because of the earlier alarm.
c. Is incorrect. Plausible if the candidate believes only one recount is required because of the frisk performed at Access Control.
d. Is correct. This demonstrates an accurate use of procedures.

Question #: 074

From the following, choose the ENTRY CONDITION and IMMEDIATE ACTIONS for 1FR-S.1, "Response to Nuclear Power Generation/ATWS."

	Entry Condition	Immediate Actions
a.	SI has initiated while reactor power remains at 99%.	1) Verify reactor trip breakers OPEN. 2) Verify AFW Pumps are RUNNING.
b.	Reactor trip CANNOT be verified while performing step 1 of 1E-0, "Reactor Trip or Safety Injection".	1) Verify reactor trip breakers OPEN. 2) Verify turbine stop valves CLOSED.
c.	ANYTIME a RED or ORANGE condition exists on the "Subcriticality" status tree.	1) Verify reactor trip breakers OPEN. 2) Verify turbine stop valves CLOSED.
d.	Reactor trip CANNOT be verified while performing step 1 of 1E-0, "Reactor Trip or Safety Injection".	1) Verify reactor trip breakers OPEN. 2) Verify AFW Pumps are RUNNING.

ANSWER:
b.
REFERENCE
SWI O-10; Operations Manual Usage, revision 50
1FR-S.1; Response to Nuclear Power Generation/ATWS, revision 12
1E-0; Reactor Trip or Safety Injection, revision 28
NEW
FUNDAMENTAL
K/A: 2.4.1 Emergency Procedures / Plan: – Knowledge of EOP entry conditions and immediate action steps. (RO 4.6)
EXPLANATION:
SWI O-10, paragraph 7.8.4.e.5 states, in part that, “all operators should be aware of the condition of the critical safety functions.” Additionally SWI O-10 requires that operators memorize the immediate actions of FR-S.1.

a. Is incorrect. Verifying AFW pumps running is not an immediate action.
b. Is correct.
c. Is incorrect. The entry condition is a false statement. Entry into 1FR-S.1 is not warranted “anytime” a red or orange path exists.
d. Is incorrect. Plausible combination of distrators a and b.

Question #: 075

Per SWI-O-10, "Operations Manual Usage", which of the following correctly describes when an emergency procedure action on the Information page, is applicable?

a. PRIOR to performing the applicable step in the main body of the procedure.

b. ANY time during the applicable procedure performance, following the completion of any applicable immediate action steps.

c. AFTER proceeding PAST the applicable step in the main body of the procedure, AND it MAY apply after a transition is made to another procedure.

d. AFTER proceeding PAST the applicable step in the main body of the procedure, BUT it will NEVER apply after a transition is made to another procedure.

ANSWER:
b.
REFERENCE
SWI-O-10; Operations Manual Usage, revision 50
NEW
FUNDAMENTAL
K/A: 2.4.14 Emergency Procedures / Plan: – Knowledge of general guidelines for EOP usage. (RO 3.8)
EXPLANATION:
a. Is incorrect.
b. Is correct.
c. Is incorrect.
d. Is incorrect.

QUESTION # 076

Given the following conditions:

- Unit 1 is in Mode 1, 100% Power
- Over the past week, a rising trend in Unidentified Leakage has been noted following the performance of SP 1001AA, DAILY REACTOR COOLANT SYSTEM LEAKAGE TEST
- This shift, SP 1001AAA, REACTOR COOLANT LEAKAGE INVESTIGATION, has been initiated
- The Current leakage rate is .45 GPM
- Personnel have just exited the Unit 1 Containment, and informed you that a steam plume appears to be coming from the top of the Pressurizer at the spray line weld connection.
- System Engineering has informed you that all the leakage is being condensed by the FCUs, and is being collected in Containment Sump A
- No other equipment appears to be impacted

Currently…

a. RCS pressure boundary leakage exceeds limits

b. RCS leakage is within limits, continue to monitor to ensure the leakage does not interfere with RCS leakage detection systems

c. RCS unidentified leakage exceeds limits

d. RCS identified leakage not within limit for reasons other than pressure boundary leakage or primary to secondary leakage

ANSWER

a.

REFERENCE

PINGP Technical Specifications, revision 201

SP 1001AA, Daily Reactor Coolant System Leakage Test, revision 54

SP 1001AAA, Reactor Coolant Leakage Investigation, revision 14

NEW

HIGHER

K/A: 000008A2.2.42 – Ability to recognize system parameters that are entry-level conditions for Technical Specifications.

EXPLANATION

a. Correct. This meets the definition of pressure boundary leakage.
b. Incorrect. Plausible if the applicant is unable to recall the TS definitions, or believes that this meets the definition of identified leakage because the effluent is being collected in a sump
c. Incorrect. Plausible if the applicant is unable to recall the TS definition of leakage.
d. Incorrect. Plausible if the applicant believes that the leakage is identified leakage because it has been physically located, but exceeds the limit of the LCO.

QUESTION # 077

Given:

- One hour ago Unit 1 was operating at full power when a 450 GPM SGTR developed in the 12 SG
- RCPs were tripped while the crew was taking actions in 1E-0, REACTOR TRIP OR SAFETY INJECTION
- While completing the actions of 1E-3, STEAM GENERATOR TUBE RUPTURE, all RCP seal cooling was lost.

Currently, actions are being taken IAW 1ES-3.1, POST-SGTR COOLDOWN USING BACKFILL with the following conditions:

- All AC Busses are energized by offsite power
- Aux Spray is in the process of being aligned
- RCP seal cooling has been restored
- Containment pressure is 1.1 PSIG
- RVLIS full range indication reads 80%
- Pressurizer level is 60%
- CETCs read 437°F
- RCS Pressure is 550 PSIG
- The crew is about commence depressurization of the RCS to backfill
- The TSC has just reported that a status evaluation for the loss of RCP seal cooling been completed, and the 11 RCP may be started

Which one of the following actions(s) should the crew take?

a. Continue the cooldown to less than 200°F, do not start the 11 RCP since it will be secured later in the procedure

b. Turn on Pressurizer heaters, raise pressurizer level, and start the 11 RCP although the 12 RCP is preferred

c. Continue the cooldown to less than 200°F, and do not start the 11 RCP since the 12 RCP is preferred

d. Dump more steam, raise pressurizer level, and start the 11 RCP, although the 12 RCP is preferred

ANSWER
b.
REFERENCE
1E-3; Steam Generator Tube Rupture, revision 22
1ES-3.1; Post-SGTR Cooldown Using Backfill, revision 12
1C3 AOP1; Post Accident Emergency Start of a Reactor Coolant Pump, revision 5
ILT LP P8170L-001a; ICCM, revision 5
NEW
HIGHER
K/A: 000015/17K2.4.6 – Knowledge of EOP mitigation strategies
EXPLANATION

a. Incorrect. Plausible if the applicant recalls that RCPs are secured later in the procedure if #1 seal minimum requirements are no longer met.
b. Correct.
c. Incorrect. Plausible if the applicant believes that the 11 RCP will have little effect.
d. Incorrect. Plausible if the applicant fails to realize that the RCS has already cooled down greater that 100°F in the past hour, and believes this will satisfy the subcooling requirement faster.

