Given the following conditions:

- Unit 1 is operating at 100% power.
- 1E 460 Volt bus is deenergized following a trip of its feed breaker.
- Tagging is in progress to allow troubleshooting of 1E 460 Volt bus.

The 1G 460 Volt bus feed breaker is opened instead of the 1E 460 Volt bus feed breaker, deenergizing the 1G 460 Volt bus.

Which of the following describes how this will affect Control Rods?

a. ALL control rods will drop into the core due to the loss of the only operating Rod Drive Motor Generator (RDMG) set.

b. ALL control rods will drop into the core due to the loss of one of the two operating RDMG sets.

c. NO control rods will drop into the core since both RDMG sets remains running.

d. NO control rods will drop into the core since one RDMG set remains running.

Answer	a	**Exam Level**	R	**Cognitive Level**	Application	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Emergency and Abnormal Plant Evolutions	**RO Group**	2	**SRO Group**	2				000003K205
003	Dropped Control Rod							**Record Number**	1
AK2.	Knowledge of the interrelations between Dropped Control Rod and the following:								
AK2.05	Control rod drive power supplies and logic circuits							2.5	2.8

Explanation of Answer

55.41.b(6) Unit one Rod Drive Motor Generator sets are powered from 2 (1E Bkr 15X and 1F Bkr 15X) of the 4 non vital 460V buses (1E, 1F, 1G, 1H). Normally, both RDMG sets are operating in parallell for redundancy in supplying power through the Reactor Trip Breakers to the control rods. With 1E 460 Volt bus deenergized from the initial conditions in the stem, removing power from the 1G 460 Volt bus will cause the only operating RDMG set to lose power. The loss of power to the control rods will cause the stationary gripper coils to un-grip, and ALL control rods will drop into the core. B is incorrect because no RDMG sets will be running, and even if one remained running no rods would drop. C is incorrect because both RDMG sets do not have power. D is incorrect because no RDMG sets are running. The modification to this question was required since there is no longer a negative flux rate Reactor trip, and the question originally asked whether the reactor would trip or not. Modified to ask about what happens to control rods.

Reference Title

Unit 1 1E 460V One Line

Unit 1 1G 460V One Line

Rod Control System Operation

Learning Objectives

RODS00E006

NCT Describe the function of the following components and how their normal and abnormal operation affects the Rod Control and Position Indication Systems:
Rod Cluster Control Assembly (RCCA)
Control Rod Drive Mechanism (CRDM)
Rod Drive MG Sets
Reactor Trip and Trip Bypass breakers
Reactor Control Unit
Power Cabinets
Logic Cabinet components:
Pulser
Master Cycler
Slave Cyclers
Bank Overlap Unit
h. DC Hold Cabinet
i. Rod Position Indicator (RPI) Coils
j. Signal Conditioning Modules
k. PulseûtoûAnalog (P to A) Converters

	l. Rod Bottom Bistables m. Rod Insertion Limit Comparator n. Step Counters
RODS00E007	NCT State the power supplies to the following Rod Control and Position Indication systems components: Rod Drive MG Sets Reactor Trip and Trip Bypass breakers Power Cabinets Logic Cabinet DC Hold Cabinet

Material Required for Examination	

Question Source:	Facility Exam Bank	**Question Modification Method:**	Editorially Modified

Question Source Comments:	Vision Q46032

Which of the following identifies the MINIMUM required AFW flow following a Rx trip, and the bases for that amount?

a. 22E4 lbm/hr. Ensures SG tubes are covered to prevent SG pressure drop and excessive tube D/P.

b. 44E4 lbm/hr. Ensures SG tubes are covered to prevent SG pressure drop and excessive tube D/P.

c. 22E4 lbm/hr. Provides sufficient flow for decay heat removal plus allowances for normal channel accuracy.

d. 44E4 lbm/hr. Provides sufficient flow for decay heat removal plus allowances for normal channel accuracy.

Answer	c	**Exam Level**	R	**Cognitive Level**	Memory	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Emergency and Abnormal Plant Evolutions	**RO Group**	1	**SRO Group**	1				000007K106
007	Reactor Trip							**Record Number**	2

EK1.	Knowledge of the operational implications of the following concepts as they apply to Reactor Trip:		
EK1.06	Relationship of emergency feedwater flow to S/G and decay heat removal following reactor trip	3.7	4.1

Explanation of Answer

55.41.b(4) At step 3 of TRIP-2, Reactor Trip Response, operators are asked if total AFW flow is >22E4 lbm/hr. The bases document states that this amount of flow is..."the minimum safeguards AFW flow requirement for heat removal plus allowances for normal channel accuracy (typically one AFW pump capacity at design pressure.)" The 44E4 lbm/hr choices are plausible because it is the minimum required AFW flow in FRSM-1. The SG pressure drop bases is found in SGTR series procedures to prevent a rapid pressure drop in a ruptured sG which would reinitiate primary to secondary leakage.

Reference Title

Reactor Trip Response

Learning Objectives

TRP002E006	Describe the basis for each step, caution, note, and continuous action summary item in 2-EOP-TRIP-2

Material Required for Examination

Question Source:	New	**Question Modification Method:**	

Question Source Comments:

Given the following conditions:

- Unit 2 is operating at 100% power when a catastrophic failure of RCS loop 21 cold leg piping occurs.
- RCS pressure rapidly dropped to 35 psig.
- 22 RHR pump failed to start, and remains stopped.
- Initial RWST level was 41.1 feet.

Of the following, which is CLOSEST to the the time available from the failure until the swap to Cold Leg recirc will be required?

a. 13 minutes.

b. 19 minutes.

c. 24 minutes.

d. 33 minutes.

Answer	b	**Exam Level**	R	**Cognitive Level**	Application	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Emergency and Abnormal Plant Evolutions	**RO Group**	1	**SRO Group**	1				000011K202
011	Large Break LOCA							**Record Number**	3

EK2.	Knowledge of the interrelations between Large Break LOCA and the following:		
EK2.02	Pumps	2.6*	2.7*

Explanation of Answer

55.41.b(8) Question stem describes LBLOCA with power. With the RCS at 35 psig, all ECCS pumps will be injecting at their maximum rate. The flow rates used are: Charging pumps 2x560= 1120gpm(page 23); SI pumps 2x675= 1350gpm (page 26); RHR 1x4500= 4500 (page 34); and Containment Spray pump flow of 2x2600=5200. (page 17) So, 1120+1350+4500+5200= 12,170 gpm total. With the initial RWST level of 41.1' equating to 370,000 gallons, and 15.2' level of 150,000, you need to pump in 220,000 gallons. That's 18.08 minutes. Distracter A is if the failed RHR pump is included. Distracter D is the time it would take to pump in the entire RWST volume. Distracter C is the time if CS pump flow is not included.

Reference Title

Tank Capacity Data

ECCS Lesson Plan

Containment Spray Lesson Plan

Learning Objectives

LOCA01E003

For the analyzed transients/accidents:

A. Inadvertent depressurization of the RCS - Stuck open safety (Condition II - Faults of Moderate Frequency)
B. Small Break LOCA - 4" diameter cold leg break (Condition III - Infrequent Fault)
C. Large Break LOCA - Double-ended break of an RCS cold leg (Condition IV 0 Limiting Fault)
   1. Describe the analysis assumptions
   2. Describe the protective features that mitigate the event
   3. Describe the expected plant response
   4. State whether the analysis indicates fuel damage and, if so, describe the expected fuel failure mechanism

Material Required for Examination: RO 3 S2.OP-TM.ZZ-0002, Tank Capacity Data, Page 28 of 34, RWST Tank Capacity

Question Source: Facility Exam Bank | **Question Modification Method:** Editorially Modified

Question Source Comments: Vision Q48704. Modified correct answer from 18 to 19 minutes to reduce probablility of candidate not choosing correct answer because it was LESS than calculated time. Also modified stem to say what is the CLOSEST time.

Given the following conditions:

- Unit 2 is operating at 30% power, steady state.
- OHA D-29, 22 RCP BKR OPEN/FLO LO is received.
- All 22 loop RC flows are 85% and dropping.
- The red START bezel for 22 RCP is illuminated.
- The reactor has NOT tripped.

Which of the following identifies what has occurred?

a. An ATWT.

b. 22 RCP shaft has seized.

c. 22 RCP shaft has sheared.

d. The 22RC9, RC FLOW common low press tap isolation valve has developed a leak.

Answer	c	**Exam Level**	R	**Cognitive Level**	Comprehension	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Emergency and Abnormal Plant Evolutions			**RO Group**	1	**SRO Group**	1		000015K210
015	Reactor Coolant Pump Malfunctions							**Record Number**	4
AK2.	Knowledge of the interrelations between Reactor Coolant Pump Malfunctions and the following:								
AK2.10	RCP indicators and controls							2.8*	2.8

Explanation of Answer

55.41.b(3,7) With a RCP shaft shear, there is no event that would cause the RCP breaker to open. For this reason, that is why the START bezel will still be illuminated, even though loop flows are all dropping. Distracter is incorrect because between 10%(P-10) and 36%(P-8), 1/4 RCS loop lo flow will NOT cause a Rx trip, the coincidence is 2/4. Distracter is incorrect because there are 3 low pressure flow taps, and 1 common high pressure flow tap. Distracter is incorrect because a seized RCP shaft would cause its supply breaker to trip on overcurrent. The indication in the stem is that the breaker is closed. Answer is correct because a sheared shaft would cause that loop flow to drop, even while the bezel indication showed the breaker is still closed.

Reference Title

Overhead Annunciators Window D

Learning Objectives

RCPUMPE008	LOR Identify and describe the Control Room controls, indications, and alarms associated with the Reactor Coolant Pump, including: The Control Room location of Reactor Coolant Pump control bezels and indications. (Licensed Operator & STA only) The function of each Reactor Coolant Pump Control Room control and indication. (Licensed Operator & STA only) The effect each Reactor Coolant Pump control has upon Reactor Coolant Pump components and operation. (Licensed Operator & STA only) The plant conditions or permissives required for Reactor Coolant Pump Control Room controls to perform their intended function. (Licensed Operator & STA only) The setpoints associated with the Reactor Coolant Pump control room alarms. (Licensed Operator & STA only)
RCPUMPE009	LOR State the setpoints, coincidence, blocks and permissives for automatic actuations associated with the Reactor Coolant Pump. (Licensed Operator & STA only)

Material Required for Examination			
Question Source:	Facility Exam Bank	**Question Modification Method:**	Direct From Source
Question Source Comments:	Vision Q70312 last NRC Exam usage 4 exams ago (12/2006)		

Given the following conditions:

- Unit 1 is operating at 100% power.
- 13 Charging pump is in service.
- Normal letdown is in service.
- The 1A 4KV to 460V bus feeder breaker opens, deenergizing the 1A 460/230V bus.

With NO operator action, which of the following identifies a consequence, if any, of this event?

a. PZR level wil remain stable.

b. VCT level will be rising at ~ 1% per minute.

c. PZR level will be lowering at ~ 1% per minute.

d. VCT level will be lowering at ~ 4% per minute.

Answer	c	**Exam Level**	R	**Cognitive Level**	Application	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Emergency and Abnormal Plant Evolutions	**RO Group**	1	**SRO Group**	1				000022K103
022	Loss of Reactor Coolant Makeup							**Record Number**	5

AK1. Knowledge of the operational implications of the following concepts as they apply to Loss of Reactor Coolant Makeup:

AK1.03 Relationship between charging flow and PZR level | 3.0 | 3.4

Explanation of Answer

55.41.b(7) Normal power operation has the Positive Diplacment charging pump in service (13), which is powered from 1A 460 volt bus. The two centrifugal charging pumps, (11 and 12) are powered from B and C 4KV buses respectively. The thumbrule for PZR level at NOT is 75 gallons per % of level.(Page 15 of Lesson Plan) When the 1A 460 volt bus is deenergized, the breaker for the 13 charging pump does NOT trip, it does NOT have a UV trip. Therefore, the interlock for automatically closing the 3 letdown oriface isolation valves is not satisfied (all 3 charging pump breakers open.) Letdown reamains in service at 75 gpm, which is normal at power letdown flow. This will cause PZR level to lower at 1% per minute. PZR level would remain stable if it is thought that a charging pump remains in operation because of not knowing correct power supplies. The VCT rule of thumb is 20 gallons per % level. With lewtdown flow still entering the VCT at 75 gpm and no chargin gpump taking suction and pumping from VCT, VCT level will be RISING at ~4% per minute, not lowering. The 1% VCT dstracter is if there is confusion about which (VCT or PZR) will be changing at 1% per minute.

Reference Title

Loss of Charging

Pressurizer and PRT Lesson Plan

Learning Objectives

ABCVC1E001	Describe the impact of the following on unit operation, as applied to AB.CVC-0001: Loss of all charging flow PZR level channel failure VCT level channel failure

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Which of the following is an automatic response for Component Cooling Water system components to a Safety Injection signal on a LOCA, and why?

Assume containment pressure peaks at 5 psig.

a. 2CC215 and 2CC113, Excess Letdown Heat Exchanger CCW isolation valves, receive a close signal to ensure this non-essential containment flow path is isolated.

b. 21 and 22CC16, RHR HX CCW isolation valves, receive an open signal to ensure that long term cooling of the RCS is in service when the swap to Cold Leg Recirc is required.

c. 2CC215 and 2CC113, Excess Letdown Heat Exchanger CCW isolation valves, receive a close signal to ensure ALL CCW supply and return from the containment is isolated.

d. 21 and 22CC16, RHR HX CCW isolation valves, receive an open signal to ensure that RHR pumps do not overheat if the RCS remains above the shutoff heat of the RHR pumps.

Answer	a	**Exam Level**	R	**Cognitive Level**	Memory	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Emergency and Abnormal Plant Evolutions	**RO Group**	1	**SRO Group**	1				000026K302
026	Loss of Component Cooling Water							**Record Number**	6

AK3. Knowledge of the reasons for the following responses as they apply to Loss of Component Cooling Water:

AK3.02 The automatic actions (alignments) within the CCWS resulting from the actuation of the ESFAS | 3.6 | 3.9

Explanation of Answer

55.41.b(7,8) The SI signal sends a close signal to the CC215 and CC113, Excess Letdown HX isolation valves, as they are Containment Phase A isolation valves. The purpose of closing Phase A isolation valves is to ensure all non-essential containment penetrations are isolated on a SI. B is incorrect because CC16s do not receive an open signal until the ARM PB is depressed and the RWST level is 15.2' and manual alignment is required to place ECCS in CLR. C is incorrect because ALL CCW supply and return are not isolated on a SI signal, the RCP CCW is still being supplied until a Phase B signal at 15 psig in cont. D is incorrect because the CC16s do not receive an open signal until the ARM PB is depressed and the RWST level is 15.2', and the RHR pumps are cooled by either flow through the pump from RWST (LBLOCA) or recirc flow (SBLOCA until pp is S/D).

Reference Title

Component Cooling Lesson Plan

Learning Objectives

CCW000E004

NCT Describe the function of the following components and how their normal and abnormal operation affects the Component Cooling Water System:
Component Cooling Water Surge Tank
Component Cooling Water Pumps
Component Cooling Water Heat Exchangers
Isolation/Control Valves
CC-190, RCP Thermal Barrier Discharge Valve
CC-117 & 118, RCP Cooling Water Inlet Valves
CC-136 & 187, RCP Bearing Cooling Outlet Valves
CC-215 & 113, Excess Letdown Heat Exchanger CCW Inlet & Outlet Valves
CC-16, RHR Heat Exchanger Outlet Isolation Valves
CC-17 & 18, CCW Pump Suction Cross-connect Valves
CC-3, Component Cooling Water Pump Outlet
CC-30 & 31, Component Cooling Heat Exchanger Outlet to Auxiliary Header (Non-safety Related Header Isolation Valves)
CC-71, Letdown Temperature Control Valve
CC-149, Surge Tank Vent Valve
CC-131, RCP Thermal Barrier Discharge Flow Control Valve
e. Radiation Monitors

Material Required for Examination

Question Source:	Facility Exam Bank	**Question Modification Method:**	Concept Used
Question Source Comments:	Vision Q42744, made 2 and 2 and added why to stem.		

Salem Unit 1 is operating at 100% power when the PZR Master Pressure Controller demand fails high.

How will this failure affect PZR pressure control components in AUTO?

PZR B/U heaters will be _____. PZR Spray Valves will be _____. BOTH PZR PORV's will be_____.

a. on. shut. open.

b. on. shut. shut.

c. off. open. open.

d. off. open. shut.

Answer	d	**Exam Level**	R	**Cognitive Level**	Application	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Emergency and Abnormal Plant Evolutions	**RO Group**	1	**SRO Group**	1				000027A101
027	Pressurizer Pressure Control Malfunction							**Record Number**	7

AA1.	Ability to operate and / or monitor the following as they apply to Pressurizer Pressure Control Malfunction:		
AA1.01	PZR heaters, sprays, and PORVs	4.0	3.9

Explanation of Answer: 55.41.b(7) Master Pressure controller scale runs from 0-100% demand. With MPC demand failing high, the output of the MPC calls for maximum spray, and no backup heaters. The PZR PORVs are controlled independently of the MPC demand from actual PZR pressure channels 1-4. Since actual PZR pressure is not high, PORVs have no open demand signal.

Reference Title

Pressurizer Pressure and Level Control

Pressurizer Power Relief Valves

Learning Objectives

ABPZR1E001	Describe operation of the Pressurizer Pressure control system as applied to S2.OP-AB.PZR-0001(Q).

Material Required for Examination

Question Source:	Facility Exam Bank	**Question Modification Method:**	Significantly Modified

Question Source Comments: Vision Q50570. Changed from MPC failing low to failing high, which changes correct answer to one of the distracers.

Given the following conditions:

- Unit 1 experienced an ATWT in MODE 1 where both Train "A" and "B" Reactor Trip Breakers (RTBs) failed to open.
- The reactor was tripped when the pressurizer heater buses were deenergized.
- The RTBs remain shut.
- An momentary inadvertent safety injection signal was generated and has cleared.

Which of the following describes the impact of depressing the Train A and Train B RESET SI pushbuttons on 2CC1??

The SI signal will...

a. reset, and Auto SI will be blocked.

b. reset, and Auto SI will NOT be blocked.

c. NOT reset, and Auto SI will be blocked.

d. NOT reset, and Auto SI will NOT be blocked.

Answer	b	**Exam Level**	R	**Cognitive Level**	Application	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Emergency and Abnormal Plant Evolutions	**RO Group**	1	**SRO Group**	1				000029K206
029	Anticipated Transient Without Scram							**Record Number**	8
AK2.	Knowledge of the interrelations between Anticipated Transient Without Scram and the following:								
AK2.06	Breakers, relays, and disconnects							2.9*	3.1*

Explanation of Answer

55.41.b(7) The SI signal can be reset as shown on 221057 grid F-2 AND box and downstream LATCH-RESET. This shows the SI can be reset if 2 conditions are present. 1. Manually pushing the reset pb, (MANUAL SI RESET AND BLOCK), and the 1-2 minute TD has timed out after the SI signal was generated. 2. The LATCH-RESET button is reset. Right next to that AND box is another AND box, whose purpose is to block a second SI after the Rx has been tripped. Since the Rx has not tripped, there is no output from this AND box, and a NO signal will be input into the NOR box to the right of it. The 0 signal into this box will produce a 1 signal out of this box, which is one of 2 inputs to the AND box to its right. This AND box needs 2 signals to produce an output (Auto SI). The second input into this AND box is a safety injection from any of the 4 auto SI signals above it. Hi Steamline Flow with lo steamline pressure or lo-lo Tavg, High Steamline Differential pressure, PZR low pressure, or Containment hi pressure.

Reference Title

RPS Safeguards Actuation Signals

Learning Objectives

RXPROTE027 — LOR Given a Reactor protection System Failure, predict the effect of the Reactor protection System failure on the following: (Licensed Operator and STA Only)

a) Control Rod Drive System
b) Main Turbine/Generator
c) Engineering Safeguards System
d) Reactor Fuel
e) Reactor Coolant System
f) Containment

Material Required for Examination

Question Source:	Facility Exam Bank	**Question Modification Method:**	Significantly Modified

Question Source Comments:	Vision Q44139. Modifed inadvertent SI signal from remaining present to being clear This changes correct naswer to one of the distracters.

Given the following conditions:

- Unit 2 will be performing a controlled shutdown and cooldown from 100% power due to a 5 gpm tube leak on 22 SG IAW S2.OP-AB.SG-0001, Steam Generator Tube Leak.
- After completing the Immediate Actions of EOP-TRIP-1, Reactor Trip or Safety Injection, following the Rx trip, the RO reports that control rod 2D2 is stuck in the fully withdrawn position.

Which of the following identifies the action, if any, the crew will perform in response to the stuck rod?

a. Initiate a rapid boration for 35 minutes during performance of EOP-TRIP-2.

b. Initiate a rapid boration for 35 minutes in S2.OP-AB.SG-0001 after exiting the TRIP series procedures.

c. No actions are required for a single stuck rod because SDM for the cooldown to 503 degrees is adequate.

d. No actions are required for a single stuck rod until the Auto SI Block is performed during RCS depressurization to 1900 psig.

Answer	b	**Exam Level**	R	**Cognitive Level**	Application	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012

Tier:	Emergency and Abnormal Plant Evolutions	**RO Group**	1	**SRO Group**	1	000038G408

038	Steam Generator Tube Rupture	**Record Number**	9

2.4	Emergency Procedures / Plan

2.4.8	Knowledge of how abnormal operating procedures are used in conjunction with EOPs.	3.8	4.5

Explanation of Answer

55.41.b(10) Step 3.26.H in AB.SG states to trip the turbine, then trip the Rx at 20% power during the shutdown. The 5 gpm size of the leak will allow for a controlled shutdown, and will also allow the crew to transition to TRIP-2. There are no SGTR diagnostic steps in TRIP-2 that would cause a transition to SGTR-1. AB.SG would be re-entered at step 3.27 following exit of TRIP-2, and 3.28 directs rapid boration for each stuck rod for 35 minutes. The rapid boration will be initiated before any depressurization starts in 3.29, so the distracter regarding depressurization is incorrect.

Reference Title

Steam Generator Tube Leak

Reactor Trip Response

Learning Objectives

ABSG01E005	For the following analyzed transients/accidents: a) None 1) Determine the expected alarms and indications 2) Describe the analysis assumptions 3) Describe the protective features that mitigate the event. 4) Describe the expected plant response.

Material Required for Examination

Question Source:	Facility Exam Bank	**Question Modification Method:**	Editorially Modified
Question Source Comments:	Vision Q85037		

Given the following conditions:

- Unit 2 is operating at 100% power, MOL.
- The Condensate Polisher is in service -full flow.
- 21 SGFP trips.
- NO operator action is taken in response to the SGFP trip, and the Rx does NOT trip.

Which of the following is an UNEXPECTED alarm if it is locked in 2 minutes after 21 SGFP trips?

a. OHA G-3, EHC SYS TRBL.

b. OHA G-44, COND POL TRBL.

c. Console Alarm RC PRESS DEVIATION HI.

d. Console Alarm RC LOOPS TAVG-TREF DEVIATION.

Answer	c	**Exam Level**	R	**Cognitive Level**	Comprehension	**Facility:**	Salem 1 & 2
Exam Date:	12/3/2012						
Tier:	Emergency and Abnormal Plant Evolutions	**RO Group**	1	**SRO Group**	1		000054G446
054	Loss of Main Feedwater	**Record Number**	10				
2.4	Emergency Procedures / Plan						
2.4.46	Ability to verify that the alarms are consistent with the plant conditions.	4.2	4.2				

Explanation of Answer

55.41.b(5)B is incorrect because the condensate polisher trouble alarm will be in due to the CN108s (auto open on a SGFP trip) AND the CN109 being open at the same time (polisher in service). D is incorrect because RC loops Tavg-Tref deviation will be expected as rods are driving in due to the turbine runback to 65%. C is correct because the RC pressure deviation would not be expected, since the setpoint (+75 psig deviation)equates to when the spray valves are full open. The spray valves should be shut after the insurge due to the load rejection and then the large amount of inward rod motion. A is incorrect because G-3 will be in alarm since it receives input from the EHC Control and Status computer, which will have a Loss of Feed pump Runback alarm in. .

Reference Title

Main Feedwater/Condensate System Abnormality

NOS05CN&FDW09

Learning Objectives

ABCN01E001 — Describe the operation of the following system as applied to S2.OP-AB.CN-0001:
- a) Full Flow Demineralizer Operation
- b) Heater string bypass valve operation (CN-4 7)
- c) Feedwater control valves (BF-19 & BF-40).
- d) Main Feed Pump trips
- e) Main Feed Pump speed control
  - i) Include effects on discharge pressure
- f) ADFWCS System

ABCN01E005 — For the following analyzed transients/accidents:
- a) Loss of Normal Feedwater
- b) Excessive Heat Removal Due to Feedwater System Malfunctions
  - i) Determine the expected alarms and indications.
  - ii) Describe the analysis assumptions.
  - iii) Describe the protective features that mitigate the event.
  - iv) Describe the expected plant response.

Material Required for Examination

Question Source: Facility Exam Bank

Question Modification Method: Editorially Modified

Question Source Comments: Vision Q113267. Removed procedure transition part of question to make RO level (alarm not expected)vs. SRO level (unexpected alarm AND what procedure addresses unexpected alarm.)

Given the following condition:

- Unit 2 was operating at 100% power when a total loss of all AC power occurred.
- 15 minutes after the power loss, operators have locally started 2B EDG.

Which of the following is an action that is REQUIRED to have been performed PRIOR to energizing 2B 4KV Vital bus, and why?

a. Shed non-essential DC loads to extend the time the Vital Instrument Inverters can power their AC loads.

b. Initiate and reset SI to prevent the auto start of a centrifugal charging pump and possible thermal shock to the RCP seals.

c. Deenergize ALL SECs and depress stop PBs for SEC actuated components to prevent overloading the 2B 4KV vital bus.

d. Start the Station Blackout Compressor to provide air for operation of 21-24AF11, AUX FEED-S/G LEVEL CONTROL VLVS, to prevent over feeding the SGs when 22 AFW pp starts.

Answer	c	**Exam Level**	R	**Cognitive Level**	Memory	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Emergency and Abnormal Plant Evolutions			**RO Group**	1	**SRO Group**	1		000055A203
055	Station Blackout							**Record Number**	11

EA2.	Ability to determine and interpret the following as they apply to Station Blackout:		
EA2.03	Actions necessary to restore power	3.9	4.7

Explanation of Answer

55.41.b(10) The Continuous Action Step for energizing a denenergized vital bus with an EDG comes AFTER the step to deenergize all SEC's. The Bases Document states on page 15 that the reason to deenergize the SECs and depress the Stop PB for all SEC controlled safety related loads is to prevent the bus from overloading. It additionally states that a further reason is to prevent charging pump automatic start and possible thermal shock to the RCP seals. SI is initiated at Step 21 NOT to prevent a charging pump from running, but rather to prevent the SI actuated valve realignment that will occur if an SI signal is sensed after power is restored. Non essential DC loads are shed at Step 35 to extend the batteries power capability. The SBO is started as part of Blackout Coping Actions in Attachment 2 Part A of AB.LOOP-1. All the distracters are actions which will be taken during an extended loss of all AC power, but the correct answer is the only one that is required to be performed AND has the correct reason for doing it prior to power restoration. D will be performed, but it is NOT the correct reason, and is required within 60 minutes of Blackout. A and B will be performed, but are not required to be performed prior to power restoration.

Reference Title

Loss of All AC Power

Learning Objectives

LOPA00E007	Describe the EOP mitigation strategy for a loss of all AC power.

Material Required for Examination			
Question Source:	Previous 2 NRC Exams	**Question Modification Method:**	Direct From Source
Question Source Comments:	08-01 NRC RO exam (May 2010) Vision Q133648		

Given the following conditions:

- Unit 2 is in MODE 4.
- RCS pressure is 290 psig.
- RHR HX inlet temperature is 270° F.
- 21 RHR pump is in service in shutdown cooling.
- 22 RHR loop is aligned for ECCS.
- A loss of all off-site power occurs.

Which of the following identifies why S2.OP-AB.LOOP-0001, Loss of Off-Site Power directs operators to initiate S2.OP-AB.RHR-0001, Loss of RHR?

a. The 2A SEC trips 21 RHR pump and does not restart it when 2A EDG connects to 2A vital bus.

b. The SEC's trip all running CCW pumps, and they do not restart when the EDGs connect to their respective vital buses.

c. S2.OP-AB.LOOP-0001 does not know what the initial plant conditions are, and always directs initiation of S2.OP-AB.RHR-0001 regardless of whether or not RHR is in operation.

d. The 22RH18, RHR HX Flow Control Valve, fails shut, and the 2RH20 RHR HX Bypass Flow Control Valve fails open. Action is contained in S2.OP-AB.RHR-0001 to re-establish positive control of RHR HX flow.

Answer	a	**Exam Level**	R	**Cognitive Level**	Comprehension	**Facility:** Salem 1 & 2	**Exam Date:** 12/3/2012
Tier:	Emergency and Abnormal Plant Evolutions	**RO Group**	1	**SRO Group**	1		000056K302
056	Loss of Off-Site Power					**Record Number**	12
AK3.	Knowledge of the reasons for the following responses as they apply to Loss of Off-Site Power:						
AK3.02	Actions contained in EOP for loss of offsite power					4.4	4.7

Explanation of Answer

55.41.b(7,8) The loss of off-site power with NO Safety Injection signal is a SEC MODE II actuation, Blackout. All 3 EDG's will start, the SEC will strip all loads of it's vital bus, shut the EDG output breaker and sequnce on BLACKOUT loads. The rHR pumps are NOT blackout loads and will not be started. AB.LOOP-1 asks, at step 3.8, if a RHR pump was running in SDC mode. If the answer is yes, it directs initiation of AB.RHR since the LOOP will result as described above. The CCW pumps WILL be started by their respective SEC's. AB.RHR is NOT always directed, only if a RHR pump was running in SDC mode. The 22RH18 fails as is, and is not the reason for initiating AB.RHR.

Reference Title

Loss of Off-site Power

Learning Objectives

ABLOP1E002	Describe, in general terms, the actions taken in S2.OP-AB.LOOP-0001(q)and the bases for the actions.

Material Required for Examination

Question Source: New | **Question Modification Method:**

Question Source Comments:

Given the following conditions:

- Salem Unit 1 is in MODE 2 performing a startup by control rods IAW S1.OP-IO.ZZ-0003, Minimum Load to Hot Standby.
- Rx power is stable at 4%.
- Vital Instrument Bus 1D inverter output breaker trips and deenergizes 1D 115VAC. Vital Instrument Bus.

One minute after the loss of 1D VIB, which of the following contains the indication(s) that will be illuminated on Reactor Status Panel 1RP4, with NO operator action?

a. Red Reactor Trip lamp.

b. Yellow RCP busses UV for "H" bus lamp.

c. Blue Over Power Rod Stop Manual Bypass for CH IV lamp.

d. Yellow High Flux PRNI CH IV for BOTH High Power and Low Power.

Answer	d	**Exam Level**	R	**Cognitive Level**	Application	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Emergency and Abnormal Plant Evolutions	**RO Group**	1	**SRO Group**	1				000057A203
057	Loss of Vital AC Instrument Bus							**Record Number**	13

AA2.	Ability to determine and interpret the following as they apply to Loss of Vital AC Instrument Bus:		
AA2.03	RPS panel alarm annunciators and trip indicators	3.7	3.9

Explanation of Answer	55.41.b(7) A is incorrect because the reactor has no trip demand from the loss of D VIB. B is incorrect because the CH IV indication is associated with "G" 4KV RCP group bus. 4 group buses are H,E,F,G fro 11, 12, 13, and 14 RCPs. C is incorrect because the over power block bypass must be nmanually aligned. D is correct because the CH IV for BOTH the High power hi flux and low power high flux will be illuminated.

Reference Title
Loss of 1D 115 Vital Instument bus

Learning Objectives

AB1151E001	Describe the operation of the following as applied to S1/S2.OP-AB.115-0001(Q), S1/S2.OP-AB.115-0002(Q), S1/S2.OP-AB.115-003(q)and S1/S2.OP-AB.115-0004(q) a) 115 V Vital Bus distribution b) UPS Operation

Material Required for Examination	

Question Source:	Facility Exam Bank	**Question Modification Method:**	Direct From Source
Question Source Comments:	Vision Q133676		

Given the following conditions:

- Salem Unit 2 is operating at 100% power.
- All Station Air Compressors trip and none can be restarted.
- The Unit 2 ECAC does not start, and cannot be started.
- The Unit 1 ECAC starts and trips after 5 minutes.

Which of the following identifies an action taken on Salem Unit 2 IAW S2.OP-AB.CA-0001, Loss of Control Air, and why?

a. An operator is dispatched to locally shut the 2DR6, AFWST M/U Valve. This is to prevent over flowing the AFWST and causing a spill of hydrogen peroxide to the storm drain system.

b. The Rx is tripped when EITHER Control Air header lowers to <80 psig. This is to prevent an automatic trip on lo-lo SG NR level when the BF19s associated with that header start to drift shut.

c. All Radwaste releases in progress are terminated. This ensures that during a gradual depressurization of the Control Air system a release is not in progress when the dilution medium flowrate may be changing.

d. An operator is dispatched to manually control 23 AFW pump speed which is running at the high speed stop. This ensures the pump does not become steam bound due to the 21-24AF11 valves failing shut with only limited recirc flow provided.

Answer	c	**Exam Level**	R	**Cognitive Level**	Memory	**Facility:** Salem 1 & 2	**Exam Date:** 12/3/2012
Tier:	Emergency and Abnormal Plant Evolutions	**RO Group**	1	**SRO Group**	1		000065G314
065	Loss of Instrument Air					**Record Number**	14
2.3	Radiation Control						
2.3.14	Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.					3.4	3.8

Explanation of Answer: 55.41.b(10) A is incorrect because while the 2DR6 will be operated locally, the concern is the overflow of water with hydrazine in it, not ammonium hydroxide. C is correct because the bases document says that on page 8 of 12. B is incorrect because BOTH CA header pressures have to be below 80 psig before the rx is directed to be tripped, but the reason is correct. D is incorrect because the action is correct, but the reason is wrong. The AF11s fail open, and pump runout is a concern with the speed failed at the high speed stop and higher steam supply pressure present. (page 9 of 12)

Reference Title

Loss of Control Air

Learning Objectives

ABCA01E002	Describe, in general terms, the actions taken in S2.OP-AB.CA-0001(q)and the bases for the actions in accordance with the Technical Bases Document.