Question #: 078

Given the following conditions:

- Unit 1 and 2 are operating at rated power.
- The 12 MD AFW pump is tagged out, inoperable, for troubleshooting activities.
- Unit 1 Reactor trip occurs.
- Unit 1 SS enters 1E-0, and the immediate actions are complete.
- The 11 TD AFW pump fails to start and cannot be started manually.
- The Unit 1 SS declares the 11 TD AFW pump inoperable.

Assuming that an SI did not actuate and is not required, which of the following 'First Out' annunciators would be indicative of a loss of normal feedwater on Unit 1 causing the Reactor trip, and is entry into LCO 3.0.3, and the plant required to be in MODE 4 within the next 13 hours?

a. RCP BUSES UNDERVOLTAGE REACTOR TRIP NO.

b. 11 STM GEN LO-LO LVL REACTOR TRIP: NO.

c. OVERPOWER ΔT REACTOR TRIP: YES.

d. PRZR LO PRESS REACTOR TRIP YES.

ANSWER:
b.
REFERENCE
Secondary Accidents Lesson Plan, P8161L-006
B46B, Station Annunciator System, Revision 3
C47017 Alarm Response Procedure
C47517 Alarm Response Procedure
NEW
HIGHER
K/A: 000054 A2.07 Loss of Main Feedwater / 4: – Ability to determine and interpret the following as they apply to the Loss of Main Feedwater (MFW): Reactor trip first-out panel indicator. (SO 3.9)
EXPLANATION:
a. Is incorrect.
b. Is correct.
c. Is incorrect.
d. Is incorrect.

Question #: 079

Given the following conditions:

- Unit 1 and 2 are operating at rated power.A LOOP occurs to both Units.All safety related equipment responds as designed.Assume no operator action.

What is the status of the voltage relays associated with Safeguards busses 16 and 26 Sequencer Cabinets exactly 75 seconds later?

a. All voltage relays associated with both busses are reset with the exception of the off-site (R and CT transformers) Source Not Available voltage relays which are still tripped.

b. All heaters energize and the spray valves go.

c. All heaters energize and the spray valves go.

d. All heaters energize and the spray valves go.

ANSWER:

a.

REFERENCE:

Non-Safeguards Distribution Lesson Plan P8186L-003

NEW

HIGHER

K/A: 000056 A2.53 Loss of Off-site Power / 6: – Ability to determine and interpret the following as they apply to the Loss of Offsite Power: Status of emergency bus under voltage relays. (SO 3.2)

EXPLANATION:

a. Is correct.
b. Is incorrect.
c. Is incorrect.
d. Is incorrect.

QUESTION # 080

Given the following sequence of events:

- At 0500 on July 9th, the 12 DDCLP was declared inoperable due to failing to start and load within one minute
- At 0900 on July 9th Bus 27 CANNOT aligned to Bus 25 due to a breaker issue

Today is July 14th:

- At 0600, the 21 Safeguards Screenhouse Exhaust Fan seized
- At 0800, the 21 CL Pump tripped, C35 AOP 2, LOSS OF PUMPING CAPACITY OR SUPPLY HEADER WITHOUT SI was implemented
- At 1200, the 12 DDCLP is repaired
- The time is currently 1400, and the Post Maintenance Test has just completed satisfactorily for the 12 DDCLP

Assuming all equipment has operated as designed, which one of the following statements below is correct?

(LCO 3.7.8 is attached)

a. LCO 3.7.8 is met and all conditions may exited at time 1400

b. Both CL trains have had Operable safeguards CL pumps since July 9th

c. Both Units are currently required to be in Mode 3

d. At 0100 today, both Units were required have be in Mode 5

ANSWER
c.
REFERENCE
C18.1; Engineered Safeguards Equipment Support Systems, revision 30
PINGP Technical Specifications, revision
PINGP Technical Specifications Bases, revision
ILT LP P8176L-003, Cooling Water System, revision 5
NEW
HIGHER
K/A: 000062 Loss of Nuclear Svc Water 2.2.42 – Ability to recognize system parameters that are entry-level conditions for Technical Specifications.
EXPLANATION

a. Incorrect. LCO 3.7.8 condition A for the Safeguards B train of CL applies. Plausible if the applicant fails to recognize that inoperability of a Safeguards Screenhouse Exhaust Fan causes inoperability of safeguards CL pump(s).
b. Incorrect. Plausible if the applicant lacks understanding of the TS bases.
c. Correct. Both units were in LCO 3.0.3 at 0600, therefore must be in Mode 3 by 1300
d. Incorrect. Plausible if the applicant lack understanding of the TS bases, and applies LCO 3.7.8 condition B for the 21 DDCLP.

Question #: 081

Given the following conditions:

- Unit 1 and 2 are operating at rated power.
- Unit 2 Reactor trip occurs.
- Unit 2 SS enters 2E-0, completes the immediate actions, and transitions to 2ES-0.1 Step 1.

Following completion of 2ES-0.1 Step 4, the Unit 2 Lead reports that:

- 21 & 22 SG wide range level indications are both 40% and lowering;
- 21 TDAFW pump has tripped & 22 MDAFW pump is running; and
- Indicated AFW flow to both 21 & 22 SGs is approximately zero.

What of the following is the correct procedure action required by the Unit 2 SS?

a. Remain in procedure 2ES-0.1 and continue with Steps 5 and 6 to restore total Feedwater flow to the SGs to greater than 200 GPM.

b. Concurrently enter 2C28.1 AOP4, Restarting Unit 2 AFW Pump After Low Suction/Discharge Pressure Trip.

c. Transition to functional recovery procedure 2FR-H.1, Response to Loss of Secondary Heat Sink, Step 1 due to RED condition.

d. Reenter procedure 2E-0 and continue with Steps 5 and 6 to restore total Feedwater flow to the SGs to greater than 200 GPM.

ANSWER:
c.
REFERENCE:
Secondary Accidents Lesson Plan, P8161L-006
Engineered Safeguards Systems Lesson Plan, P8180L-006
NEW
HIGHER
K/A: W/E05 A2.2 Inadequate Heat Transfer - Loss of Secondary Heat Sink / 4: – Ability to determine and interpret the following as they apply to the (Loss of Secondary Heat Sink): Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments. (SO 4.3)
EXPLANATION:
a. Is incorrect.
b. Is incorrect.
c. Is correct.
d. Is incorrect.