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: Vision Q41379 modified from the release valves are shut because…to what do you have to do (terminate any release in progress) and why. The why was the original questions 4 choices. Also modified per technical review comment that B could also be correct

Given the following conditions:

- Operators are performing actions in 2-EOP-FRCC-1, Response to Inadequate Core Cooling.
- With no other RCPs in service, 23 RCP has been started IAW direction in FRCC-1 and Rx core temperature is lowering.
- 23 RCP was the only RCP able to be started.

Which of the following identifies why 23 RCP would be stopped IAW FRCC-1?

a. RVLIS level has risen to >57% which shows that the fuel is covered and injection flow is present.

b. ALL SG NR levels have lowered <9% which indicates insufficient heat transfer will be available in any RCS loop.

c. At least 2 RCS Thots have lowered to <350°Fwhich indicates the core is cool and RCP forced circulation is no longer required.

d. 23 RCP #1 seal D/P has lowered to less than 250 psid which is less than the minimum required to prevent mechanical damage to RCP.

Answer	c	**Exam Level**	R	**Cognitive Level**	Memory	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Emergency and Abnormal Plant Evolutions	**RO Group**	2	**SRO Group**	2				000074K304
074	Inadequate Core Cooling							**Record Number**	15
EK3.	Knowledge of the reasons for the following responses as they apply to Inadequate Core Cooling:								
EK3.04	Tripping RCPs							3.9	4.2

Explanation of Answer

55.41.b(10)The step to isolate the ECCS accumulators is 28.1, just prior to stopping RCPs at step 29. The ECCS accumulators are isolated after intermittent RHR flow has been verified, since this means they have discharged based on RHR discharge pressure capacity. However, SI or Charging system flow is not checked until AFTER the running RCP is stopped in step 29 when RCS Thots (at least 2) are <350°F. There is no concern for RCP damage based on seal D/P. As discussed at Step 23, normal conditions for RCP operation are desired, but not required. The only thing that will prevent RCP start is no SG NR level (<9%) based on potential creep failure of the high temperature SG tubes, but LOSS of SGNR level does not require stopping the RCPS. RVLIS level must be >57% at step 31and be combined with at least 2 RCS Thots <350 and all RCP's already stopped to exit FRCC-1 to LOCA-1.

Reference Title

Inadequate Core Cooling

Learning Objectives

FRCC00E006	Describe the basis for each step, caution, and note in the following: A. EOP-CFST-1, Figure 2 B. 2-EOP-FRCC-1 C. 2-EOP-FRCC-2 D. 2-EOP-FRCC-3
FRCC00E002	Describe the EOP mitigation strategy for the following: A. Response to Inadequate Core Cooling. B. Response to Degraded Core Cooling. C. Response to Saturated Core Cooling Conditions

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:	

With the Unit 2 Rx operating at 100% power, which of the following radiation monitors would be the FIRST to provide indication that a nuclear fuel rod had developed a substantial leak?

a. 2R31, Letdown Line.

b. 2R12B, Containment Iodine.

c. 2R2, Containment 130' elevation.

d. 2R41D, Plant Vent Noble Gas Release Rate.

Answer	a	**Exam Level**	R	**Cognitive Level**	Comprehension	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Emergency and Abnormal Plant Evolutions	**RO Group**	2	**SRO Group**	2				000076A104
076	High Reactor Coolant Activity							**Record Number**	16
AA1.	Ability to operate and / or monitor the following as they apply to High Reactor Coolant Activity:								
AA1.04	Failed fuel-monitoring equipment							3.2	3.4

Explanation of Answer

55.41(11) With the plant operating at 100% power, letdown will be in service. Failed fuel would immediately release fission products into the RCS. The letdown line would transport these radionuclides and be detected quickly by the 2R31. The 2R12B would only see the failed fuel if there were a way for it to get out of the RCS, and then it would have to expand and travel throughout containment. The 2R2 would respond the same way as the 212B but even slower, since its an area monitor and area radiation levels would take a long time to rise. The 2R41D would see increased radiation levels if the letdown fluid were to get outside the letdown line or VCT or charging line, and would be after the 2R31 saw the increase.

Reference Title

Radiation Monitoring System

Charging, Letdown, and Seal Injection

Learning Objectives

ABRC02E001	Describe the operation of the following systems as applied to S2.OP-AB.RC-0002: a) 2R31 Letdown Line Failed Fuel Monitor b) CVCS Demineralizer Operations

Material Required for Examination			
Question Source:	Facility Exam Bank	**Question Modification Method:**	Direct From Source
Question Source Comments:	Vision Q125679		

Given the following conditions:

- Unit 2 is operating at 100% power when the operators receive several alarms related to the 500KV grid.
- The Electric System Operator calls Unit 2 and directs them to perform a rapid load reduction to 875 MW due to grid instability issues.

Which of the following describes how the load reduction will be performed IAW S2.OP-AB.GRID-0001, Abnormal Grid?

At the EHC Console the PO will depress...

a. the GO pushputton, and ensure the runback automatically stops at ~66% turbine power.

b. SMD #2 RUNBACK and GO PBs, then depress HOLD when Main Generator load lowers < 875 MW.

c. SMD #2 RUNBACK and GO PBs, and ensure the load reduction stops automatically at ~66% turbine power.

d. EITHER the GO pushbutton OR SMD #2 RUNBACK and GO PBs, then depress HOLD when Main Generator load lowers < 875 MW.

Answer	b	**Exam Level**	R	**Cognitive Level**	Memory	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012

Tier:	Emergency and Abnormal Plant Evolutions	**RO Group**	1	**SRO Group**	1	000077A102
077	Generator Voltage and Electric Grid Disturbances			**Record Number**	17	

AA1.	Ability to operate and/or monitor the following as they apply to Generator Voltage and Electric Grid Disturbances:		
AA1.02	Turbine / generator controls	3.8	3.7

Explanation of Answer

55.41.b(10,7) AB.GRID directs the load reduction directed by the ESO due to grid instability be performed IAW Att 4, which says to push SMD #2 if the required end point is >765MW AND <942 MW. It says to press HOLD when the MW value is less than or equal to that directed by the ESO. While depressing the GO PB would work, the procedure says to do it a certain way to ensure consistency amongst the crews (Note on Att 4), so it would be wrong. The MT is normally set up to do a 15% per minute runback to 66% turbine load (~810 Mwe), so while it would get load about where it is supposed to, the procedure doesn't allow you to do it that way. Candidate needs to know how to initiate load reduction, and how it is directed to be stopped.

Reference Title

Abnormal Grid

Learning Objectives

ABGRIDE003	Describe, in general terms, the actions taken in S2.OP-AB.GRID-0001(Q) and the bases for the actions.

Material Required for Examination

Question Source:	Facility Exam Bank	**Question Modification Method:**	Editorially Modified

Question Source Comments: Vision Q120134. Modified one distracter from SMD 2 or SMD 3 to SMD 2 or GO PB since SMD 3 only appeared in one choice, and GO only appeared in one choice.

Given the following:

- Unit 1 experienced a Rx trip and Safety Injection from full power due to a RCS leak.
- The control room crew is currently performing 1-EOP-TRIP-3, Safety Injection Termination.
- 11 Charging pump is in service and 12 Charging pump has been secured.
- Charging pump flow through the BIT has been isolated and 1CV68 and 1CV69, Charging Discharge Valves, have been opened.
- The RO fully opens 1CV55, Charging Flow Control Valve, and reports current PZR level is 45% and lowering slowly.

Which of the following describes the action the control room crew should take IAW 1-EOP-TRIP-3?

a. Re-establish charging flow through the BIT and close 1CV68 and 1CV69.

b. Initiate Safety Injection and return to 1EOP-TRIP-1, Rx Trip or Safety Injection, Step 1.

c. Re-start 12 charging pump to establish PZR level stable or rising, and continue in TRIP-3.

d. Allow 5-10 minutes to allow for system conditions to stabilze while monitoring PZR level.

Answer	a	**Exam Level**	R	**Cognitive Level**	Memory	**Facility:** Salem 1 & 2	**Exam Date:** 12/3/2012
Tier:	Emergency and Abnormal Plant Evolutions	**RO Group**	2	**SRO Group**	2		00WE02K201
E02	SI Termination					**Record Number**	18
EK2.	Knowledge of the interrelations between SI Termination and the following:						
EK2.1	Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.					3.4	3.9

Explanation of Answer

55.41(7,8) After stopping one of two centrifugal charging pumps at step 4, charging flow is re-directed from BIT to normal charging line. If this flowpath cannot maintain stable or rising PZR level, the operator will re-establish BIT flow and go to LOCA-2, Post LOCA Cooldown and depressurization since control of RCS inventory is greater than the capacity of normal charging. The basis document specifically says not to re-start the idled CVCS pump, because that would restore subcooling/PZR level and you would end up back at the same step if you went to LOCA-1, then back to TRIP-3. C is incorrect because TRIP-3 is not continued. 12 CVCS pump MIGHT be started IAW CAS to start ECCS pumps as necssary since stem is non-specific about actual PZR level, but continuing in TRIP-3 is not true. Step 7 has operators restablish charging flow through the BIT and go to LOCA-2. There is no cAS action to initiate SI and go back to TRIP-1.. There is no prision (nor reason) to allow plant conditions to stabilize, maximum charging flow should act on PZR level in seconds, not minutes to change level.

Reference Title

Safety Injection Termination

Learning Objectives

TRP003E003	Describe the plant response to actions taken in the following EOP step sequence(s): 2, 3, 6, 8, 10, 12, 25
TRP003E005	Determine the indications that are monitored to ensure proper system/component operation for each step in 2-EOP-TRIP-3

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following conditions:

- Unit 2 was operating at 100% power when the RCS developed a SBLOCA.
- 45 minutes after the trip, 2B 4KV vital bus locked out on bus differential.
- Containment pressure is 2.4 psig and lowering very slowly.
- Operators are now performing actions in 2-EOP-LOCA-2, Post LOCA Cooldown and Depressurization.

Which of the following contains an alarm, which if received during performance of LOCA-2, would require the associated response?

a. PZR Low Level alarm at 17%. Start ECCS pumps as necessary.

b. RWST Lo Level console alarm at 15.2 feet. Transfer RCS to Cold Leg Recirculation.

c. 21 SG Program Deviation Setpoint Actual console alarm at 28% NR level. Open 21AF21 SG Level Control Valve to raise level in 21 SG.

d. OHA A-6 RMS HI RAD OR TRBL associated with 2R53A, 21 MS Line Rad Monitor. Dispatch operators to locate the LOCA Outside Containment.

Answer	b	**Exam Level**	R	**Cognitive Level**	Comprehension	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Emergency and Abnormal Plant Evolutions	**RO Group**	2	**SRO Group**	2				00WE03K103
E03	LOCA Cooldown and Depressurization							**Record Number**	19

EK1.	Knowledge of the operational implications of the following concepts as they apply to LOCA Cooldown and Depressurization:		
EK1.3	Annunciators and conditions indicating signals, and remedial actions associated with the (LOCA Cooldown and Depressurization).	3.5	3.8

Explanation of Answer	55.41.b(10,11) RWST lo level at 15.2' indicates the need to transfer RCS cooling to cold leg recirculation, as identified by the CAS action in LOCA-2. The SG program deviation setpoint is +/-5%, so the alarm at 28% would be valid. However, 22 AFW pump has no power, so opening the 21AF21 would have no effect. The R53s are N2 monitors in the Main Steam Lines, and after the Rx is shutdown do not provide indication of any use. The PZR level at which ECCS pumps are started is 11%, (19% adverse). With containment pressure at 2.4 psig, normal values would be used. The PZR low level alarm comes in at 5% below program, which would be ~22% with the low Tavg expected during a SBLOCA.

Reference Title

Post LOCA Cooldown and Depressurization

Learning Objectives

LOCA02E005	Determine the indications that are monitored to ensure proper system/component operation for each step in POST LOCA COOLDOWN AND DEPRESSURIZATION.

Material Required for Examination			
Question Source:	Facility Exam Bank	**Question Modification Method:**	Editorially Modified
Question Source Comments:	Vision Q127166. Removed procedure from LOCA outside containment, removed response from opening AF21 valve as part of distracter.		

Given the following conditions:

- Unit 2 is attempting to identify and isolate a 400 gpm LOCA into the RHR system which occurred while operating at 75% power.
- 2-EOP-LOCA-6, LOCA Outside Containment, was entered from 2-EOP-TRIP-1, Rx Trip or Safety Injection.
- The source of the water is back leakage from the 23 cold leg injection line.
- A large leak in the RHR system is located on the piping between 21 and 22RH19s, RHR HX DISCH X-CONN VALVES.

Which of the following components, if it failed to respond when directed by LOCA-6, would prevent isolation of the RCS leak outside containment?

a. 2SJ69, RHR SUCT FROM RWST.

b. 22SJ49, RHR DISCH TO COLD LEGS.

c. 22RH19, RHR HX DISCH X-CONN VALVE.

d. 21SJ49, RHR HX DISCH X-CONN VALVE.

Answer	d	**Exam Level**	R	**Cognitive Level**	Application	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Emergency and Abnormal Plant Evolutions	**RO Group**	1	**SRO Group**	1				00WE04A101
E04	LOCA Outside Containment							**Record Number**	20

EA1. Ability to operate and / or monitor the following as they apply to LOCA Outside Containment:

EA1.1 Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features. 4.0 4.0

Explanation of Answer

55.41.b(7,8,10) The leakage from 23 Cold leg flows back through the 21SJ49, then through the 21RH19 cross connect to reach the leak. Leak isolataion is attempted first by ensuring closed the RCS-RHR suction isolation valves RH1 and RH2. Then BOTH the RH19s are shut. Since the leakage from the RCS has to flow through the 21RH19 to reach the leak, and it is a normally open valve. Its failure to reposition when directed would prevent leak isolation. The SJ69 is closed after the leak is isolated as above. The 22SJ49 is on the opposite RHR train. Candidate also has to know which RHR train feeds which cold legs when not cross connected.. 21 RHR feeds 21 and 23 cold legs, while 22 RHR train feeds 22 and 24

Reference Title

LOCA Outside Containment

ECCS Simplified Drawing

Learning Objectives

LOCA06E002	Describe the plant response to actions taken in LOCA OUTSIDE CONTAINMENT

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

FRHS-1, Response to Loss of Secondary Heat Sink, Step 3 asks, "Is RCS pressure greater than ANY intact or ruptured SG pressure".

Which of the following statements is correct if the operator answers NO?

a. IMMEDIATELY go to Step 23, Bleed and Feed Initiation, since there is no decay heat removal occurring through the SGs.

b. Return to Procedure in effect. Attempts to establish a secondary heat sink would be ineffective at reducing RCS temperature since SG pressure is higher than RCS pressure.

c. Return to Procedure in effect. The RCS has experienced a LOCA large enough that a secondary heat sink is NOT required because decay heat is being removed by break flow.

d. IMMEDIATELY trip all RCPs to prevent further loss of reactor coolant through the LOCA, since a LOOP later in the event could cause a more severe loss of reactor coolant or two-phase RCS flow.

Answer	c	**Exam Level**	R	**Cognitive Level**	Memory	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Emergency and Abnormal Plant Evolutions	**RO Group**	1	**SRO Group**	1				00WE05K302
E05	Loss of Secondary Heat Sink							**Record Number**	21
EK3.	Knowledge of the reasons for the following responses as they apply to Loss of Secondary Heat Sink:								
EK3.2	Normal, abnormal and emergency operating procedures associated with (Loss of Secondary Heat Sink).							3.7	4.1

Explanation of Answer	55.41.b(10, 14) The reason for checking RCS pressure > intact or ruptured SG pressure is to check if there is a need to be worried about a secondary heat sink. If RCS pressure is below SG pressures, then a LOCA of sufficient size is present, and break flow will be removing decay heat, along with ECCS injection. Distracter A is incorrect because the criteria for going to bleed and feed is SG WR level. Distracter B is incorrect because a secondary heat sink could actually be established, and could reduce RCS temperature by dumping steam from the SGs. Distracter D is incorrect it is not a CAS of FRHS but is reason for tripping RCPs in TRIP-1 or LOCA-1.

Reference Title
Loss of Secondary Heat Sink

Learning Objectives	
FRHS00E010	Describe the basis for each step, caution, and note, in 2-EOP-FRHS-1 thru 5

Material Required for Examination			
Question Source:	Facility Exam Bank	**Question Modification Method:**	Direct From Source
Question Source Comments:	Vision Q127090		

Given the following conditions:

- Unit 2 has experienced a steam line break inside containment.
- Operators have entered FRTS-1, Response to Imminent Pressurized Thermal Shock.

Why will the operators be instructed to terminate SI and start RCP(s) if possible?

a. The soak required by FRTS-1 requires SI to be secured and RCPs running to provide the ability to use spray to depressurize the primary.

b. The soak required by FRTS-1 requires SI to be secured. RCPs should be started to equalize boron concentration throughout the primary to ensure proper shutdown margin as the RCS cools.

c. Safety Injection flow is a significant contributor to any cold leg temperature decrease or overpressure condition and must be terminated. RCPs are started to minimize temperature gradient across S/G tube sheets.

d. Safety Injection flow is a significant contributor to any cold leg temperature decrease or overpressure condition and must be terminated. RCPs are started to provide mixing of cold SI and warm reactor coolant water.