Question # 082

Given the following conditions:

- Unit 1 is in a refueling outage
- Core offload is in progress
- SRNI N-31 and N-32 read 3 and 5 CPS and are stable
- 11 RHR Pump is running maintaining RCS temperature 110°F
- ERCS Refueling water level has not changed for the past 8 hours
- A fuel assembly is in the fuel transfer cart
- 1R-12 has just alarmed
- 1R-22 levels are rising
- All other plant radiation monitors are normal
- The fuel handling crew in containment and the SFP area HAVE NOT reported any abnormalities

The refuel floor SRO should have the fuel handling crew . . .

a. continue moving the fuel assembly to the SFP area and implement C1.6, “Containment Evacuation.”

b. stop the fuel transfer cart, and implement C1.6, “Containment Evacuation” and D5.2 AOP 4, “Spent Fuel Pool Area Evacuation – Refueling.”

c. move the transfer cart back into containment, place the fuel assembly in a pre-designated safe location with the refueling machine and implement C1.6, “Containment Evacuation.”

d. move the transfer cart back into containment, implement C1.6, “Containment Evacuation” and direct closure of any opening logged in C19.9, “Containment Boundary Control During Mode 5, Cold Shutdown and Mode 6, Refueling

Answer
a.
References:
C1.6; Containment Evacuation, revision 9
D5.2 AOP 1; Damaged Fuel Assembly, revision 5
Radiation Monitor List included with LP P8182L-002 materials
NEW
HIGHER

K/A: 000060 Accidental Gaseous Radwaste Rel. 2.1.7 – Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.

Explanation: This question is SRO level because it requires knowledge of fuel handling procedures. 1R-12 and 1R-22 detect gaseous activity. Rising levels on only these monitors, suggests a gaseous release in containment.

a. Correct. Completing the transfer of the fuel cart places the fuel in a known safe location and does not introduce the possibility of complications. The unexpected rising gaseous radiation levels warrant containment evacuation.
b. Incorrect. Plausible misconception of placing the fuel in a safe location.
c. Incorrect. Plausible if the applicant has a misconception as to the time frame in which to complete actions, or what constitutes a safe location.
d. Incorrect. Plausible is the applicant has the misconception that reversing the transfer cart is preferred.

QUESTION # 083

Given the following conditions:

- You are the Unit 2 SS
- A fire has occurred in the G Panel in the Control Room
- The Unit 1 SS has determined that the Control Room is NOT habitable and implemented F5 Appendix B, CONTROL ROOM EVACUATION (FIRE)
- You have just reported to the HSPs and note the following indications on Train A HSP:

11 SG WR level	62%
21 SG WR level	40%
Unit 1 RCS T-hot	547°F
Unit 1 RCS T-cold	547°F
Unit 2 RCS T-hot	530°F
Unit 2 RCS T-cold	530°F
Unit 1 Pressurizer level	19%
Unit 2 Pressurizer level	0%

Which of the following resulted in the indications noted above?

a. Unit 2 has a faulted SG, implement 2EP-2 concurrently while performing F5 Appendix B actions

b. The Unit 2 Reactor Operator was unable to complete all of his/her actions prior to evacuating the CR, continue with F5 Appendix B actions

c. The Unit 1 Reactor Operator was unable to complete all of his/her actions after evacuating the CR, complete those actions missed and continue F5 Appendix B actions

d. The CR fire has damaged the cold calibrated pressurizer level instrumentation for Unit 2, continue with F5 Appendix B actions

ANSWER
b.
REFERENCE
F5 Appendix B; Control Room Evacuation, revision 45
2C1.3 AOP1; Shutdown from Outside the Control Room - Unit 2, revision 15
ILT LP P8170L-006; Pressurizer Level Control System, revision 4
ILT LP P8197L-009; F5 Appendix B/D Review, revision 5
NEW
HIGHER
K/A: 000068 (BW/A06) A2.07 – Ability to determine and interpret the following as they apply to the Control Room Evacuation: PZR level
EXPLANATION
Both Unit ROs are to trip their reactors and then turbines prior to evacuating the CR. The Unit 2 RO failed to trip the Unit 2 turbine, however the Unit 1 RO locally verifies both turbines are tripped when leaving the CR. With the Unit 2 reactor tripped, and the turbine not tripped, an SI would have occurred on Unit 2

a. Incorrect. Plausible if the applicant fails to note that there is still level in the Unit 2 SG, but no delta-t.
b. Correct. See EXPLANATION above.
c. Incorrect. If the Unit 2 RO completed his/her required actions, the Unit 1 RO's inaction would have produced the indications noted.
d. Incorrect.

Question #: 084

Given the following conditions:

- Unit 1 and 2 were operating at rated power.
- At 10:10 am, Unit 2 SI and Reactor trip occurred.
- Unit 2 SS entered 2E-0, REACTOR TRIP OR SAFETY INJECTION, completed the immediate actions, and continued through the procedure.
- At 10:25 am, Unit 2 SS transitioned to 2ECA-1.2, LOCA OUTSIDE CONTAINMENT, Step 1, due to High Radiation in the Aux Bldg.
- At 10:35 am, Unit 2 SS transitioned to 2ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION, due to inability to isolate the leak outside containment.
- At 10:50 am, the crew is at the step in 2ECA-1.1 to determine if SI should be terminated.

Current conditions are as follows:

- Containment pressure, temperature, radiation, and Sump B level are all normal.
- RCS pressure is 1100 psig.
- 21 and 22 RCPs are secured.
- 21 SI pump is running.
- 21 Charging pump is running in manual.
- 21 and 22 RHR pumps are secured.
- RVLIS Full Range indicates 73%.
- RCS Subcooling indicates 10°F.

Based upon the conditions given above, which one of the following correctly describes the minimum required injection flow rate, and which running pump should be secured?

a. 187 gpm; stop the 21 SI pump.

b. 200 gpm; stop the 21 SI pump.

c. 200 gpm; stop the 21 Charging pump.

d. 187 gpm; stop the 21 Charging pump.

ANSWER:
c.
REFERENCE:
2ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION, Revision 11
B12A, Chemical and Volume Control, Revision 11
B18A, Safety Injection System, Revision 10
NEW
HIGHER
K/A: 000074 2.1.25 (W/E06&E07) Inad. Core Cooling / 4: – Ability to interpret reference materials, such as graphs, curves, tables, etc. (SO 4.2)
Explanation: This question requires the candidate to assess plant conditions (normal, abnormal, or emergency) and then select a procedure or section of a procedure with which to proceed, and is therefore SRO-only.

a. Is incorrect. This is the wrong minimum injection flow value and wrong pump to secure.
b. Is incorrect. This is the correct minimum injection flow, but the incorrect pump to secure.
c. Is correct. This is correct per Figure ECA11-1, and the charging pump is not currently required by the procedures.
d. Is incorrect. This is the wrong minimum injection flow value with the correct pump to secure.

Question #: 085

Given the following conditions:

- Unit 1 and Unit 2 were operating at rated power.
- Unit 1 SI and Reactor Trip occurred due to a SBLOCA inside containment.
- Containment radiation level indicates 250 R/hr and rising slowly.
- Containment pressure is 4 psig and stable.
- You are the Unit 2 SS.

The Shift Manager directs you to investigate what should be done to reduce containment radiation levels and permit longer Aux Building stay times.

Which one of the following is the appropriate action for the Unit 1 SS to direct:

a. Place the Containment Spray System in service as directed by 1E-1, RESPONSE TO LOSS OF REACTOR OR SECONDARY COOLANT.

b. Place the Containment Cleanup System in service per 1C19.2, as directed by 1FR-Z.3, RESPONSE TO HIGH CONTAINMENT RADIATION.

c. Place the Containment Post-LOCA Ventilation System in service per C19.4, as directed by 1FR-Z.3, RESPONSE TO HIGH CONTAINMENT RADIATION.

d. Place all Containment Dome Recirculation fans in service as directed by step 16 of 1E-1, RESPONSE TO LOSS OF REACTOR OR SECONDARY COOLANT.

ANSWER:
b.
REFERENCE
Radiation Monitoring System Lesson Plan, P8182L-002
1FR-Z.3, Response to High Containment Radiation, Revision 4
1C19.2, Containment System Ventilation Unit 1, Revision 22
NEW
HIGHER
K/A: W/E16 A2.1 High Containment Radiation / 9: – Ability to determine and interpret the following as they apply to the (High Containment Radiation) Facility conditions and selection of appropriate procedures during abnormal and emergency operations. (SO 3.3)
EXPLANATION:
This question requires the candidate to assess plant conditions (normal, abnormal, or emergency) and then select a procedure or section of a procedure with which to proceed, and is therefore SRO-only.

a. Is incorrect. Plausible if the candidate does not understand that CS will not be effective for high radiation.
b. Is correct. Although a yellow path is not required to be performed yet, it may if all other CSF status trees are green, and this will reduce activity in containment.
c. Is incorrect. Not required to respond to high radiation in containment per the FR.
d. Is incorrect. Although required by 1E-1, this is to mitigate hydrogen gas buildup not high radiation.

QUESTION # 086

Given the following conditions:

- Unit 1 is has recently completed refueling
- RCS temperature is 120°F and stable
- All RCCAs are latched
- RCS draining is in progress using the 11 RHR Pump
- MV-32206; 11 RHR HX TO 11 SI PUMP, shows dual indication in the CR
- MV-32162; 11 SI PUMP SUCT, shows full open indication in the CR
- RWST level is 80% and rising at 850 GPM
- RP and Maintenance personnel are standing by for permission to set the reactor vessel head
- Refueling cavity water level has just dropped below the 4 inch reactor vessel "lip"

Before the reactor vessel head can be set, the Unit 1 SS should direct operators to . . .

a. close MV-32206 from the CR and secure the RHR drain lineup.

b. stop the 11 RHR pump when RCS level is 1 foot below the RV flange.

c. close MV-32206 locally, and standby for further draining.

d. close MV-32162 from the CR and secure the RHR drain lineup.

Answer
a.
References:
Figure C1-40; Refueling Water Levels, revision 3
ILT LP P8140L-208; Conduct RCS Fill and Vent Operations, revision 1
1C4.2; RCS Inventory Control - Post Refueling, revision 28
ILT LP P8180L-005; Emergency Core Cooling System, revision 3
ILT LP P8180L-003; Residual Heat Removal System, revision 7
NEW
HIGHER

K/A: 005 Residual Heat Removal 2.1.28 – Knowledge of the purpose and function of major system components and controls.

Explanation:
Though MV-32206 is initially opened locally, it must be closed from the CR to prevent rapid draining of the reactor vessel. Final draining of the reactor vessel to allow for setting the head is not accomplished by the RHR system. This question is SRO level because it requires knowledge of fuel handling procedures, specifically the sequence for vessel reassembly

a. Correct. See explanation above.
b. Incorrect. Stopping RHR pumps would cause the RWST to gravity train back to the vessel. Plausible if the applicant recognizes that the head can be set once vessel level is 1 foot below the flange.
c. Incorrect. Local closure is not desirable or procedurally directed since the RCS is draining at a rate that would rapidly drain the vessel. Plausible if the applicant recognizes that an interlock exists between MV-32206 and MV-32162, but has a misconception as to how that interlock functions.
d. Incorrect. Though this would stop draining of the RCS, this action may result in over pressurization of the SI pump suction piping, and is not directed by procedure. Plausible if the applicant recognizes that this action will stop the RCS draining.

QUESTION # 087

Given the following conditions for Unit 1:

- A Large Break LOCA has occurred
- RWST level is 7% and stable
- Switchover to recirculation phase was completed 1 hour ago
- 11 RHR pump is running
- 12 RHR pump tripped on overcurrent
- Containment pressure is 10 PSIG and lowering slowly
- Containment Sump B Wide Range level is 3 feet and rising slowly
- Containment Sump B Narrow Range level is 60% and lowering
- Long term plant status is being evaluated IAW 1E-1, "Loss of Reactor or Secondary Coolant", Step 27
- 11 RHR pump amps were rising over the past hour, and have begun to oscillate

What is occurring (1) AND what procedure steps should be implemented(2)?

a. (1) Lowering Containment pressure is allowing flashing to occur in the RHR pump suction lines,
   (2) transition to 1ECA-1.1, “Loss of Emergency Coolant Recirculation” to reduce recirculation flow

b. (1) Sump B strainers are beginning to clog with loose debris
   (2) transition to 1ECA-1.3, "Recirculation Sump Blockage" to secure pumps and/or reduce recirculation flow as needed

c. (1) 11 RHR Pump is cavitating due insufficient sump water level
   (2) ensure all actions of 1ES-1.3, “Transfer To Recirculation With One Safeguard Train Out of Service" have been performed

d. (1) Boiling is occurring in the RHR heat exchangers
   (2) complete the steps in 1ES-1.3, "Transfer To Recirculation With One Safeguard Train Out of Service " to align more CC flow to 11 RHR HX

Answer
b.
References:
1ES-1.2; Transfer to Recirculation, revision 21
1E-1; Loss of Reactor or Secondary Coolant, revision 23
1ECA-1.1; Loss of Emergency Coolant Recirculation, revision 11
1ECA-1.3; Recirculation Sump Blockage, revision 1
B18B; Emergency Core Cooling System, revision 10
1ES-1.3 Transfer to Recirculation With One Safeguard Train Out of Service, revision 16
NEW
HIGHER
K/A: 006 Emergency Core Cooling A2.05 – Ability to (a) predict the impacts of the following malfunctions or operations on the ECCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Improper amperage to the pump motor

Explanation: Fluctuating RHR pump amperage is indicative of recirculation sump blockage in this condition. This question is SRO level, because it requires the applicant to assess conditions and select appropriate procedures.

a. Incorrect. Plausible if the applicant is unaware of the entry conditions for 1ECA-1.3, or does not understand the mechanisms involved with sump screen blockage.
b. Correct.
c. Incorrect. Plausible if the applicant recognizes RHR cavitation is occurring, but fails to understand that all actions of 1ES-1.3 have been taken, and conditions now warrant direct entry in to 1ECA-1.3
d. Incorrect. Since containment has been on recirculation for 1 hour the CC alignment has been assumed to be adequate for the next 23 hours. Plausible if the applicant recognizes that CC alignment is eventually required, but misinterprets other indications available in the stem.