Answer	d	**Exam Level**	R	**Cognitive Level**	Memory	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Emergency and Abnormal Plant Evolutions	**RO Group**	1	**SRO Group**	1				00WE08K102
E08	Pressurized Thermal Shock							**Record Number**	22

EK1.	Knowledge of the operational implications of the following concepts as they apply to Pressurized Thermal Shock:		
EK1.2	Normal, abnormal and emergency operating procedures associated with (Pressurized Thermal Shock).	3.4	4.0

Explanation of Answer

55.41.b.(10,8)
A- incorrect - purpose for RCPs is not priority in FRTS-1, soak is not basis for SI. B - incorrect - soak not basis. C - incorrect - SI basis correct, RCP basis not accurate. D-correct -page

Reference Title

Response to Imminent Pressurized Thermal Shock Conditions

Learning Objectives

FRTS00E007	Describe the basis for each step, caution, and note in 2-EOP-FRTS-1 & 2, and EOP-CFST-1, Figure 4 & 4A

Material Required for Examination

Question Source:	Facility Exam Bank	**Question Modification Method:**	Direct From Source
Question Source Comments:	Vision Q73425. Originallyon Seabrook 2003 NRC exam.		

Given the following conditions:

- Operators are performing a natural circulation rapid cooldown on Unit 1 IAW 1-EOP-TRIP-5, Natural Circulation Rapid Cooldown Without RVLIS.
- NO RCPs are running or can be started.
- The control room crew has completed the initial RCS cooldown / depressurization to 500°F / 1600 psig.
- The current time is 1300.

Of the following, which one identifies the EARLIEST time RCS Thot temperatures could be reduced below 450°?

Assume the cooldown will start at 1300 and instantaneously be at the maximum rate allowed.

a. 1316.

b. 1331.

c. 1401.

d. 1501.

Answer	b	**Exam Level**	R	**Cognitive Level**	Application	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Emergency and Abnormal Plant Evolutions	**RO Group**	1	**SRO Group**	1				00WE10A102
E10	Natural Circulation with Steam Void in Vessel with/without RVLIS							**Record Number**	23

EA1.	Ability to operate and / or monitor the following as they apply to Natural Circulation with Steam Void in Vessel with/without RVLIS:		
EA1.2	Operating behavior characteristics of the facility.	3.6	3.8

Explanation of Answer

55.41.b(10) RCS temp is at 500 degrees per the stem. The next temp reduction will be to 450° starting at step 9, and the cooldown is directed to be performed at <100°F per hour. This means 30 minutes of cooldown is required. The 1401 distracter is if the 50°/hr rate of step 7 (initial cooldown to 500°F) is used. The 1316 distracter is if the 200°/hr PZR cooldown limit per TS 3.4.10.2.b is used. The 1501 distracter is for both continuity of the choices, and if the 100°F per hour rate is used for an entire hour.

Reference Title

Natural Circulation Rapid Cooldown Without RVLIS

Learning Objectives

TRP004E004	Demonstrate understanding of indications that are monitored during a Natural Circulation Cooldown.

Material Required for Examination

Question Source:	Facility Exam Bank	**Question Modification Method:**	Editorially Modified

Question Source Comments: Vision Q116968. Modified to include procedure name. Added that the C/D to 500°F has been performed. Added 1 minute to each choice since the stem asks when can be below 450, and 30 minutes of cooldown would only get tO 450.

Given the following conditions for Unit 1:

- A reactor trip and SI occurred at 0700 due to a 1500 gpm RCS LOCA.
- RHR system problems have resulted in a loss of recirculation capability.
- Current time is 1300 hours.

Conditions present when transitioning to 1-EOP-LOCA-5, Loss of Emergency Recirculation, from 1-EOP-LOCA-1, Loss of Reactor Coolant, due to the loss of recirc capability are:

- RCS subcooling is 10°F.
- All RCPs are secured
- 11 and 12 Charging Pumps are running
- BIT flow - 350 gpm
- RVLIS full range 95%
- 11 SI Pump flow - 250 gpm
- 12 SI Pump flow - 250 gpm
- Containment pressure 4.1 psig

Which of the following identifies the ECCS pumps that should be run following determination of Minimum SI Flow for Decay Heat Removal?

Assume:

- Equal flow from each Charging Pump, and each pump will supply half the original total flow if the other charging pump is secured.

- Each SI pump flow remains constant if the other SI pump is secured.

- RCS subcooling remains between 10°F - 45°F for the duration of this question.

a. ONE charging pump and BOTH SI pumps.

b. ONE Charging pump and ONE SI pump.

c. ONE Charging pump only.

d. ONE SI pump only.

Answer	d	**Exam Level**	R	**Cognitive Level**	Application	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Emergency and Abnormal Plant Evolutions	**RO Group**	1	**SRO Group**	1				00WE11A202
E11	Loss of Emergency Coolant Recirculation							**Record Number**	24

EA2.	Ability to determine and interpret the following as they apply to Loss of Emergency Coolant Recirculation:		
EA2.2	Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.	3.4	4.2

Explanation of Answer	55.41.b(10,8) A 1,500 gpm RCS LOCA will deplete the RWST in 2.3 hours. The transfer to CL Recirc will already have been performed. During step 14 of LOCA-5, charging pumps will be reduced to ONE centrifugal, and SI pumps will be reduced to ONE. Starting at Step 19 of LOCA-5, with RCP's secured with <50 degrees subcooling, will use Figure A to determine th ECCS flow required vs. time after trip. 6 hours equals 360 minutes, which is ~225 gpm, but definitely LESS THAN 250 gpm. With the stem stating that charging pump flows remain the same, a single charging pump will be insufficient to supply the required flow. A single SI pump, however, supplying 250 gpm will supply sufficient flow.

Reference Title

Loss of Emergency Recirculation

Learning Objectives

LOCA05E007	Determine a discrete path through the LOSS OF EMERGENCY RECIRCULATION.

Material Required for Examination	RO 24 1-EOP-LOCA-5 flowchart pages 1 & 2

Question Source:	Facility Exam Bank	**Question Modification Method:**	Editorially Modified
Question Source Comments:	Vision Q42181, used 3 NRC exams ago (Class 07-01, Aug 2008)		

Given the following conditions:

- Unit 1 has experienced a MSLB at the Main Turbine inlet steam piping.
- All attempts at Main Steamline Isolation have failed.
- Operators have transitioned out of 1-EOP-TRIP-1, Reactor Trip or Safety Injection.
- RCS cooldown rate is 120°/hr.
- RCS pressure is 1300 psig and dropping.
- Charging system SI flowmeter indicates 290 gpm.
- The RCS cooldown is NOT being controlled.

Which choice identifies an action that must be performed IAW 1-EOP-LOSC-2, Multiple Steam Generator Depressurization, and why?

a. Trip all RCP's to minimize heat input to the RCS.

b. Reduce AFW to minimize cooldown while still keeping the SG tubes wet.

c. Stop BOTH RHR pumps to prevent damage to RHR pumps from continued operation above shutoff head.

d. Send operators to close all BF19's, BF40's, and BF22's to re-establish a secondary pressure boundary in any SG.

Answer	b	**Exam Level**	R	**Cognitive Level**	Application	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Emergency and Abnormal Plant Evolutions	**RO Group**	1	**SRO Group**	1				00WE12K101
E12	Uncontrolled Depressurization of all Steam Generators					**Record Number**	25		

EK1.	Knowledge of the operational implications of the following concepts as they apply to Uncontrolled Depressurization of all Steam Generators:
EK1.1	Components:, capacity, and function of emergency systems. 3.4 3.8
Explanation of Answer	55.41.b(10,4)Once out of TRIP-1, no actions other than attempting to close MSLI valve are taken in LOSC-1 prior to going to LOSC-2. Maintaining >1E4 lbm/hr to each S/G keeps tubes from drying out, among other things. Do not trip RCP's because pressure is dropping due to cooldown, and the reason is wrong. Doesn't matter if it's uncontrolled or not. Distracter A is incorrect because we don't close BF22's, but the reason is right. Don't stop RHR pumps because pressure is still dropping, reason is right.

Reference Title

Multiple Steam Generator Depressurization

Learning Objectives

LOSC02E004	state its bases

Material Required for Examination	
Question Source:	Facility Exam Bank
Question Modification Method:	Direct From Source
Question Source Comments:	Vision Q59885. Used on "H" RO NRC Exam (2004, 5 NRC exams ago.)

Given the following conditions:

- Unit 1 has experienced a LBLOCA.
- The crew is responding IAW the EOP network.
- 3 hours after the transfer to Cold Leg Recirc has been accomplished, the STA reports a Purple Path exists for containment Environment due to containment sump level being 80%.

Which of the following would assist in validating that a high containment sump level actually exists?

a. PWST contains 200,000 gallons.

b. Fire Protection Storage Tank levels are both 85%.

c. 3 SW pumps in service with SW header pressure 150 psig.

d. 5 CFCUs running in low speed with SW flow of ~ 1000 gpm each.

Answer	d	**Exam Level**	R	**Cognitive Level**	Application	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Emergency and Abnormal Plant Evolutions	**RO Group**	2	**SRO Group**	2				00WE15G145
E15	Containment Flooding							**Record Number**	26
2.1	Conduct Of Operations								
2.1.45	Ability to identify and interpret diverse indications to validate the response of another indication.							4.3	4.3

Explanation of Answer

55.41b.(7,9)FRCE-2 checks for possible sources of the excessive sump level in containment. They are: CFCU SW flow, FP to containment isolation valve position, CCW Surge Tank level, Demin Water Storage Tank Level, Primary Water Storage Tank Level. PWST of 200,000 gallons is not enough to have raised containment sump level that much. Tank Capacity completely full is ~240,000 gallons.
SW header pressure of 108 psig is within the normal operating range of 105-115 and would have to be a lot lower with 3 SW pumps in service to inject enough water into containment.
Each CFCU SW flow is normally ~1600 gpm, and indicates a 3,000 gpm leak into containment is possible. Fire protection Storage tank level change would be inadequate to raise sump level that much.

Reference Title

Response to High Containment Sump Level

Tank Capacity Data

Learning Objectives

FRCE00E005	Determine the indications that are monitored to ensure proper system/component operation for each step in 2-EOP-FRCE-1 thru 3 and EOP-CFST-1, Figure 5

Material Required for Examination			
Question Source:	Facility Exam Bank	**Question Modification Method:**	Direct From Source
Question Source Comments:	Vision Q87611		

Which of the following identifies the radiation monitor(s) that must be sensing high radiation conditions for either channel of the Subcooling Margin Monitor to automatically shift to the ADVERSE Mode?

a. EITHER R44A OR R44B, Containment High Range.

b. BOTH R44A AND R44B, Containment High Range.

c. EITHER R2, Containment 130', OR R7 In-Core Seal Table.

d. BOTH R2, Containment 130', AND R7 In-Core Seal Table.

Answer	a	**Exam Level**	R	**Cognitive Level**	Memory	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012

Tier:	Emergency and Abnormal Plant Evolutions	**RO Group**	2	**SRO Group**	2	00WE16A202

E16	High Containment Radiation	**Record Number**	27

EA2.	Ability to determine and interpret the following as they apply to High Containment Radiation:		
EA2.2	Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.	3.0	3.3

Explanation of Answer	55.41.b(11) Either of the Containment high Range monitors reaching 1E5 R/hr will automatically place the SMM in ADVERSE Mode. The other area monitors in containment listed do not input into the SMM.

Reference Title

Abnormal Radiation

Learning Objectives

RMS000E007	Identify and describe the Control Room controls, indications, and alarms associated with the Radiation Monitoring System, including: The Control Room location of Radiation Monitoring System control bezels and indications. (Licensed Operator & STA only) The function of each Radiation Monitoring System Control Room control and indication. (Licensed Operator & STA only) The effect each Radiation Monitoring System control has upon Radiation Monitoring System components and operation. (Licensed Operator & STA only) The plant conditions or permissives required for Radiation Monitoring System Control Room controls to perform their intended function.
FRCE00E005	Determine the indications that are monitored to ensure proper system/component operation for each step in 2-EOP-FRCE-1 thru 3 and EOP-CFST-1, Figure 5

Material Required for Examination			
Question Source:	Facility Exam Bank	**Question Modification Method:**	Editorially Modified
Question Source Comments:	Vision Q73944 replaced area monitors outside containment with 2 others inside containment. Removed window dressing.		

Given the following conditions:

- Unit 2 is operating at 70% power and stable after a load reduction was completed 10 minutes ago.
- Rod Control is in MANUAL control.
- The highest actual Tave-Tref deviation is 4.0°F.

Which choice identifies the rod speed that would be present initially if the Rod Control Selector Switch were placed in AUTO?

a. 24 spm.

b. 40 spm.

c. 48 spm.

d. 56 spm.

Answer	b	**Exam Level**	R	**Cognitive Level**	Application	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012

Tier:	Plant Systems	**RO Group**	2	**SRO Group**	2	001000A101
001	Control Rod Drive System			**Record Number**	28	

A1.	Ability to predict and/or monitor changes in parameters associated with operating the Control Rod Drive System controls including:		
A1.01	T-ave. and no-load T-ave	3.8	4.2

Explanation of Answer

55.41.b(6,7)Rod speed is determined in AUTO by Auct High Tavg vs. Tref (PT-505, Turbine steamline Inlet Pressure). It doesn't matter which channel is connected to Terr recorder on 2RP3. The AUTO rod speed program is 8 spm from 1.5-3.0°F deviation. From 3.0-5.0 it ramps up linearly from 8 spm to 72 spm. This correlates to 16 spm per 1/2°F temp change. 8 spm (@3.0) + 32 spm (from 3.0 to 4.0)= 40 spm.
The 56 spm distracter is if a linear ramp from 1.5 - 5.0 degrees was used.
The 24 spm distracter is if the Terr connected to the chart was used.
The 48 spm distracter is normal manual rod control speed. The Power mismatch circuit in rod control will have cycled through over 5 time constants and its effect on rod speed will be zero ten minutes after the load reduction has been stopped.

Reference Title

Rod Control System Lesson Plan

Learning Objectives

RODS00E012	State the setpoints, coincidence, blocks and permissives for automatic actuations associated with the Rod Control and Position Indication Systems

Material Required for Examination			
Question Source:	Facility Exam Bank	**Question Modification Method:**	Direct From Source
Question Source Comments:	Vision Q88062		

Given the following conditions:

- Unit 1 is operating at 30% power returning from a mid-cycle outage.
- Rod Control is in Manual.
- 14 RCP trips.

Which of the following describes how 14 RC Loop Tavg will be affected 5 minutes later with NO operator action when compared to pre-event Tavg in 14 RC Loop?

14 RC Loop Tavg will be…

a. Lower because backflow from the unaffected loops will cause Tc to rise and equal Thot.

b. Higher because backflow from the unaffected loops will cause Tc to rise and equal Thot.

c. Lower because less heat transfer will occur in 14 SG due to the loss of forced flow in that loop.

d. Higher because less heat transfer will occur in 14 SG due to the loss of forced flow in that loop.

Answer	c	**Exam Level**	R	**Cognitive Level**	Comprehension	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Plant Systems	**RO Group**	1	**SRO Group**	1				003000K503
003	Reactor Coolant Pump System							**Record Number**	29

K5.	Knowledge of the operational implications of the following concepts as they apply to the Reactor Coolant Pump System:		
K5.03	Effects of RCP shutdown on T-ave., including the reason for the unreliability of T-ave. in the shutdown loop	3.1	3.5

Explanation of Answer

55.41.b(3) With Rx power <36%, the reactor will not trip upon a loss of a single RCP. 14 RCP will cost down, and flow will reverse in that loop. A differential pressure will exist across a loop in which a RCP is not running due to the head of other RCPs applied to the cold leg side of the vessel. This high pressure (cold leg side) will induce flow in an idle loop in the reverse direction. This means reactor coolant from the cold leg will flow back through the idle RCP and SG. This will lower Tavg in the idle loop. Th and Tc in the affected loop will reverse, and the original Tc will be greater than Th. The 2 distracters with the Th/Tc equaling are incorrect because Tc will rise >Th in that loop.

Reference Title

Reactor Coolant System Lesson Plan

Learning Objectives

RCPUMPE016	LOR Given a Reactor Coolant Pump failure, predict the effect of the Reactor Coolant Pump failure on the following: (License Operator and STA only) Reactor Coolant System Steam Generators Main and auxiliary feedwater Reactor Protection System

Material Required for Examination

Question Source:	New	**Question Modification Method:**	
Question Source Comments:			

Given the following conditions:

- Unit 2 is operating at 100% power.
- 21 Charging pump is C/T
- 23 Charging pump trips.
- 22 Charging pump cannot be immediately started.

Which of the following describes the impact of the loss of Seal Injection Flow?

a. VCT level will lower due to the lower seal leakoff flow, and auto makeup to the VCT will be initiated.

b. If seal injection cannot be restored within 5 minutes, operators will trip the Rx IAW CAS action in S2.OP-AB.RCP-0001, Reactor Coolant Pump Abnormality.

c. Flow from the RCS past the Thermal Barrier heat exchanger will maintain RCP seal temperature and allow plant operation to continue while attempting to restore charging flow.

d. 13 Charging pump (Unit 1) will be started and supply Unit 2 charging header IAW S2.OP-AB.CVC-0001, Loss of Charging. This will require a Unit 1 shutdown to be initiated due to 13 Charging pump suction alignment to the Unit 1 RWST.