QUESTION # 088

Given the following conditions:

- Unit 1 is at 100% Power, Steady State and normal alignment
- A malfunction occurs, causing a complete loss of CC flow to the letdown HX
- No operator actions have been taken

Which of the following correctly describes the impact of the conditions above (1), AND select the procedure that will mitigate its effects?

a. (1) VCT level will lower to a new stable value.
   (2) Implement 1E-0, “Reactor Trip or Safety Injection”

b. (1) Pressurizer level will rise until the reactor trips.
   (2) Place Excess Letdown in service IAW 1C12.1; “Letdown, Charging, and Seal Water Injection – Unit 1”

c. (1) Charging Pumps will become gas bound and trip.
   (2) Enter 1C12.1 AOP 2, “Loss of Charging Flow to the Regen HX”

d. (1) Ion exchanger resins will not release contaminants from damage
   (2) Implement 1C14 AOP 1, “Loss of Component Cooling”

Answer:
d.
References:
1C14 AOP1; Loss of Component Cooling, revision 18
ILT LP P8172L-002; Component Cooling, revision 5
ILT LP P8172L-001A; CVCS, revision 5
ARP 47015-0408; “LTDN FLOW HI TEMP”, revision 35
1C14; Component Cooling System - Unit 1, revision 33
1C12.1 AOP4; Alternate Letdown Flowpaths, revision 1
1C12.1 AOP3; Loss of Letdown Flow to the VCT, revision 0
1C12.1; Letdown, Charging, and Seal Water Injection - Unit 1, revision 18
1C12.1 AOP2; Loss of Charging Flow to the Regen HX, revision 1
1C12.1 AOP1; Loss of RCP Seal Injection, revision 4
NEW
HIGHER

K/A: 008 Component Cooling Water A2.09 – Ability to (a) predict the impacts of the following malfunctions or operations on the CCWS, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Results of excessive exit temperature from the letdown cooler, including the temperature effects on ion-exchange resins

Explanation: Upon a loss of CC to the letdown HX, letdown temperature will rapidly rise above setpoint of 140°F for divert valve CV-31204, with will automatically direct all letdown flow to the VCT, protecting the ion exchangers from damage. This question is SRO level, because it requires knowledge of emergency and abnormal procedure content.

a. Incorrect. Plausible if the applicant concludes that RCP temperatures may reach a manual trip setpoint from a loss of CC, and incorrectly assumes that letdown is isolated.
b. Incorrect. Plausible if the applicant assumes letdown has been lost.
c. Incorrect. Plausible outcome, however, 1C12.1 AOP 1 would be the correct procedure in this case.
d. Correct. See explanation above. In this case letdown must be isolated, which is accomplished by 1C14 AOP 1, additionally clear thresholds for a reactor trip are established due to a CC malfunction.

Question #: 089

1ES-0.2, as well as many additional Emergency Operating Procedures in the control room contains the following CAUTION prior to the step to 'Reset SI.'

> *IF offsite power is lost after SI reset, THEN manual action may be required to restart safeguards equipment.*

(1) Which one of the following choices correctly describes the basis for this statement?

and

(2) Does the 'Manual Operator Action' maintain OPERABILITY of the safeguards equipment?

a. (1) The SI logic requires a manual operator action to remove the undervoltage signal to reset the circuitry.
(2) Yes.

b. (1) Manual action must be taken to realign all safeguards equipment, otherwise the equipment would not restart.
(2) No.

c. (1) SI must be reset to allow operating equipment to be shutdown, and automatic SI signals are not blocked after SI reset.
(2) Yes.

d. (1) Normal sequencing of safeguard equipment loading onto emergency buses after diesel generator startup will not occur.
(2) No.

ANSWER:
d.
REFERENCE:
Tech Spec LCO 3.3.2, Engineered Safety Feature Actuation System Instrumentation
Engineered Safeguards System Lesson Plan, P8180L-006
B18C, Engineered Safeguards System, Revision 8
MODIFIED – PBNP 031.00.LP0000.000 002
HIGHER
K/A: 013 2.1.27 Engineered Safety Features Actuation: – Knowledge of system purpose and/or function. (SO 4.0)
EXPLANATION:
This question requires the candidate to demonstrate knowledge of TS bases that are required to analyze TS required actions and terminology, and is therefore SRO-only.

a. Is incorrect. This is an inaccurate rational along with an incorrect OPERABILITY application.
b. Is incorrect. This is an inaccurate rational along with a correct OPERABILITY application.
c. Is incorrect. This is an inaccurate rational along with an incorrect OPERABILITY application.
d. Is correct. This is the bases for the caution, and correctly states that the ‘manual operator action’ is not maintaining OPERABILITY.

Question #: 090

Given the following conditions:

- Unit 1 was operating at rated power.
- Unit 1 SI and Reactor trip occurred.
- You are the Unit 1 SS.

You directed entry into 1E-0, REACTOR TRIP OR SAFETY INJECTION, completed the immediate actions, and continued through the procedure.
The following conditions were reported for Unit 1:

- RCS pressure 1350 psig and stable.
- Containment pressure 26 psig and rising slowly.
- RWST level indicates 64% and lowering.
- 11 and 12 RCPs were secured.
- 11 and 12 SI pumps running; 11 and 12 RHR pumps will not start.
- 13 Containment FCU 'SI NOT READY' light LIT and the Containment FCU does not indicate running in SLOW.

When you transitioned to 1E-1, LOSS OF REACTOR OR SECONDARY COOLANT, the following conditions were noted for Unit 1:

- 11 and 12 SG pressures 900 psig and stable.
- 11 and 12 SG levels 45% WR and rising slowly.
- RCS pressure 1250 psig and stable.
- PZR level 21% and rising slowly.

You transitioned to 1ECA1-1, LOSS OF EMERGENCY COOLANT RECIRCULATION.
The following conditions are currently noted for Unit 1:

- Containment pressure 20 psig and stable.
- RWST level indicates 35% and lowering.

Using the reference table from Step 5 of the procedure in effect, which one of the following is the correct combination of Containment FCUs and Containment Spray pumps required based upon the above conditions?

a. 0 Containment FCUs and 0 Containment Spray pumps.

b. 1 Containment FCUs and 1 Containment Spray pumps.

c. 3 Containment FCUs and 1 Containment Spray pumps.

d. 3 Containment FCUs and 0 Containment Spray pumps.

ANSWER:
c.
REFERENCE:
E-1 & E-2 Procedure Review Lesson Plan, P8197L-012
1ECA1-1, Loss of Emergency Coolant Recirculation, Revision 11
B18C, Engineered Safeguards System, Revision 8
NEW
HIGHER
K/A: 103 2.1.25 Containment: – Ability to interpret reference materials, such as graphs, curves, tables, etc. (SO 4.2)
EXPLANATION:
This question requires the candidate to assess plant conditions and then select a procedure or section of a procedure with which to proceed, and is therefore SRO-only.

a. Is incorrect.
b. Is incorrect.
c. Is correct. With RWST level at 35%, Containment pressure at 20 psig, and only 3 CFCUs available, 1 Containment Spray pump is still required.
d. Is incorrect.

Proposed references to be provided to applicants during examination:
1ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION, Revision 11 - through Step 5

Question #: 091

Given the following conditions:

- You are the Unit 1 SS.
- Unit 1 was operating at rated power.
- Annunciator C47017-0402, PRZR LO PRESS and 0502, CONTAINMENT HI PRESS SI alarmed.
- The turbine tripped, control rods were stepping in, but the reactor failed to trip.
- The Unit 1 RO (OATC) announced the turbine trip, failure of the reactor to trip, and inserted a Manual reactor trip; the reactor trip breakers did NOT OPEN.
- Reactor power was >5% on all channels.
- You directed entry into 1E-0, REACTOR TRIP OR SAFETY INJECTION, and immediately transition to 1FR-S.1, RESPONSE TO NUCLEAR POWER GENERATION/ATWS, Step 1.

When you reach the step to “Check Core Exit T/Cs,” the following plant conditions exist:

- The reactor trip breakers are being opened locally.
- Control rods are NOT Fully Inserted.
- The turbine is tripped.
- Emergency boration has been established.
- All power range channels read 5% reactor power.
- Both intermediate range (NFM) channels read 0.0 DPM.
- PZR level is off scale low.
- RCS pressure is 300 psig.
- Core exit thermocouples indicate 1350°F and lowering slowly.

Based upon the conditions given above, which one of the following correctly describes your next action and the reason why?

a. Transition to 1SACRG-1, SEVERE ACCIDENT CONTROL ROOM GUIDELINE INITIAL RESPONSE, Step 1; attempts to restore core cooling have failed, core damage cannot be prevented and the operators should go to the SAMGs.

b. Transition to 1FR-C.1, RESPONSE TO INADEQUATE CORE COOLING, Step 1; core exit thermocouples reading >1200°F indicate an extreme challenge to the fuel clad/matrix barrier and requires immediate operator action.

c. Remain within 1FR-S.1, RESPONSE TO NUCLEAR POWER GENERATION/ ATWS; continue to borate the RCS, check for other sources of positive reactivity, and perform actions of other FR procedures in effect which do not cooldown.

d. Return to Procedure and Step In Effect, and transition to 1FR-C.1, RESPONSE TO INADEQUATE CORE COOLING, Step 1; SI has not been successful in cooling the core, and the operators must perform alternative actions for establishing core cooling.

ANSWER:
c.

REFERENCE:
Inadequate Core Cooling Monitor Lesson Plan, P8170L-001a
B10, Incore Instrumentation System, Revision 5
1FR-S.1, Response to Nuclear Power Generation/ATWS, Revision 12
NEW
HIGHER
K/A: 017 A2.02 In-core Temperature Monitor: – Ability to (a) predict the impacts of the following malfunctions or operations on the ITM system; and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations: Core Damage. (SO 4.1)
EXPLANATION:
This question requires the candidate to demonstrate knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures, and is therefore SRO-only

a. Is incorrect. Transition to the SAMGs is only directed if the CETCs readings are still increasing, which is tangible proof that core damage has resulted, and currently reading are lowering.
b. Is incorrect. Transition to any other FR procedure is not allowed, but if candidate understands that core damage may not have occurred, then this next priority FR may seem prudent.
c. Is correct. Remaining within this FR procedure is correct until positive indication of subcriticality is obtained.
d. Is incorrect. This action is only allowed when the conditions necessary to satisfy the CSF have been re-established, although the candidate may believe that reactor power and SUR indication are adequate to return to the procedure and step in effect to address other post-LOCA conditions.

QUESTION # 092

What would be the implications of Unit 1 operating in Mode 2 without the 18 inch blind flanges of the In-service Purge system installed?

a. Unit 1 would enter LCO 3.0.3 for both trains of SBVS being INOPERABLE

b. A LOCA would always result in declaration of a SAE for 2 failed fission product barriers

c. None, as long as overall containment leakage was verified less than 1.0 L_a

d. PINGP may not satisfy the conditions of the Unit 1 Operating license

Answer:
d.
References:
ILT LP P8180L-009E; Containment Purge & In-Service Purge Ventilation System, revision 3
PINGP 1576; EAL Matrix, revision 6
PINGP Technical Specification Bases, revision 211
PINGP Technical Specifications, revision 201
NEW
HIGHER
K/A: 029 Containment Purge 2.2.38 – Knowledge of the conditions and limitations in the facility license.

Explanation: This question is SRO level because it requires knowledge of TS Bases.

a. Incorrect. SBVS would not necessarily be inoperable. Plausible because it may be conceived that the SBVS will unable to maintain D/P.
b. Incorrect. Plausible if it is believed that containment integrity is not intact as a result of the stem.
c. Incorrect. Plausible misinterpretation of TS 3.6.3 Bases. Also there is no specific condition spelled out for this inoperability. Competent inference by an SRO is required to arrive at the correct condition. A complete proper analysis, closed book, to determine the correct specific condition is not desirable, lending credibility to this choice.
d. Correct. PINGP Tech Specs require these flanges to be installed prior to entry into Mode 4 via SR 3.6.3.2, therefore TS 3.6.3 is not satisfied, and Unit 1 is not in compliance with the conditions of the facility's operating license

Question #: 093

Given the following conditions:

- Unit 1 is in a refueling outage.
- You are the Containment SRO.
- Core reload refueling activities are in progress.
- An irradiated fuel assembly is being lowered into the core with the HOIST JOG SWITCH.
- The ENTERING CORE SLOW ZONE light has just extinguished.
- The manipulator crane operator continues lowering the irradiated fuel assembly into the core, now using the HOIST CONTROL LEVER.

Approximately 2 minutes later, the manipulator crane operator reports the following conditions:

- The hoist abruptly stopped, the mast support tube shaking noticeably.
- The INTERMEDIATE CORE ZONE light is illuminated.
- The ENTERING CORE SLOW ZONE light and BOTTOM CORE SLOW ZONE light are both extinguished.
- The UNDERLOAD light and SLACK CABLE light are both illuminated.
- Gas bubbles are visible rising from the vicinity of the fuel assembly.

Which of the following manipulator crane hoist safety feature/interlock has failed, and what is your required action based upon the reported conditions?

a. The UNDERLOAD condition control signal which prohibits downward motion if excessive resistance is encountered. Implement D5.2 AOP1, Damaged Fuel Assembly and C1.6 AOP1, Containment Evacuation.

b. The BOTTOM CORE ZONE control signal which disables the HOIST CONTROL LEVER failed to actuate. Implement D5.2 AOP1, Damaged Fuel Assembly and C1.6 AOP1, Containment Evacuation.

c. The BOTTOM CORE ZONE control signal which disables the HOIST CONTROL LEVER failed to actuate. Implement D5.2 AOP4, Spent Fuel Pool Area Evacuation-Refueling and C1.6 AOP1, Containment Evacuation.

d. The UNDERLOAD condition control signal which prohibits downward motion if excessive resistance is encountered. Implement D5.2 AOP1, Damaged Fuel Assembly and D5.2 AOP4, Spent Fuel Pool Area Evacuation-Refueling.