Answer	c	**Exam Level**	R	**Cognitive Level**	Memory	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Plant Systems	**RO Group**	1	**SRO Group**	1				003000K602
003	Reactor Coolant Pump System							**Record Number**	30
K6.	Knowledge of the of the effect of a loss or malfunction on the following will have on the Reactor Coolant Pump System:								
K6.02	RCP seals and seal water supply							2.7	3.1

Explanation of Answer

55.41.b(7) A is incorrect because when all 3 charging pump breakers are open, letdown oriface isolation valves automatically shut, so letdown flow is zero. Charging pumps are using no VCT capacity. Seal return will still be going to VCT, so VCT level will be rising. B is incorrect because as AB.RCP directs Rx trip if BOTH seal injection and Thermal Barrier flows are lost, not just one or the other. C is correct because it describes the flowpath of RCP seal cooling flow when normal seal injection flow has been lost. D is incorrect because AB.CVC-1 direct lining up Unit 1 PDP (13), it would require a Unit 2 shutdown, not a Unit 1 shutdown based on higher borated water supplied from Unit 1 RWST to Unit 2 RCS. Trainee may recognize that TS 3.0.3 is present when no charging pumps are available, but correct answer is worded to allow time to attempt to restore charging flow.

Reference Title

Loss of Charging

Reactor Coolant Pump Abnormality

Learning Objectives

RCPUMPE008	LOR Identify and describe the Control Room controls, indications, and alarms associated with the Reactor Coolant Pump, including: The Control Room location of Reactor Coolant Pump control bezels and indications. (Licensed Operator & STA only) The function of each Reactor Coolant Pump Control Room control and indication. (Licensed Operator & STA only) The effect each Reactor Coolant Pump control has upon Reactor Coolant Pump components and operation. (Licensed Operator & STA only) The plant conditions or permissives required for Reactor Coolant Pump Control Room controls to perform their intended function. (Licensed Operator & STA only) The setpoints associated with the Reactor Coolant Pump control room alarms. (Licensed Operator & STA only)

Material Required for Examination

Question Source:	Facility Exam Bank	**Question Modification Method:**	Concept Used
Question Source Comments:	Q41806 concept of RCS flowing up shaft to supply seals. Added what is the effect part to match K/A.		

2CV21, Letdown Demin Bypass Valve, will automatically reposition to bypass the CVCS Mixed Bed Demineralizers at the Letdown HX Outlet temperature of....

a. 120°F to prevent decomposition of the resin beads.

b. 136°F to prevent decomposition of the resin beads.

c. 120°F to prevent excessive boron retention in the resin.

d. 136°F to prevent excessive boron retention in the resin.

Answer	b	**Exam Level**	R	**Cognitive Level**	Memory	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Plant Systems	**RO Group**	1	**SRO Group**	1				004000K416
004	Chemical and Volume Control System							**Record Number**	31

K4.	Knowledge of Chemical and Volume Control System design feature(s) and or interlock(s) which provide for the following:		
K4.16	Temperature at which the temperature control valve automatically diverts flow from the demineralizer to the VCT; reason for this diversion	2.6	3.0

Explanation of Answer

55.41.b(5) The 2CV21 will reposition to divert flow from the CVCS demineralizers at 136°F. All the demineralizers are in series, with the Mixed bed normally in service. The Cation Bed is placed in service for Lithium control at power, and the deborating bed is placed in service at EOL from RCS boron concentration control. The stem states "Mixed Bed Demin". As temperature rises, resins affinity for boron lowers and boron is released in system, not removed more.

Reference Title

PWR Components Lesson Plan

CVCS Demineralizers-Normal Operations

Overhead Annunciators Window E

Learning Objectives

CVCS00E004

LOR NCT Describe the function of the following components and how their normal and abnormal operation affects the Chemical and Volume Control System:
Letdown/Charging
Letdown Isolaiton Valves, CV2, CV277
Regenerative Heat Exchanger
Letdown Orifices
Letdown Orifice Isolation Valves, CV3, CV4, CV5
Letdown Releif Valve, CV6
Letdown Line Containment Isolation Valve, CV7
RHR Flow Control Valve, CV8
Letdown Heat Exchanger
Low Pressure Letdown Control Valve, CV18
Temperature Control Valve, CV21
Demineralizers (Mixed Bed, Cation, and Deborating
Inlet Valve to Deborating Demin, CV27
Reactor Coolant Filter
Diversion Valve, CV35
CVCS Holdup Tanks
Volume Control Tank
VCT Isolation Valves, CV40, CV41
Chemical Mixing Tank
Charging Pumps (Centrifugal and PD)
Miniflow Recirc. Valves, CV139, CV140
Seal pressure Control Valve, CV71
Chg. Line Containment Isol. Valves, CV68, CV69
Charging to Loop 3 Valve, CV77, Loop 4 Valve, CV79
PZR Auxiliary Spray Valve, CV75
CCP Flow Control Valve, CV55
b. RCP Seal Water
Seal Water Injection Filters
Seal Bypass Flow Valve, CV114
Seal Water Return Isolation Valve, CV104
Seal Water Return Relief Valve, CV115
Seal Return Cont. Isol. Valves, CV116, CV284
Seal Return Filter

Seal Water Heat Exchanger
c. Excess letdown
Excess Letdown Isolation Valves, CV278, CV131
Excess Letdown Heat Exchanger
Excess letdown Flow Cotrol Valve, CV132
Excess Letdown Diversion Valve, CV134
d. Makeup
Primary Water Storage Tank
Primary Water Makeup Pumps
Boric Acid Batch Tank
Boric Acid Tanks
Boric Acid Transfer Pumps
Boric Acid Filter
Boric Acid Blender
Primary Water Flow Control Valve, CV179
Boric Acid Flow Control Valve, CV172
Charging Pump Suction Valve, CV185
VCT Makeup Isolation Valve, CV181
Rapid Borate Stop Valve, CV175

Material Required for Examination	
Question Source:	Facility Exam Bank
Question Modification Method:	Concept Used
Question Source Comments:	Vision Q39286 Used concept of letdown flow divert at 136°F, made 2 and 2 and also added why it diverts.

Given the following conditions:

- Unit 2 is in MODE 4.
- RCS Cooldown is in progress.
- 21 RHR Pump and Heat Exchanger are in service to provide shutdown cooling.
- 22 RHR loop is aligned for ECCS.
- CRS directs the RCS cooldown rate be REDUCED.

Of the following, which describes how RHR system flow will be adjusted to lower the coldown rate?

a. Throttle closed on 21RH18, RHR Heat Exchanger Flow Control valve, while throttling closed on 2RH20, RHR Heat Exchanger Bypass valve to maintain total RHR flow constant.

b. Throttle open on 21RH18, RHR Heat Exchanger Flow Control valve, while throttling closed on 2RH20, RHR Heat Exchanger Bypass valve to lower total RHR flow.

c. Throttle open on 21RH18, while throttling open on 2RH20 to raise RHR Heat Exchanger bypass flow.

d. Throttle closed on 21RH18, while throttling open on 2RH20 to maintain total RHR flow constant.

Answer	d	**Exam Level**	R	**Cognitive Level**	Application	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Plant Systems	**RO Group**	1	**SRO Group**	1				005000A402
005	Residual Heat Removal System							**Record Number**	32
A4.	Ability to manually operate and/or monitor in the control room:								
A4.02	Heat exchanger bypass flow control							3.4*	3.1

Explanation of Answer

55.41.b(5,8)Throttling closed on the RHR HX outlet valve while throttling the bypass valve open will pass less water through the RHR heat exchanger, therefore reducing the cooldown rate while maintianing stable total RHR system flow. A is incorrect because throttling closed both the RH18 and RH20 will not maintain flow constant. B & C are incorrect because it would raise the cooldown rate by passing more flow through RHR HX.

Reference Title

Initiating RHR

RHR Simplified Drawing

Learning Objectives

RHR000E004

LOR NCT Describe the function of the following components and how their normal and abnormal operation affects the Residual Heat Removal System:
a) RHR Pumps
b) Refueling Water Storage Tank
c) Heat Exchangers
d) Motor Operated Valves
  i) RH1 and RH2, Inlet Isolation Valves
  ii) RH4, Pump Suction Isolation Valves
  iii) SJ44, Containment Sump Isolation Valves
  iv) SJ69, RWST to RHR Suction
  v) RH29, Miniflow Recirc. Valves
  vi) RH19, Loop Isolation Valves
  vii) SJ45, RHR to SI or Charging/SI Pump Suction
  viii) SJ113, CCP-SIP Suction Cross-Connect Valves
  ix) SJ49 Outlet Isolation Valve
  x) RH26, RHR Hot Leg Isolation
  xi) CS36, Spray Recirculation from RHR Valve
e) Air-Operated Valves
  i) RH18, RHR HX Outlet Valves
  ii) RH20, RHR HX Bypass Valve
f) Other System Valves

i) RH3, RCS to RHR Inlet Relief Valve
ii) RH25, RHR to RCS Hot Leg Relief Valve
iii) SJ48, RHR to RCS Cold Leg Relief Valves
iv) RH12, RHR HX Bypass Valve
v) RH17, RHR to CVCS Letdown
vi) RH21, RHR to RWST

g) Containment Sump Anti-Vortex Baffle
h) Orifices

Material Required for Examination	
Question Source:	Facility Exam Bank
Question Modification Method:	Editorially Modified
Question Source Comments:	Vision Q77960

Given the following conditions:

- Unit 2 is in MODE 5.
- BOTH loops of RHR are in service for Shutdown Cooling.
- RCS temperature is 190°F and stable.
- Each loop is supplying 1800 gpm flow.
- 2RH20 is 10% open.
- Conditions to transition to MODE 4 are NOT met.
- The air line supplying the 21RH18, RHR HX Outlet FCV breaks, and air is lost to 21RH18.

Which of the following describes the initial effect this airline failure will have?

a. There will be no effect on the RHR system.

b. 2RH20 will have to be throttled in the open direction to prevent a RCS cooldown.

c. 22RH18 will have to be throttled in the closed direction to prevent a RCS cooldown.

d. 22RH18 will have to be throttled opened to ensure RCS temperature is maintained <200°F.

Answer	a	**Exam Level**	R	**Cognitive Level**	Memory	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Plant Systems			**RO Group**	1	**SRO Group**	1		005000K410
005	Residual Heat Removal System							**Record Number**	33
K4.	Knowledge of Residual Heat Removal System design feature(s) and or interlock(s) which provide for the following:								
K4.10	Control of RHR heat exchanger outlet flow							3.1	3.1

Explanation of Answer: 55.41.b(7,8) A is correct because the RH18 valves are fail as-is valve. Losing the air supply will not affect the stable system conditions as sdescribed in the stem. B is incorrect but plausible if it is thought the 21RH18 fails open. C is incorrect but plausible if it is thought the 21RH18 fails open. D is incorrect but plausible if it is thought the 21RH18 fails shut.

Reference Title
Loss of Control Air
RHR Simplified Drawing

Learning Objectives

RHR000E004 — LOR NCT Describe the function of the following components and how their normal and abnormal operation affects the Residual Heat Removal System:

a) RHR Pumps
b) Refueling Water Storage Tank
c) Heat Exchangers
d) Motor Operated Valves
  i) RH1 and RH2, Inlet Isolation Valves
  ii) RH4, Pump Suction Isolation Valves
  iii) SJ44, Containment Sump Isolation Valves
  iv) SJ69, RWST to RHR Suction
  v) RH29, Miniflow Recirc. Valves
  vi) RH19, Loop Isolation Valves
  vii) SJ45, RHR to SI or Charging/SI Pump Suction
  viii) SJ113, CCP-SIP Suction Cross-Connect Valves
  ix) SJ49 Outlet Isolation Valve
  x) RH26, RHR Hot Leg Isolation
  xi) CS36, Spray Recirculation from RHR Valve
e) Air-Operated Valves
  i) RH18, RHR HX Outlet Valves
  ii) RH20, RHR HX Bypass Valve
f) Other System Valves
  i) RH3, RCS to RHR Inlet Relief Valve
  ii) RH25, RHR to RCS Hot Leg Relief Valve
  iii) SJ48, RHR to RCS Cold Leg Relief Valves

iv) RH12, RHR HX Bypass Valve
v) RH17, RHR to CVCS Letdown
vi) RH21, RHR to RWST

g) Containment Sump Anti-Vortex Baffle
h) Orifices

Material Required for Examination	

Question Source:	New	**Question Modification Method:**	
Question Source Comments:			

Salem Unit 1 is operating at 100% power when a total loss of ALL AC power occurs.

Which of the following identifies a consequence if ALL AC power remains deenergized for at least one day?

a. Loss of ECCS pumped injection capability coupled with RCP seal leakage will result in core uncovery.

b. Containment degradation due to the sustained pressure above 15 psig after RCDT reliefs lift and remain open.

c. Flooding in containment as RCS inventory is released which will complicate recovery when AC power is restored.

d. Loss of makeup capability to the RWST will result in lowering level below Tech Spec required for accident recovery.

Answer	a	**Exam Level**	R	**Cognitive Level**	Memory	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Plant Systems	**RO Group**	1	**SRO Group**	1				006000K301
006	Emergency Core Cooling System							**Record Number**	34
K3.	Knowledge of the effect that a loss or malfunction of the Emergency Core Cooling System will have on the following:								
K3.01	RCS							4.1	4.2

Explanation of Answer: Containment pressure is expected to rise to ~ 3 psig and 40°F as the RCS drains through the RCP seals. Time to core uncovery as shown on Figure 40 in lesson plan, best case, is <20 hours. RCS inventory will be released to containment, however, the containment is designed for a LBLOCA in which all the mass in the RCS is released to the containment and long term recovery is not affected. RWST level will not be lowering, since it has no flow path nor motive force.

Reference Title
Loss of All AC Power Lesson Plan

Learning Objectives	
LOPA00E002	Explain the response of the reactor coolant pumps seals to a temporary and a sustained loss of seal cooling

Material Required for Examination			
Question Source:	Other Facility	**Question Modification Method:**	Concept Used
Question Source Comments:	DC Cook 2002 NRC Exam, modified to Salem conditions and replaced poor distracters.		

Given the following condition:

- Unit 1 has intiated a Safety Injection in response to a LBLOCA.

Choose the set of valves which would prevent some portion of ECCS injection flow from occurring if they did NOT reposition upon the SI signal.

a. 1SJ12 AND 1SJ13, BIT Outlet.

b. 11-14SJ54, ECCS Accumulator Outlet.

c. 11SJ49 AND 12SJ49, RHR Discharge to Cold Leg.

d. 11SJ44 AND 12SJ44, Containment Sump Isolation.

Answer	a	**Exam Level**	R	**Cognitive Level**	Application	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Plant Systems	**RO Group**	1	**SRO Group**	1				006000K610
006	Emergency Core Cooling System							**Record Number**	35

K6.	Knowledge of the of the effect of a loss or malfunction on the following will have on the Emergency Core Cooling System:		
K6.10	Valves	2.6	2.8

Explanation of Answer

55.41.b(8) A is correct because BIT inlet (SJ4/5) and outlet valves (SJ12/13) are normally shut and receive an open signal from SSPS on a SI. B is incorrect because SJ54 valves are opened and deenergized at 1,000 psig during a plant startup. Stem states LBLOCA so plausible since accumulators will inject during LBLOCA. C is incorrect because SJ49 valves are normally open at power, and do not re-position during a LOCA, but would expect to have ECCS injection flow from RHR pumps during LBLOCA. D is incorrect because SJ44 valves are opened in LOCA-3 during transfer to CL recirc, and do not provide any ECCS injection flow.

Reference Title

ECCS Simplified Drawing

Preparation of the Safety Injection system for Operation

Learning Objectives

ECCS00E016	Given a Emergency Core Cooling System failure, predict the effect of the Emergency Core Cooling System failure on the following: (License Operator and STA only) Reactor Coolant System Containment Nuclear Fuel

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Which of the following describes Pressurizer Relief Tank (PRT) response when a bubble is being drawn in the PZR after a vacuum refill of the Reactor Coolant System IAW S2.OP-SO.RCS-0002, Vacuum Refill of the RCS?

PRT....

a. pressure will rise slowly as operators vent air and non-condensibles by opening the Pressurizer PORVs.

b. level will rise rapidly as the Pressurizer PORVs cycle automatically in response to the solid PZR expanding.

c. pressure will rise rapidly as the Pressurizer PORVs cycle automatically in response to the solid PZR expanding

d. level will rise slowly as operators maintain Pressurizer pressure during RCP bumps by opening the Pressurizer PORVs.

Answer	a	**Exam Level**	R	**Cognitive Level**	Memory	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Plant Systems	**RO Group**	1	**SRO Group**	1				007000K502
007	Pressurizer Relief Tank/Quench Tank System							**Record Number**	36

K5. Knowledge of the operational implications of the following concepts as they apply to the Pressurizer Relief Tank/Quench Tank System:

K5.02	Method of forming a steam bubble in the PZR	3.1	3.4

Explanation of Answer

55.41.b(3) A is correct because operators will perform a 10-15 minute vent of the PZR while drawing a bubble (Step 5.3.28), with PZR level 40-60% (Step 5.3.5). There will be minimal liquid carryover, but venting will slowly raise PRT pressure. B and C are incorrect because the PORVs are controlled in manual. D is incorrect because the RCP bumps are performed prior to a vacuum being used in the RCS, and PORVs are in auto during bumps, but will be opened after the rCP is secured for venting. (Step

Reference Title

Vacuum Refill of the RCS

Learning Objectives

PZRPRTE012	NCT Discuss the procedural requirements associated with the Pressurizer and Pressurizer Relief Tank, including an explanation of major precaution and limitations in the Pressurizer and Pressurizer Relief Tank procedures

Material Required for Examination			
Question Source:	New	**Question Modification Method:**	
Question Source Comments:			

Which one of the following describes the normal and loss-of-air positions of the Component Cooling Water Surge Tank Vent Valve 2CC-149?

2CC149 is normally _____ and fails _____ upon a total loss of its air supply.

a. shut; open.

b. open; open.

c. shut; shut.

d. open; shut.