ANSWER:
b.
REFERENCE:
Fuel Handling System Lesson Plan, P8182L-00-3
B17, Fuel Handling System, Revision 7
C17, Fuel Handling System, Revision 43
SWI O-41, Duties & Responsibilities of Fuel Handling Personnel, Revision 13
D5.2, Reactor Refueling Operations, Revision 54
NEW
HIGHER
K/A: 034 K4.01 Fuel Handling Equipment: – Knowledge of design feature(s) and/or interlock(s) which provide for the following: Fuel protection from binding and dropping. (SO 2.6)

EXPLANATION: This question requires the candidate to assess plant conditions and then select a procedure or section of a procedure with which to proceed, and is therefore SRO-only.

a. Is incorrect. This is the wrong control signal failure, since the underload light is illuminated, even though the procedures are correct.
b. Is correct. This is the correct control signal failure and the correct procedures to be implemented.
c. Is incorrect. Although this is the correct control signal failure, the implemented procedures are not correct.
d. Is incorrect. This has both the wrong control signal failure and incorrect procedures to implement.

Question #: 094

Given the following conditions:

- You are the Unit 1 SS.
- Unit 1 was operating at rated power, MOL.
- Control Rod J-4 (CBC) ROD BOTTOM light energized, and the ROD AT BOTTOM annunciator goes into alarm, but the reactor failed to trip on High Negative Flux Rate.
- The Unit 1 RO (OATC) announced the failure for the reactor to trip and inserted a Manual Reactor trip; the Reactor trip breakers did NOT OPEN.
- Reactor power was >5% on all channels.
- You directed entry into 1E-0, REACTOR TRIP OR SAFETY INJECTION, and immediately transition to 1FR-S.1, RESPONSE TO NUCLEAR POWER GENERATION/ATWS, Step 1.

When you reach the step to 'Verify the Reactor Subcritical,' the following plant conditions exist:

- The reactor trip breakers are NOT OPEN.
- Control rods are NOT Fully Inserted.
- Normal and emergency boration CANNOT be established, including aligning the RWST to charging, due to flow path blockages.
- The turbine is tripped.
- All power range channels read 3% to 4% power.
- Both intermediate range (NFM) channels read +0.1 DPM
- CETC indicate <700°F.

Based upon the condition given above, which one of the following correctly describes the action you will take next and the reason why?

a. Allow the RCS to heat up while continuing with efforts to establish normal or emergency boration. The RCS heat up will insert negative reactivity.

b. Maintain stable RCS temperature and return to the procedure and step in effect. Stable temperatures preclude positive reactivity insertion by dilution.

c. Exit 1FR-S.1 and return to the procedure and step in effect. Power is less than 5%, which is the design power level for auxiliary feedwater heat removal capability.

d. Exit 1FR-S.1 and transition to 1FR-S.2, RESPONSE TO LOSS OF CORE SHUTDOWN, since power channels are less than 5% with an intermediate range startup rate that is NOT more negative than -0.2 DPM.

ANSWER:

a.

REFERENCE:

B5, Rod Control System, Revision 5

B6, Rod Position Indication System, Revision 10

1F-0.1, Subcriticality Status Tree, Revision 2

MODIFIED – PBNP 043.03.LP1996.013 001

HIGHER

K/A: 2.1.43 Conduct of Operations: – Ability to use procedures to determine the effects on reactivity of plant changes, such as reactor coolant system temperature, secondary plant, fuel depletion, etc. (SO 4.3)

EXPLANATION:

This question requires the candidate to demonstrate knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures, and is therefore SRO-only.

a. Is correct. This action is procedurally defined, as is the reason for the action.
b. Is incorrect. Plausible based upon the inability to borate, but the positive action to allow the RCS to heat up is more stable
c. Is incorrect. This action does not take positive action to continue to establish a method of borating.
d. Is incorrect. If not complete with the FR procedure, you should not transition to a lower priority or severity.

Question #: 095

Given the following conditions:

- You are the Unit 1 SS.
- The Unit 2 SS is at the RCA access exit, following completion of an AUX BLDG tour.
- The Shift Manager is in the PI Training Center participating in an NRC exit meeting.
- Unit 1 and 2 are operating at rated power.
- The control room receives notification from the NRC Headquarter's Operation Center of an IMMINENT Airborne Threat, using the CORRECT Authentication Code.
- You have directed entry into abnormal procedure AB-8, Response to Security Threats, and have completed Step 1, Imminent Airborne Threat-IN PROGRESS, through Step 4, Initiate Plant Evacuation.
- YOU and the OATC ROs are NOW the ONLY personnel remaining in the control room.

What are the minimum control room staffing requirements for the current plant conditions, and what action is required next?

	Minimum Control Room Staffing	**Required Action**
a.	2 SROs and 3 ROs	Invoke 10 CFR 50.54(x) and Continue with AB-8
b.	2 SROs and 2 ROs	Have the Unit 2 SS go to the Control Room and Continue with AB-8
c.	1 SRO and 2 ROs	Continue with AB-8 – Immediately Trip Both Units and enter EOPs
d.	1 SRO and 1 ROs	Continue with AB-8 - Immediately Trip Both Units and enter EOPs

ANSWER:
c.
REFERENCE
SWI O-2, Shift Organization, Operation, and Turnover, Revision 73
AB-8, Response to Security Threats, Revision 20
NEW
HIGHER
K/A: 2.1.5 Conduct of Operations: – Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc. (SO 3.9)
EXPLANATION: This question requires the candidate to assess plant conditions and then select a procedure or section of a procedure with which to proceed, and is therefore SRO-only.

a. Is incorrect. This is the Tech Spec required manning for onsite, not the control room and therefore invoking 10 CFR 50.54(x) is not currently required.
b. Is incorrect. If candidate believes that 2 SROs are required with both Units in Mode 1, this action would correct the situation.
c. Is correct. This is the minimum manning for the control room the next action is to trip the units per step 5 of AB-8.
d. Is incorrect. If the candidate believes the Security abnormal procedure allows a reduced control room manning, this action would be correct

QUESTION # 096

Tech Spec 3.4.6 prohibits starting an RCP if any RCS cold leg temperature is less than or equal to the Over Pressure Protection System (OPPS) enable temperature specified in the PTLR when any S/G secondary water temperature is greater than or equal to 50°F above each of the RCS cold leg temperatures.

What is the basis for this restriction?

a. Prevent outsurge from emptying the pressurizer following an RCP start

b. Minimize RCS pressure transient caused by reverse heat transfer from a hot SG

c. Minimize RCS pressure transient caused by additional heat transfer from the core

d. Minimize RCS pressure transient due to additional RCP pressure head added to RCS pressure

Answer

b.

References:

PINGP Technical Specification Bases, revision 211

PINGP Technical Specifications, revision 201

NEW

FUNDAMENTAL

K/A 2.2.25 Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.

Explanation: The question is SRO level because it requires knowledge of Tech Spec bases. LCO 3.4.6 bases places restriction on starting an RCP to prevent a low temperature overpressure event due to the thermal transient when an RCP is started. The heat added by the SGs will expand the RCS, causing a pressure rise.

a. Incorrect. Higher SG temperature will cause an insurge to the Pzr, due to expansion of the coolant.
b. Correct, see explanation above.
c. Incorrect. The RCS pressure transient will not be caused by heat transfer out of the core. It will be into the core.
d. Incorrect. RCP pressure head while increased will not cause a large pressure transient. The RCP head pressure will only be felt on the U-Tubes of the S/G.