Answer	d	**Exam Level**	R	**Cognitive Level**	Memory	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Plant Systems	**RO Group**	1	**SRO Group**	1				008000K408
008	Component Cooling Water System							**Record Number**	37

K4.	Knowledge of Component Cooling Water System design feature(s) and or interlock(s) which provide for the following:		
K4.02	Operation of the surge tank, including the associated valves and controls	2.9	2.7

Explanation of Answer: 55.41.b(7). 2CC149 is a normally open vent valve, and fails shut on loss of air (and loss of control power.)

Reference Title

No. 2 Unit Component Cooling

Component Cooling Lesson Plan

Learning Objectives

CCW000E006	NCT Outline the interlocks associated with the following Component Cooling Water System components: CC-149, Surge Tank Vent Valve CC-131, RCP Thermal Barrier Discharge Flow Control Valve CC-16, RHR Heat Exchanger Outlet Isolation Valves

Material Required for Examination

Question Source:	Facility Exam Bank	**Question Modification Method:**	Editorially Modified

Question Source Comments: Vision Q39302 editorially modified to remove window dressing and common components found in all choices.

Following a loss of offsite power, 2A 4KV Vital Bus fails to reenergize.

Which of the following describes the PZR heater group(s) which are available, or will be made available, to maintain PZR pressure while responding IAW TRIP series EOPs?

a. Backup heater group 21 only.

b. Backup heater group 22 only.

c. Both backup heater groups only.

d. All backup and control heater groups.

Answer	a	**Exam Level**	R	**Cognitive Level**	Application	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Plant Systems	**RO Group**	1	**SRO Group**	1				010000K201
010	Pressurizer Pressure Control System							**Record Number**	38
K2.	Knowledge of bus power supplies to the following:								
K2.01	PZR heaters							3.0	3.4

Explanation of Answer

55.41.b(7) Control Group heaters are powered from 2G non vital bus, and does not have an emergency power supply. 21 Backup Heater Group is normally powered from 2G non vital bus, but has an emergency power supply from the 2C vital bus. 22 Backup Heater Group is normally powered from 2E non vital bus, but has an emergency power supply from the 2A vital bus.

Reference Title

2EP 480V Pressurizer Heater Bus One-Line

2GP 480V Pressurizer Heater Bus One-Line

Learning Objectives

PZRP&LE005 — NCT State the power supply to the following Pressurizer Pressure and Level Control components:
a) Variable Heaters
b) Backup Heaters
c) PORV Block Valves

Material Required for Examination

Question Source: Facility Exam Bank

Question Modification Method: Significantly Modified

Question Source Comments: Vision Q58200. Changed loss of 2B bus to loss of 2A bus which changes correct answer from both backup heater groups to only 22 backup heater group.

Given the following conditions:

- Unit 2 is operating at 100% power.
- A power reduction from 100% to 20% Rx power will be performed at 1% per minute IAW S2.OP-AB.LOAD-0001.
- Prior to initiating the down power, the PZR Master Flow Controller is placed in manual and is NOT adjusted during the downpower.

Which of the following is CLOSEST to what actual PZR level will be when the downpower is completed and RCS Tavg is exactly on program?

a. 22%.

b. 28%.

c. 47%.

d. 59%.

Answer	b	**Exam Level**	R	**Cognitive Level**	Comprehension	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Plant Systems	**RO Group**	2	**SRO Group**	2				011000K604
011	Pressurizer Level Control System							**Record Number**	39

K6.	Knowledge of the of the effect of a loss or malfunction on the following will have on the Pressurizer Level Control System:		
K6.04	Operation of PZR level controllers	3.1	3.1

Explanation of Answer

55.41.b(7). Program PZR level is clipped at 59%. As the downpower occurs, there will be an outsurge from the PZR as the RCS contracts due to lowering Tavg. At 20% power, RCS Tavg exactly on program is 551.6°, (AB.ROD-3 Attachment 1) which would give a program level of 28.3%. (AB.ROD-3 Attachment 2). The PZR mass does not change from the downpower. A is incorrect but plausible since it is the no load PZR prograsm level. D is incorrect but plausible if the candidate thinks that with charging in manual PZR level remains at its current level. C is for continuity.

Reference Title

Continuous Rod Motion

Learning Objectives

PZRP&LE008	Identify and describe the Control Room controls, indications, and alarms associated with the Pressurizer Pressure and Level Control System, including: (Licensed Operator & STA only) The Control Room location of Pressurizer Pressure and Level Control System control bezels and indications. The function of each Pressurizer Pressure and Level Control System Control Room control and indication. The effect each Pressurizer Pressure and Level Control System control has upon Pressurizer Pressure and Level Control System components and operation. The plant conditions or permissives required for Pressurizer Pressure and Level Control System Control Room controls to perform their intended function.

Material Required for Examination

Question Source:	New	**Question Modification Method:**	
Question Source Comments:			

With Unit 2 is at 100% power, Containment Pressure Channel I (one) indication became erratic and the channel was removed from service IAW S2.OP-SO.RPS-0005, Placing Containment Pressure Channel in Tripped Condition.

Predict the plant response if Containment Pressure Channel IV (four) subsequently fails high.

a. No response other than channel related alarms.

b. An AUTO Safety Injection actuation on 2/3 channels tripped.

c. Safety Injection, Containment Spray, Main Steamline Isolation and Phase B Isolation all actuate.

d. Main Steamline Isolation and Phase B Isolation. Containment Spray valves reposition but the pumps do not start.

Answer	a	**Exam Level**	R	**Cognitive Level**	Application	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Plant Systems	**RO Group**	1	**SRO Group**	1				012000A301
012	Reactor Protection System							**Record Number**	40
A3.	Ability to monitor automatic operations of the Reactor Protection System including:								
A3.01	Individual channel							3.8	3.9

Explanation of Answer

55.41.b(7) Cont press Channel I only feeds Cont Hi-Hi (Spray act) it does not feed the Cont Hi (SI) circuits. Containment Spray system bistables are energized to actuate, so when the failed channel is removed from service, its Spray actuation bistable is NOT tripped, it is removed from inputting to Spray coincidence to prevent one of the remaining channels from actuating cont spray if it fails. This leaves the SI ciruitry still 2/3 on channels I, II, and III, and the containment spray actuation goes to 2/3 of the remaining channels.

Reference Title

RPS Safeguards Actuation System

Placing a Containment Pressure Channel in the Tripped Condition

Learning Objectives

RXPROTE012	LOR State the setpoints, coincidence, blocks and permissives for all Reactor Trips and Safety Injections actuations (Licensed Operator and STA Only) NCT List all Reactor Trips and Safety Injections (Non-Licensed Operator)

Material Required for Examination			
Question Source:	Facility Exam Bank	**Question Modification Method:**	Direct From Source
Question Source Comments:	Vision Q134992		

During a LOCA, choose the ONLY one of the following which automatically occurs at 15 psig in containment.

Assume there is a 3 minute ramp in containment pressure from 4 psig to 15 psig.

a. Phase A Isolation.

b. Feedwater Isolation.

c. Main Steamline Isolation.

d. Containment Ventilation Isolation.

Answer	c	**Exam Level**	R	**Cognitive Level**	Memory	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012

Tier:	Plant Systems	**RO Group**	1	**SRO Group**	1	013000A403

013	Engineered Safety Features Actuation System	**Record Number**	41

A4.	Ability to manually operate and/or monitor in the control room:		
A4.03	ESFAS initiation	4.5	4.7

Explanation of Answer: 55.41.b(7) All the distracters occur at 4 psig in containment pressure (SI). Only the MSLI occurs on the Hi-Hi Containment pressure signal. The stem states there is a 3 minute time between the 2 pressures, so it is not a relay race as might occur during a LBLOCA.

Reference Title

Safeguards Actuation Signals

Learning Objectives

ESF000E021	State the setpoints for automatic actuations associated with the Engineered Safety Features

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Of the following, which one describes the purpose of the Enginneered Safety Features IAW Salem FSAR?

a. Limiting peak fuel clad temperature to 2500°F.

b. Limiting fission product dispersal to minimize population exposure for an accidental release beyond the containment.

c. Ensure the design of the core, in conjunction with reactor control and protection systems will prevent release of fission products beyond the fuel cladding.

d. Ensure retention of fission products in the Reactor Coolant System (RCS), and for operational and accidental releases beyond the RCS, retention of fission products by the containment.

Answer	d	**Exam Level**	R	**Cognitive Level**	Memory	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Plant Systems		**RO Group**	1	**SRO Group**	1			013000G127
013	Engineered Safety Features Actuation System							**Record Number**	42
2.1	Conduct Of Operations								
2.1.27	Knowledge of system purpose and/or function.							3.9	4.0

Explanation of Answer

55.41.b(3,9) A is incorrect because it is one of the ECCS Acceptance Criteria, not a pupose of ESF. B is incorrect because ESF is designed to keep fission products in containment. C is incorrect because it is a precursor to having ESF components. That is, core design is meant to keep fission products in the fuel, whereas ESF is designed to keep them in the RCS, and upon leakage of the RCS, to keep them in containment. D is correct per Salem UFSAR, Section 6, page 6.1-1 which states:"The engineered safety features are the provisions in the station which embody methods 2 and 3 above....." Methods 2 and 3 are the 2 parts of the correct answers.

Reference Title

Salem UFSAR

Learning Objectives

ESF000E001	State the purpose of the Engineered Safety Features. Include in this the following: a) Residual thermal energy b) Barriers to fission product release

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following conditions:

- Unit 2 is operating at 80% power performing a Tech Spec required load reduction at 1% per minute.
- Rx power must be below 50% in the next 35 minutes.
- Boration and automatic rod control are maintaing RCS Tavg. 1.0-2.0°F above program.

Power Range Nuclear Instrument Channel IV, 1N44, fails HIGH.

Which of the following identifies:

1. The effect this failure will have.
2. The action which should be performed.

a. Control rods begin stepping OUT at 72 spm. Stop the load reduction until ROD STOP BYPASS is placed in BYPASS.

b. Control rods begin stepping IN at 72 spm. Stop the load reduction until ROD STOP BYPASS is placed in BYPASS.

c. Control rods begin stepping IN at 72 spm. Place control rods in manual.

d. Control rods begin stepping OUT at 72 spm. Place control rods in manual.

Answer	c	**Exam Level**	R	**Cognitive Level**	Application	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Plant Systems	**RO Group**	2	**SRO Group**	2				015000A202
015	Nuclear Instrumentation System							**Record Number**	43

A2.	Ability to (a) predict the impacts of the following on the Nuclear Instrumentation System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:		
A2.02	Faulty or erratic operation of detectors or compensating components	3.1	3.5*

Explanation of Answer

55.41.b(6,10) With a Tech Spec required power reduction underway, and having slim margin to achieve it, the load reduction will continue. As in both AB.ROD-3 (Continuous Rod Motion) and AB.NIS-1 Nuclear Instrumentation Malfunction, operators are directed to place control rods in manual (steps 3.1 and 3.1.1 respectively) and adjust rods in manual to control Tavg (steps 3.5 and 3.1.2 respectively.) AB.NIS has step for terminating load reduction if in progress as a OR step with placing rods in manual. The ROD STOP is for outward rod motion only, and does not affect inward rod motion.

Reference Title

Continuous Rod Motion

Nuclear Instrumentation System Malfunction

Learning Objectives

ABNIS1E003	a) determine the appropriate abnormal procedure in accordance with this lesson plan. b) describe the plant response to actions taken in the abnormal procedure in accordance with this lesson plan. c) describe the final plant condition that is established by the abnormal procedure in accordance with this lesson plan.

Material Required for Examination			
Question Source:	New	**Question Modification Method:**	
Question Source Comments:			

Given the following conditions:

- A LOCA is in progress.
- Operators have transitioned out of EOP-TRIP-1, Reactor Trip or Safety Injection.

Which of the following indicates a superheat condition exists in the core, and what CFST is applicable?

a. 5 or more CETs > 1200°F. RED path for Core Cooling.

b. 5 or more CETs > 1200°F. PURPLE path for Core Cooling.

c. 5 or more CETs > 700°F with RVLIS Full Range 51%. RED path for Core Cooling.

d. 5 or more CETs > 700°F with RVLIS Full Range 51%. PURPLE path for Core Cooling.

Answer	a	**Exam Level**	R	**Cognitive Level**	Application	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Plant Systems	**RO Group**	2	**SRO Group**	2				017000K503
017	In-Core Temperature Monitor System							**Record Number**	44

K5.	Knowledge of the operational implications of the following concepts as they apply to the In-Core Temperature Monitor System:		
K5.03	Indication of superheating	3.7	4.1

Explanation of Answer

55.41.b.(5,10)With CETs indicating >1200°F...."this temperature indicates that most liquid inventory has already been removed from the RCS and that core decay heat is superheating steam in the core."(page 6) This is a FRCC RED path entry to FRCC-1 Response to Inadequate Core Cooling. With CETs >700°F..."superheat at the core exit is indicated. An ICC condition will exist if in the next block RVLIS indicates <3.5 feet collapsed level in the core" 39% RVLIS=3.5'. This means distracter C is incorrect because RVLIS is 51%. D is incorrect because it is a RED path entry to FRCC-1, not a PURPLE path entry into FRCC-2. B is incorrect because it is the wrong (PURPLE path) priority.

Reference Title

Critical Safety Functions Status Trees Basis Document

Learning Objectives

FRCC00E001	State the Red paths for the core cooling status tree

Material Required for Examination

Question Source:	New	**Question Modification Method:**	

Question Source Comments:

Given the following conditions:

- Unit 2 has experienced a LBLOCA coincident with a loss of off site power.
- 2C 4KV vital bus locked out on bus differential.
- 2B SEC did not actuate.

Assuming one train of ECCS equipment is operating, which of the following identifies the FIRST action which will restore the minimum complement of equipment to assure containment integrity is maintained IAW Salem FSAR?

a. Resetting 2C SEC.

b. Depressing START PB for 21 CFCU.

c. Depressing START PB for 22 AND 24 CFCUs.

d. Rotating key switch to ON for 21 Containment Spray pump.

Answer	c	**Exam Level**	R	**Cognitive Level**	Application	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Plant Systems			**RO Group**	1	**SRO Group**	1		022000A102
022	Containment Cooling System							**Record Number**	45

A1. Ability to predict and/or monitor changes in parameters associated with operating the Containment Cooling System controls including:

A1.02 Containment pressure 3.6 3.8

Explanation of Answer

5541.b.(8)FSAR Section 6 and 15 both state that the minimum complement of Containment Spray Pump/CFCUs required to ensure containment integrity along with a train of ECCS in operation is 1 CS pump and 3 CFCUs. With the conditions in the stem, only 21 CS pump will be running on A bus, C bus will be deenergized because a bus differential signal locks out all power to the bus. Additionally, the power supplies to the CFCUs are A,B,C,B,C for 21-25 CFCUs, so only 21 CFCU will be in operation. A is incorrect because the SEC can't start any loads until the bus has power. B is incorrect because it is already running. C is correct because that will restore the 3 CFCUs needed. D is incorrect because 21 CS pump will already be running..

Reference Title

Salem FSAR

Learning Objectives

CSPRAYE002	Describe the design bases of the Containment Spray System.
CONTMTE002	Describe the design bases and values for the following parameters associated with the Containment and Containment Support Systems. (Licensed Operator & STA only) Internal Negative Pressure Internal Positive Pressure Internal Temperature

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: Vision Q113269. Modified Distracter D from 22 to 21 CS pumps since 2 of the answers (A and D) had equipment that would not be available due to loss of 2C bus.

Given the following conditions:

- A loss of reactor coolant has occurred which results in containment pressure rapidly rising to 18 psig.

While walking down the control boards 25 minutes later to prepare for a crew brief, which of the following locked in Overhead alarms would be EXPECTED for these conditions?

a. E-5, SR DET VOLT TRBL.

b. D-43, SPRY ADD TK LVL LO.

c. C-29, 24 CFCU WTRFLO TRBL.

d. B-7, TURB AREA SW HDR PRESS HI.

Answer	b	**Exam Level**	R	**Cognitive Level**	Application	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Plant Systems	**RO Group**	1	**SRO Group**	1				026000G431
026	Containment Spray System							**Record Number**	46
2.4	Emergency Procedures / Plan								
2.4.31	Knowledge of annunciator alarms, indications, or response procedures.							4.2	4.1

Explanation of Answer

55.41.b(8) Spray eductor flow is nominally 75 gpm. Normal level 75%. Administratively max level 90% (3900 gal as shown in SC.CH-AD.CS-0415) Alarm occurs at 67%. Normal level 3,400 gallons. Alarm at 3,050 gallons. D-43 will occur ~ 5 minutes into event. Transition out of TRIP-1 ~ 15 minutes. E-5 is not expected to be in alarm 25 minutes after the trip because it is indication of the SR instruments being energized when they should not be with respect to turbine power above or below 15%. Source range automatically energize between 15-18 minutes following a trip. C-29 is not expected because it indicates CFCU SW valve alignment problem with the CFCU running. It is plausible because the CFCU Airflow Trouble alrm WILL be in alarm, as it occurs when the damper alignment is not correct for running in HIGH speed, and the CFCU will be running in LO speed following an accident. B-7 would not be in alarm as SW to the TGA is automatically isolated by the SECs.

Reference Title

Overhead Annunciators B,C,D,E

Tank Curves

Learning Objectives

CSPRAYE008	Identify and describe the Control Room controls, indications, and alarms associated with the Containment Spray System, including: The Control Room location of Containment Spray System control bezels and indications. (Licensed Operator & STA only) The function of each Containment Spray System Control Room control and indication. (Licensed Operator & STA only) The effect each Containment Spray System control has upon Containment Spray System components and operation. (Licensed Operator & STA only) The plant conditions or permissives required for Containment Spray System Control Room controls to perform their intended function. (Licensed Operator & STA only) The setpoints associated with the Containment Spray System control room alarms. (Licensed Operator & STA only)

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following conditions:

- Salem Unit 1 was operating at 100% power when a LOCA occurred.
- A manual reactor trip and manual SI were initiated.
- When the Main Generator output breakers opened, a loss of off-site power occurred.
- 1A vital bus locked out on bus differential.

Which of the following identifies which Containment Iodine Removal Units (IRUs) can be started if required?

a. 11 IRU ONLY.

b. 12 IRU ONLY.

c. 11 or 12 IRUs.

d. NEITHER IRU is available.

Answer	d	**Exam Level**	R	**Cognitive Level**	Memory	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Plant Systems	**RO Group**	2	**SRO Group**	2				027000K201
027	Containment Iodine Removal System							**Record Number**	47
K2.	Knowledge of bus power supplies to the following:								
K2.01	Fans							3.1*	3.4*

Explanation of Answer	55.41.b(9) Containment IRUs are powered from G and E non-vital 460VAC. With the loss of off-site power, none of the non vital busses are energized. The distracters are based on the operator knowing that the loading of a vital bus in Mode IV doesn't have any bearing on IRU operation.

Reference Title
1E1 Aux Building 460-230V One line
1G1 Aux Building 460-230V One line

Learning Objectives	
CONTMTE004	State the power supply to the following Containment and Containment Support Systems components, including voltage level and 1E/Non 1E. Containment Fan Cooling Units, including breaker alignment for Fast and Slow speed. Containment Iodine Removal Fans (Licensed Operator & STA only) Control Rod Drive Ventilation Fans (Licensed Operator & STA only) Reactor Nozzle Support Ventilation Fans (Licensed Operator & STA only) Reactor Shield Ventilation Fans (Licensed Operator & STA only) Hydrogen Recombiners (Licensed Operator & STA only)

Material Required for Examination			
Question Source:	New	**Question Modification Method:**	
Question Source Comments:			

Which of the following describes how uncovering of the fuel in the Spent Fuel Pool (SFP) is prevented if a leak were to develop on the in-service Spent Fuel Pool Cooling Pump?

a. Automatic trip of the in-service SFP Cooling pump on low level in the SFP.

b. Automatic makeup to the SFP combined with the lo level alarm to alert the Control Room.

c. Locating the SFP Cooling pump suction line close to the surface of the pool, and an anti-siphon hole in the return line to the SFP.

d. Locating the SFP Cooling pump discharge line close to the surface of the pool and an anti-siphon hole in the suction line to the SFP Cooling pumps.

Answer	c	**Exam Level**	R	**Cognitive Level**	Memory	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Plant Systems		**RO Group**	2	**SRO Group**	2			033000K403
033	Spent Fuel Pool Cooling System							**Record Number**	48
K4.	Knowledge of Spent Fuel Pool Cooling System design feature(s) and or interlock(s) which provide for the following:								
K4.03	Anti-siphon devices							2.6	2.9

Explanation of Answer

55.41.b(4) There are neither auto SFP pump trips nor auto M/U to the SFP. The suction line is located 4 ft below the surface to minimize level lost if a leak were to develop below the level of the pool in the SFP Cooling system, while the return to the SFP is located 6' above the fuel. The return line has a 1/2' hole drilled in it to prevent it from siphoning the level back to the SFP Cooling pumps on a leak.

Reference Title

Spent Fuel Pool Cooling

Learning Objectives

SFP000E013	Given a Spent Fuel Pool Cooling System failure, predict the effect of the Spent Fuel Pool Cooling System failure on the following: (License Operator and STA only) Spent Fuel Ventilation Radiation Monitoring System Spent Fuel Temperature

Material Required for Examination			
Question Source:	Facility Exam Bank	**Question Modification Method:**	Concept Used
Question Source Comments:	Vision Q60120 regarding when in service pump would lose suction		

During a steam leak, the CRS directs the PO to FAST close 11MS167 from the 1CC3 bezel. The PO depresses the NORMAL close PB on 1CC2 instead, and the 11MS167 starts closing hydraulically.

Which choice describes what will happen if the operator then pushes the FAST close PB for 11MS167?

a. The vent valves11MS169 and 11MS171 immediately open, allowing hydraulic pressure to close valve against only atmospheric pressure.

b. The MSIV hydraulic pump immediately stops, depressurizing the hydraulic header, and allows main steam pressure to close the valve.

c. The hydraulic sequence will continue until the Valve Fully Closed (33CVO) contact is closed. All other operation of the valve is locked out until the hydraulic pump is deenergized.

d. A solenoid valve immediately opens, equalizing hydraulic pressure on both sides of the operating piston, and vent valves 11MS169 and 11MS171 open to allow main steam pressure to close the valve.

Answer	d	**Exam Level**	R	**Cognitive Level**	Application	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Plant Systems			**RO Group**	1	**SRO Group**	1		039000G128
039	Main and Reheat Steam System							**Record Number**	49
2.1	Conduct Of Operations								
2.1.28	Knowledge of the purpose and function of major system components and controls.							4.1	4.1

Explanation of Answer

55.41.b(4) Using Logic drawings 239916 and 239917: The Emergency Trip signal is generated from the (Fast) CLOSE PB, and acts just the same as a Safeguards Train MSLI or High Stm Line Flow SI signal. SV-1 closes (was open to direct hydraulic pressure to bottom of hydraulic piston.) SV-3 opens, equalizing hydraulic pressure on both sides of the hydraulic piston, and allows the hydraulic fluid to act as buffer to prevent 11MS167 from slamming closed. The solenoids for 11MS169 and 11MS171 open, venting air, and the valve open to allow MS pressure on the bottom of the lower operating piston to drive the disc up against atmospheric pressure. The hydraulic pump does immediately stop running.

Reference Title

Main Steam System Stop Valves Vent Valves

Main Steam System 11MS167 Stop Valves Hydraulic Control

Learning Objectives

MSTEAME005	NCT State the power supply to the following Main Steam System components: Main Steam Isolation Valve û Hydraulic Pumps Main Steam Isolation Valve Vent Valves

Material Required for Examination

Question Source:	Facility Exam Bank	**Question Modification Method:**	Direct From Source
Question Source Comments:	Vision Q80671		

Given the following conditions:

- A Unit 2 plant startup is in progress.
- Reactor power is stable at 18%.
- The Main Generator is rolling unloaded at 1800 rpm.
- Main Steam Dumps are controlling in AUTO in MS Pressure control.

MS Dump Pressure setpoint is lowered 5 psig.

With no other operator action, several minutes later you will notice:

a. Reactor Power is <18%.

b. Reactor power is > 18%.

c. Control rods have stepped in.

d. Control rods have stepped out.

Answer	b	**Exam Level**	R	**Cognitive Level**	Application	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012

Tier:	Plant Systems	**RO Group**	1	**SRO Group**	1	039000K508
039	Main and Reheat Steam System			**Record Number**		50

K5.	Knowledge of the operational implications of the following concepts as they apply to the Main and Reheat Steam System:		
K5.08	Effect of steam removal on reactivity	3.6	3.6

Explanation of Answer	55.41.b(5)With steam dumps open in MS pressure control auto, lowering the setpoint will cause stema dumps to open to reduce steam header pressure to setpoint pressure This will cause a higher steam flow and lower temperature, which causes higher Rx power. Control rods are not placed in auto until >P-2, which is 15% Turbine power, which is not online yet, so rods will be in manual and no operator action stated in stem.

Reference Title

Learning Objectives

RXOPERE021	Explain the relationship between steam flow and reactor power given specific conditions.

Material Required for Examination			
Question Source:	Facility Exam Bank	**Question Modification Method:**	Direct From Source
Question Source Comments:	Vision Q67278 modified from rod pull to lowering MS dump setpoint.		

Given the following conditions:

- Unit 1 is operating at 100% power when a Main Turbine trip causes a reactor trip.
- The Main Steam Dumps do NOT arm.

Which of the following describes the effect this failure to arm will have on the Reactor Coolant System over the next 5 minutes.

a. RCS will exceed design pressure of 2485 psig.

b. RCS pressure will rise and PZR spray will keep pressure below 2335 psig.

c. RCS Tavg will stabilize at a temperature below 547°F based on reset curve of SG Safety Valves which have lifted.

d. RCS Tavg will stabilize at a temperature above 547°F based on 11-14MS10 operation in conjunction with at least one SG Safety Valve being open.

Answer	b	**Exam Level**	R	**Cognitive Level**	Application	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Plant Systems	**RO Group**	2	**SRO Group**	2				041000K302
041	Steam Dump System and Turbine Bypass Control							**Record Number**	51
K3.	Knowledge of the effect that a loss or malfunction of the Steam Dump System and Turbine Bypass Control will have on the following:								
K3.02	RCS							3.8	3.9

Explanation of Answer

55.41.b(4,5,) With a reactor trip, core heat production will lower rapidly. The Steam Generator Atmosphereic Reliefs, MS10s, will open to establish RCS temp ~551-552 psig. The RCS pressure will not rise enough to open the PORV's, much less the PZR Safeties, which are meant to relieve a loss of load with the Rx still at power. PZR Sprays will open rapidly and fully id required to prevent PORV operation. The Tavg distracters will not occur as the Safeties will not open, and would reset well before lowering pressure <1005 psig (no load temp for 547°F)

Reference Title

PZR and PRT Lesson Plan

Learning Objectives

STDUMPE011	LOR State the setpoints, coincidence, blocks and permissives for automatic actuations associated with the Steam Dump System. (Licensed Operator & STA only)

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Of the following, choose the choice which contains ONLY actions that automatically occur on a Unit 2 Main Turbine trip from 100% power, with NO operator action.

I. Running EHC pumps trip
II. 500KV breakers 1-9 and 9-10 open.
III. Emergency Bearing Oil pumps start.
IV. 4KV Vital buses swap power supplies.
V. 4KV Group buses swap power supplies
VI. Main Generator Exciter Field Breaker opens.

a. I, II, III.

b. I, IV, VI.

c. II, V, VI.

d. III, IV, V.

Answer	c	**Exam Level**	R	**Cognitive Level**	Memory	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Plant Systems	**RO Group**	2	**SRO Group**	2				045000A311
045	Main Turbine Generator System							**Record Number**	52
A3.	Ability to monitor automatic operations of the Main Turbine Generator System including:								
A3.11	Generator trip							2.6*	2.9*

Explanation of Answer

55.41.b(4) Running EHC pumps do not auto stop, but plausible because F-32 DEHC trip occurs on turbine trip. 1-9 and 9-10 are the Unit 2 Main Generator output breakers, and they open automatically on every turbine trip. Emergency bearing oil pumps do not start but plausible because aux bearing oil pump will start. 4 KV group buses are powered from APT when Main Generator is operating, an automatically swap to Station Power Transformers powered from off site power upon when the output breakers open. 4KV vital bus swap does not occur as vital buses are powered from off site source. Exciter Field breaker trips upon a Main Turbine trip

Reference Title

Overhead Annunciators Windows F,G,H

Generator Voltage regulator Exciter Field Breaker

Learning Objectives

MNTURBE006	LOR NCT Outline the interlocks associated with the following Main Turbine System components: (Licensed Operator & Non-licensed Operator only) Turning Gear Turbine Drain Valves
EXCTR2E009	Describe the interlocks associated with the following Unit 2 Main Generator Exciter and 25 Kv Systems components: A. Exciter Field Breaker B. Key interlocks associated with the Power Rectifiers C. Main Transformer cooling fans D. Auxiliary Transformer cooling fans

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following conditions:

- Unit 1 is operating at 45% steady state power.
- All Heater Drain Pumps are O/S.
- All Steam Flows and Feed Flows are 40% and stable
- NI's indicate 45% on each channel and stable.
- RCS Tavg/Tref deviation is 0.0°F.

Which of the following is CLOSEST to the programmed value of SG Feed Delta-P IAW S2.OP-SO.CN-0002, Steam Generator Feed Pump Operation?

a. 50 psid.

b. 60 psid.

c. 80 psid.

d. 150 psid.

Answer	c	**Exam Level**	R	**Cognitive Level**	Application	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Plant Systems	**RO Group**	1	**SRO Group**	1				059000A107
059	Main Feedwater System							**Record Number**	53

A1.	Ability to predict and/or monitor changes in parameters associated with operating the Main Feedwater System controls including:		
A1.07	Feed Pump speed, including normal control speed for ICS	2.5*	2.6*

Explanation of Answer

55.41.b(4) SG Feed D/P (delta between feed pressure and SG pressure) is controlled by adjusting SGFP speed, and is programmed based on total % Steam Flow. Actual SGFP speed in rpm is only a result, not a controlled parameter. The K/A intent is met by asking how the feed pressure D/P is controlled, which itself controls SGFP speed. 50 is the minimum D/P from 0-15%. 150 psig is the 100% D/P. 60 psid is if candidate used linear scale from 0%-100% steam flow.

Reference Title

Steam Generator Feed Pump operation

Learning Objectives

CN&FDWE008	LOR Identify and describe the Control Room controls, indications, and alarms associated with the Condensate and Feedwater System, including: The Control Room location of Condensate and Feedwater System control bezels and indications. (Licensed Operator & STA only) The function of each Condensate and Feedwater System Control Room control and indication. (Licensed Operator & STA only) The effect each Condensate and Feedwater System control has upon Condensate and Feedwater System components and operation. (Licensed Operator & STA only) The plant conditions or permissives required for Condensate and Feedwater System Control Room controls to perform their intended function. (Licensed Operator & STA only) The setpoints associated with the Condensate and Feedwater System control room alarms. (Licensed Operator & STA only)

Material Required for Examination

Question Source:	New	**Question Modification Method:**	
Question Source Comments:			

Which of the following identifies how over-cooling of the RCS is prevented on an uncomplicated manual Rx trip from 100% power?

a. P-10 actuates.

b. P-12 actuates.

c. Feedwater Isolation.

d. Feedwater Interlock.

Answer	d	**Exam Level**	R	**Cognitive Level**	Memory	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Plant Systems	**RO Group**	1	**SRO Group**	1				059000K105
059	Main Feedwater System							**Record Number**	54

K1. Knowledge of the physical connections and/or cause-effect relationships between Main Feedwater System and the following:

K1.05 RCS 3.1* 3.2

Explanation of Answer

55.41.b(7)Feedwater interlock actuates when 3/4 RCS Tavgs <554°F and at least one Reactor Trip and associated bypass breaker open. This shuts the BF19's and BF40 Feed Reg Valves. Feedwater Isolation occurs when 2/3 SG NR levels on 1/4 SG's reaches 67% OR on a SI signal, and shuts BF19s, 40s, 13s, and trips the SGFPs. On an uncomplicated Rx trip this will not occur. P-10 is 3/4 PRNIs <10% power, and blocks the low power Rx trips. P-12 is 3/4 RCS Tavgs <543°F, and will shut the Steam Dump valves. On an uncomplicated trip, steam dumps will modulate to control Tavg at 547°, and temp will not reach 543.

Reference Title

Reactor Trip Response

Learning Objectives

CN&FDWE006 LOR NCT Outline the interlocks associated with the following Condensate and Feedwater System components:
Condensate Polishing System Inlet and Bypass Valves
Steam Generator Feedwater Pump Warmup and Suction Isolation Valves
Condensate Pump Discharge and Condensate Polisher Return Chemical Injection Valves

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following conditions:

- Salem Unit 2 is operating at 100% power.
- 21 AFW pump is C/T for pump oil bubbler repair.
- A 400 gpm tube rupture occurs on 23 SG.
- The Rx is tripped and a SI initiated successfully.
- 2A 4KV vital bus locks out on Bus Differential.

Which of the following describes how 23 AFW pump should be utilized IAW 2-EOP-SGTR-1, Steam Generator Tube Rupture?

a. Lower 23 AFW pump speed to minimum and trip 23 AFW pump regardless of MDAFW pump status to terminate the unmonitored radioactive release from its steam exhaust.

b. Lower 23 AFW pump speed to minimum and trip 23 AFW pump. Do not restart until 23MS45 23 SG TO AF PUMP TURB STOP VALVE is shut.

c. Continue running 23 AFW pump since 22 AFW pump will only be supplying feed to a single SG.

d. Continue running 23 AFW pump because it is the only source of feed flow to the SGs.

Answer	b	**Exam Level**	R	**Cognitive Level**	Application	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Plant Systems	**RO Group**	1	**SRO Group**	1				061000K103
061	Auxiliary / Emergency Feedwater System							**Record Number**	55
K1.	Knowledge of the physical connections and/or cause-effect relationships between Auxiliary / Emergency Feedwater System and the following:								
K1.03	Main steam system							3.5	3.9

Explanation of Answer

55.41.b(4,10,13) 23 Turbine Driven AFW pump is supplied from 21 and 23 Main steam lines. Each steam line has a tap upstream of the MSIV. Each line has its own isolation valve (21MS45 and 23MS45) and then its own check valve before the 2 lines combine into one line which feeds the TDAFW pump. During a SGTR, the TDAFW pump remians in service ONLY if it is the SOLE source of feed flow to the SG's. In the stem, 22 AFW pump will be running, since it is powered off of 2B 4KV vital bus, and it has received an auto start signal on SG level. A is incorrect but plausible because there IS an unmonitored rlease from 23 AFW pump since it exhausts directly to atmosphere, but is only secured if there is another source of feed. B is correct per Steps 4.4, 4.5, and 4.7 of SGTR-1, which state to lower speed and trip if it is not the sole source of feed, and to not restart until 23MS45 has been shut. C is incorrect because a single SG being fed is sufficient for heat sink status and 23 AFW pump will not continue to be run. D is incorrect because 22 AFW pump is powered from 2B 4KV vital bus and it remains energized. 21 AFW pump is powered from 2A 4KV vital bus which was lost.