QUESTION # 097

Given the following conditions:

- You are the Unit 1 SS
- A Unit 1 reactor startup is in progress following an inadvertent plant trip
- The crew is performing steps of 1C1.2, "Unit 1 Startup Procedure"
- Control Bank ‘A’ is being withdrawn
- The last two 1/M plots indicate that criticality will be achieved on Control Bank "B" at approximately 100 steps
- The calculated ECC is CBD at 180 steps

Which of the following actions is required for these conditions (1), and (2) the appropriate procedure that will accomplish those actions?

a. (1) Manually trip the reactor
   (2) 1E-0, “Reactor Trip or Safety Injection”

b. (1) Manually insert ALL Control and Shutdown Bank rods
   (2) 1C5, “Control Rod and Rod Position Indication Systems”

c. (1) Manually insert the Control Banks to zero steps and verify minimum boron concentration
   (2) C1B, “Appendix - Reactor Startup”

d. (1) Hold the startup, and determine if criticality will be within 750 pcm of the ECC prior to proceeding
   (2) C1A, “Reactivity Calculations”

Answer:
c.
References:
1C5; Control Rod and Rod Position Indication Systems, revision 16
C1B; Appendix - Reactor Startup, revision 18
C1A; Reactivity Calculations, revision 24
NEW
HIGHER
K/A: 2.2.1 Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity.
Explanation: C1B provides direction if 1/M plots suggest that criticality will occur below the RIL to insert control banks and

a. Incorrect. Plausible since there is a significant unexpected response.
b. Incorrect. Plausible misconception of the direction contained in C1B
c. Correct.
d. Incorrect. Plausible if there is a misconception regarding the relationship between the 750 pcm threshold and the RIL.

QUESTION # 098

Which of the one of the following choices below describes the radiation monitor that is required to be operable in the event of a fuel handling accident, AND gives the correct basis for the requirement?

a. R-25, to prevent offsite dose from exceeding the limits of 10CFR100

b. R-31, to prevent the CR operator dose from exceeding the limits of 10CFR50.67

c. R-5, to alert operators to evacuation of the SFP area is warranted

d. R-28, to ensure that a single train of SFPSVS is capable of filtering SFP enclosure air before it is vented to the AB ventilation system

ANSWER
a.
REFERENCE
ARP 47048 1R-30, revision 35
ARP 47022-0108, revision 37
ARP 47022-0208, revision 46
ILT LP P8182L-002; Radiation Monitoring System, revision 8
B11; RADIATION MONITORING SYSTEM, revision 10
NEW
FUNDAMENTAL
K/A: 2.3.15 Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.
EXPLANATION
This QUESTION is SRO level because it requires knowledge of tech specs and their bases.

a. Correct. LCO 3.3.7 requires either R-25 or R-31, SFP ventilation radiation monitors, to be operable during the movement of irradiated fuel to prevent exceeding the dose limits of 10CFR100.
b. Incorrect. R-31 is a correct monitor, but TS Bases does not credit this monitor in limiting CR operator dose. Plausible if the applicant understands R-31 is required, but does not understand the basis for the requirement.
c. Incorrect. R-5, SFP area monitor, provides no safety function credited by the facility's Tech Specs, though it does provide indication that an accident has occurred. Plausible since the R-5 is one of 7 radiation monitors required to move fuel in the SFP.
d. Incorrect. R-28, New fuel pool area criticality monitor, also provides no safety function credited by the facility's Tech Specs, but provides indication that an accident has occurred. Plausible since the R-28 is also one of 7 radiation monitors required to move fuel in the SFP.

QUESTION # 099

Given the following conditions:

- Today is Sunday
- The Shift Manager became incapacitated and was transported offsite 30 minutes ago
- 15 Minutes ago a LOCA has occurred on Unit 1
- The Plant Manager is the duty Emergency Director, but is not onsite yet
- An Non-Licensed Operator (NLO) has sustained life-threatening injuries in a high dose area while attempting to isolate the uncontrolled release

Of the following choices, an operator performing a rescue can receive a maximum dose of _____ Rem TEDE, and this action must be authorized by _____.

a. 5, Unit 1 SS

b. 25, Unit 1 SS

c. 25, Unit 2 SS

d. 50, Unit 2 SS

Answer d.

References: ILT LP P7410L-035; Evacuation, Accountability, and Search & Rescue, revision 1
F3-1; Onsite Emergency Organization, revision 25
F3-12; Emergency Exposure Control, revision 19

K/A: 2.3.4 Knowledge of radiation exposure limits under normal or emergency conditions

NEW

HIGHER

Explanation: F3-1 states that the line of succession for the ED in this case is the non-affected unit SS. F3-12, states that ED permission must be obtained to exceed 10CFR20 limits (5 Rem TEDE). Exposures should be limited to 25 Rem, if practical, but can exceed that threshold if additional requirements are followed. Choices a, b, and c are combinations plausible misconceptions regarding these requirements.

a. Incorrect.
b. Incorrect.
c. Incorrect.
d. Correct. See explanation above

Question #: 100

Given the following conditions:

- You were the Unit 2 SS.
- Unit 2 was operating at rated power.
- Unit 2 SI and Reactor trip occurred.
- The RO acknowledges the audible alarm and commences 2E-0 Immediate Actions.

During performance of the 2E-0 Immediate Actions, the RO notices that the 22 SI pump and 22 RHR pump did not automatically start.

Per SWI O-10 and 2E-0, when are the 22 SI pump and 22 RHR pump required to be manually started?

a. Immediately upon discovery.

b. After 2E-0 Immediate Action have been verified.

c. When procedurally directed by 2E-0 Attachment L.

d. These pumps are not required to be operated since Train A pumps are running.

ANSWER:
c.
REFERENCE:
SWI O-10, Operations Manual Usage, Revision 50
FP-OP-COO-01, Conduct of Operations, Revision 10
2E-0, Reactor Trip or Safety Injection, Revision 27
NEW
FUNDAMENTAL
K/A: 2.4.5 Emergency Procedures / Plan: – Knowledge of the organization of the operating procedures network for normal, abnormal, and emergency evolutions. (SO 4.3)
EXPLANATION: This question requires the candidate to demonstrate the knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps, and is therefore SRO-only.

a. Is incorrect. The operators must perform the immediate actions of 2E-0 (including necessary RNO actions) prior to any other equipment manipulations.
b. Is incorrect. The immediate actions of 2E-0 only verify if an SI signal has actuated, and if not the RNO directs a manual SI actuation if one is required; it does not address individual SI components.
c. Is correct. Step 1 of Attachment L checks to verify automatic actions have occurred and performs component alignment as required via the RNO actions.
d. Is incorrect. Although this is a true statement for accident analysis, if the pumps will operate, Attachment L Step 1 RNO actions require starting the pumps.