Reference Title

Steam Generator Tube Rupture

Unit 2 Main Steam

Learning Objectives

SGTR01E009	A. Determine a discrete path through the EOP. B. Determine an appropriate transition out of the EOP

Material Required for Examination

Question Source:	New	**Question Modification Method:**	
Question Source Comments:			

Given the following conditions:

- Unit 2 has tripped from 100% power from a faulty SSPS relay on the unaffected train during SSPS testing.
- No Safety Injection has occurred or is required.

Which of the following describes the effect if 21 AFW pump fails to start?

a. Operator action to throttle the 21-24AF11, S/G LEVEL CONTROL VALVES, will be directed to prevent overfeeding the SGs, since 23 AFW pump speed will NOT be lowered to minimum speed unless BOTH AFW pumps are running in EOP-TRIP-2, Rx Trip Response.

b. Operator action to throttle the 21-24AF11, S/G LEVEL CONTROL VALVES, will be directed to prevent overfeeding the SGs, since 23 AFW pump will NOT be secured unless BOTH AFW pumps are running in EOP-TRIP-1 Rx Trip or Safety Injection.

c. Overcooling of the RCS will occur during the first 5 minutes following the trip and a MSLI will be required to limit the excessive cooldown.

d. Overcooling of the RCS will occur during the first 5 minutes following the trip and P-12 will actuate to limit the excessive cooldown.

Answer	a	**Exam Level**	R	**Cognitive Level**	Application	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Plant Systems	**RO Group**	1	**SRO Group**	1				061000K302
061	Auxiliary / Emergency Feedwater System							**Record Number**	56

K3.	Knowledge of the effect that a loss or malfunction of the Auxiliary / Emergency Feedwater System will have on the following:

K3.02	S/G	4.2	4.4

Explanation of Answer

55.41.b(4) 21 MDAFW pump supplies AFW flow to 23 and 24 SG. 23 TDAFW pp supplies all 4 SGs. Following a Rx trip, operators will transition to TRIP-2 after the Immediate Actions of TRIP-1 are performed and a SI is not required. After stopping the SGFPs in TRIP-2, step 3, 23 AFW pp speed is lowered to minimum or 22E4 lbm/hr. Since there would be no flow to 23 and 24 SGs if speed was lowered to minimum, operators will throttle the AF11s to balance flow to each of the SGs and maintain levels and pressures approximate. C and D are incorrect because overfeeding will NOT occur since operators are directed to lower AFW flow. B is incorrect because operators will have transitioned to TRIP-2.

Reference Title

Reactor Trip Response

Learning Objectives

AFW000E015	NCT Given plant conditions, relate the Auxiliary Feedwater System with the following: Steam Generators Main Feedwater System Main Steam System Reactor Coolant System Condensate System Demineralized Water System Service Water System Fresh Water Fire Protection System Room Coolers Safeguards Equipment Controllers (SEC)

Material Required for Examination

Question Source:	Facility Exam Bank	**Question Modification Method:**	Direct From Source
Question Source Comments:	Vision Q134732. Used on Salem 07-01 NRC RO exam.(Q52, 3 exams ago)		

Which of the following describes how control room instrumentation will be affected if the 1C Vital Instrument Bus (VIB) Inverter were to experience a latched transfer?

Instrumentation powered from 1C VIB…

a. will be unaffected by the transfer since it occurs in milli-seconds.

b. must be declared INOPERABLE until the VIB inverter is restored to its normal power supply.

c. will indicate flashing low during the transfer (1-2 seconds), but return to full functional status.

d. AND 1D VIB will momentarily be lost during the transfer since their inverters are powered from the same 230 VAC source.

Answer	a	**Exam Level**	R	**Cognitive Level**	Memory	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Plant Systems	**RO Group**	1	**SRO Group**	1				062000A103
062	A.C. Electrical Distribution							**Record Number**	57

A1.	Ability to predict and/or monitor changes in parameters associated with operating the A.C. Electrical Distribution controls including:		
A1.03	Effect on instrumentation and controls of switching power supplies	2.5	2.8

Explanation of Answer

55.41.b(7) B is incorrect because the VIB, and the instrumentation powered from it, remain OPERABLE as long as the inverter is powering the Vital Bus. (P&L 3.5) The transfer of a VIB inverter takes 2/3 of 1 cycle, which is 11.1 milli seconds, which will not give enough time for the lights to respond on the instrumentation. D is incorrect because indication won't be lost, and 1D VIB is powered from 1B bus.

Reference Title

1C Vital Instrument Bus UPS System Operation

Overhead Annunciator Window B

Learning Objectives

115VACE004	NCT Describe the function of the following components and how their normal and abnormal operation affects the 115VAC Electrical System: A, B, C, and D Vital Bus Inverters Essential Control Power System Emergency Lighting Power System Unit 2 RMS Power System SPDS Power System Control Rod Control and Indication Power Systems Security Power System Telecommunications Power System
115VACE005	NCT State the power supply to the following 115VAC Electrical System components: A, B, C, and D Vital Bus Inverters Essential Control Power System Emergency Lighting Power System Unit 2 RMS Power System SPDS Power System Control Rod Control and Indication Power Systems Security Power System Telecommunications Power System

Material Required for Examination

Question Source:	Facility Exam Bank	**Question Modification Method:**	Direct From Source
Question Source Comments:	Vision Q111931		

Choose the one component below which does NOT receive power from the Unit 1 Vital 125 VDC system.

a. 11 Essential Controls Inverter.

b. 1D Vital Instrument Bus Inverter.

c. 1H 4KV Group Bus control power.

d. Supervisory Control and Data Acquisition (SCADA) System.

Answer	d	**Exam Level**	R	**Cognitive Level**	Memory	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Plant Systems	**RO Group**	1	**SRO Group**	1				063000K201
063	D.C. Electrical Distribution							**Record Number**	58
K2.	Knowledge of bus power supplies to the following:								
K2.01	Major dc loads							2.9*	3.1*

Explanation of Answer

55.41.b(4) The SCADA system is powered from the Circ Water 125 VDC system, not the vital 125VDC system. All the distracters are powered from the 125 VDC vital buses. 11 Essentail controls inverter recieves power from 1C 125 VDC bus. 1D Vital Instrument Bus Inverter recieves power from 1B 125 VDC bus. The 1H 4KV Group Bus recieves its control power from 1A 125 VDC bus (reg) and 1B 125 VDC (emerg)

Reference Title

No. 1 Unit 125 VDC One Line

DC Electrical Systems Lesson Plan

Learning Objectives

DCELECE007	NCT Identify and describe the local controls and indications associated with the DC Electrical System, including: The location of DC Electrical System local controls and indications. (Licensed Operator & Non-licensed Operator only) The function of DC Electrical System local controls and indications. (Licensed Operator & Non-licensed Operator only) The plant and conditions or permissives required for DC Electrical System local controls to perform their intended function. (Licensed Operator & Non-licensed Operator only) The setpoints associated with the DC Electrical Systems local alarms.

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Which of the following describes the Design Basis for the capacity of the EDG Diesel Fuel Oil Storage Tanks (DFOSTs) following a LOCA coincident with a LOOP?

_____ DFOST(s) are / is designed to supply _____ EDGs continuously for 4.5 days.

a. EACH; TWO.

b. EACH; THREE.

c. The COMBINED volume of BOTH; TWO

d. The COMBINED volume of BOTH; THREE.

Answer	c	**Exam Level**	R	**Cognitive Level**	Memory	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Plant Systems	**RO Group**	1	**SRO Group**	1				064000K103
064	Emergency Diesel Generators							**Record Number**	59

K1.	Knowledge of the physical connections and/or cause-effect relationships between Emergency Diesel Generators and the following:		
K1.03	Diesel fuel oil supply system	3.6	4.0

Explanation of Answer

55.41.b(8) Salem FSAR states....The combined volume of both 30,000 gallon fuel oil storage tanks contains sufficient fuel oil at the Technical Specification minimum volume to supply two diesel generators, operating at the most limiting accident mitigation profile for LOCA with loss of offsite power, for approximately 4.5 days."

Reference Title

Salem UFSAR

Learning Objectives

EDG000E004	NCT Describe the function of the following components and how their normal and abnormal operation affects the Emergency Diesel Generators: Lubricating Oil System Jacket Cooling Water System Fuel Oil System Starting Air System Turbo-charger Turbo Boost Air System Exciter û Regulator Governor/Speed Control
EDG000E002	LOR Describe the design bases of the Emergency Diesel Generators. (Licensed Operator & STA only)

Material Required for Examination			
Question Source:	New	**Question Modification Method:**	
Question Source Comments:			

If a leak were to develop on 21 RHR pump room cooler, which of the following tanks would show rising level because of the leak?

a. Auxiliary Building Sump Tank.

b. In-service Waste Hold Up Tank.

c. In-service CVCS Hold Up Tank.

d. Laundry, Hot Shower, and Chemical Drain Tank.

Answer	b	**Exam Level**	R	**Cognitive Level**	Application	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Plant Systems	**RO Group**	2	**SRO Group**	2				068000K107

068	Liquid Radwaste System	**Record Number**	60
K1.	Knowledge of the physical connections and/or cause-effect relationships between Liquid Radwaste System and the following:		
K1.07	Sources of liquid wastes for LRS	2.7	2.9

Explanation of Answer

55.41.b(13) RHR pump rooms each have a sump for recieveing draains and leakage in the room. Each RHR pump room sump is pumped to the inservice Waste Hold up Tank. A is incorrect because the Aux Building sump tank collects floor drains from locations above it, and it is located on 64'. C is incorrect because the CVCS HUT system recieves influent which can be processed and recovered as CVCS quaility water, not floor drains. D is incorrect because Laundry and Hot Shower drain collection points are provided for the contaminated laundry, showers, and sink utilized for protective clothing and personnel decontamination.

Reference Title

Floor Drains - Contaminated

Waste Disposal Liquid

Learning Objectives

WASLIQE004	NCT Describe the function of the following components and how their normal and abnormal operation affects the Radioactive Liquid Waste System: Waste Evaporator Feed Pump Waste Monitor Holdup Tank Pump Auxiliary Building Sump Tank Pumps Chemical Drain Tank and Laundry and Hot Shower Tank Pumps Reactor Coolant Drain Tank Pumps Containment Sump Pumps Reactor Sump Pump RHR Sump Pumps FHB Sump Pump

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following conditions:

- Salem Unit 1 is operating at 100% power, with no equipment out of service.
- Salem Unit 2 is in MODE 5, during a return from a refueling outage.
- Unit 2 is performing a normal Liquid Release from 21 CVCS Monitor Tank via 22 SW header to Unit 1 CW IAW S2.OP-SO.WL-0001, Release of Radioactive Liquid Waste from 21 CVCS monitor Tank.
- The Radwaste Overboard Discharge Flow Recorder 2FR-1064 is OPERABLE.
- The 2R18 Liquid Waste Disposal and the 2R41D Plant Vent Release Rate monitors are OPERABLE.
- After commencing the release, the field operator is recording the Initial Release Data.
- When recording the 2R18 reading, it reads 10 E6 cps.
- The 2R18 red alarm light is lit on the 104 panel.
- The 2WL51 Liquid Release Stop Valve indicates OPEN in the control room.

Which of the following describes what these indications mean, and how the operating crew should proceed?

a. The 2R18 reading is an expected short duration spike for a release of a monitor tank filled with refueling activities liquid. The 2WL51 closure time delay must time out before the 2WL51 would be expected to shut. Continue with the liquid release and ensure 2R18 reading returns to normal.

b. With the 2FR-1064 and 2R41D OPERABLE, block the 2R18 input to the 2WL51 on 2RP1 prior to the time delay timing out and continue the liquid release.

c. The 2WL51 should have immediately automatically closed on the 2R18 high radiation alarm. The NCO in the control room should shut 2WL51.

d. The 2WL51 should have immediately automatically closed on the 2R18 high radiation alarm. The NEO should shut 2WL51 locally.

Answer	c	**Exam Level**	R	**Cognitive Level**	Application	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Plant Systems	**RO Group**	1	**SRO Group**	1				073000K301
073	Process Radiation Monitoring System							**Record Number**	61
K3.	Knowledge of the effect that a loss or malfunction of the Process Radiation Monitoring System will have on the following:								
K3.01	Radioactive effluent releases							3.6	4.2

Explanation of Answer

55.41.b(13) 6.82E5 cps is the ALARM setpoint which will automatically shut the 2WL51 (S2.IC-CC.RM-0028). The current reading is above the setpoint at which the 2WL51 should have automatically shut, and the NCO should shut the valve remotely. The red alarm light for high radiation indicates the 2WL51 should have shut, memorization of alarm setpoint is not necessary. Additionally, S2.OP-SO.WL-0001 for releasing the tank, Step 5.5.9, says if the 2R18 alarms, then the NEO is to inform the NCO to shut the 2WL51. There is no provision for closing the valve locally. There is no time delay for 2WL51 closure, nor is there any provision for releasing a hot tank or is there an expected spike.

Reference Title

Release of Radioactive Liquid Waste from 21 CVCS monitor Tank.

2R18 Liquid Waste Disposal Process Radiation Monitor

Learning Objectives

RMS000E005	NCT Outline the interlocks associated with the following Radiation Monitoring System components:

R1B, Control Room Inlet Duct Monitor
R5, FHB û SFP Area Radiation Monitor
R7, In-core Seal Table Area Radiation Monitor
R9, FHB û New Fuel Storage Area Radiation Monitor
R10A, Personnel Hatch û Containment Elev 100Æ Area Monitor
R10B, Personnel Hatch û Containment Elev 130Æ Area Monitor
R11A, R12A, R12B, Containment Particulate, Noble Gas, and Iodine Monitor
R13A, B, C D & E CFCU Service Water Monitors
R17A and B, Component Cooling Liquid Monitor
R18, Liquid Waste Disposal
R19A, B, C, & D, Steam Generator Blowdown Liquid Monitors
R32A, Fuel Handling Crane Area Radiation Monitor
R36, Evaporator and Feed Preheaters Condensate Monitor
R41D, Plant Vent Radiation Monitor
2R52, Liquid PASS Room Area Radiation Monitor

Material Required for Examination	
Question Source:	Facility Exam Bank
Question Modification Method:	Editorially Modified
Question Source Comments:	Vision Q83676 modified to provide more info in stem about operable equipment and made answers more homogenous.

Given the following conditions:

- Unit 2 is operating at 100% power steady state.
- A field operator reports a SW leak in 2C EDG room, just upstream of 23SW39, 2C DIESEL CLG SW VLV.
- The RO reports Service Water pressure on both 21 and 22 headers has lowered from 112 to 101 psig and continues to lower.

Which of the following describes the expected SW system response, and how the operating crew will respond IAW S2.OP-AB.SW-0001, Loss of Service Water Header Pressure?

Assume SW header pressure can be restored.

The standby SW pump will start when SW header pressure lowers to…

a. 95.0 psig. Lock out 2C EDG and declare it INOPERABLE, shut 21SW21 AND 22SW21, DIESEL CLG SW INLET VALVES, to isolate the leak.

b. 99.5 psig. Lock out 2C EDG and declare it INOPERABLE, shut 21SW21 AND 22SW21, DIESEL CLG SW INLET VALVES, to isolate the leak.

c. 95.0 psig. Lock out 2C EDG and declare it INOPERABLE, isolate the leak by shutting 21SW37 AND 22SW37, 2C DIESEL CLG SW INLET VALVES.

d. 99.5 psig. Lock out 2C EDG and declare it INOPERABLE, isolate the leak by shutting 21SW37 AND 22SW37, 2C DIESEL CLG SW INLET VALVES.

Answer	c	**Exam Level**	R	**Cognitive Level**	Memory	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Plant Systems	**RO Group**	1	**SRO Group**	1				076000A202
076	Service Water System							**Record Number**	62

A2. Ability to (a) predict the impacts of the following on the Service Water System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:

A2.02 Service water header pressure | 2.7 | 3.1

Explanation of Answer: 55.41.b(7, 10)Lock out the EDG(s) that will be affected and isolate the leak. Step 3.11 has operators isolate the leak. EDG must be locked out to prevent starting with no SW available. The only way to isolate the leak is to isolate both supplies from both SW headers by closing both SW37's. Cannot isolate both SW21s per Att 4 , Steps 4.0 B and C because it would render ALL EDGs inoperable. OHA for SW header pressure low states auto start for standy SW pumps is 95.5 psig.

Reference Title

Loss of Service Water Header Pressure

Overhead Annunciators Window B

Learning Objectives

ABSW01E004	Describe, in general terms, the actions taken in S2.OP-AB.SW-0001 and the bases for the actions in accordance with the Technical Bases Document.

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Concept Used

Question Source Comments: Vision Q77578. Changed from what to do to isolate leak (4 choices) to what pressure auto pump will start and how to isolate (made into a "2 and 2" question.

Which of the following identifies normal Control Air header pressure, and the pressure at which the Emergency Air Compressor will automatically start?

a. 100 psig; 85 psig.

b. 110 psig; 90 psig.

c. 100 psig; 90 psig.

d. 110 psig; 85 psig.

Answer	a	**Exam Level**	R	**Cognitive Level**	Memory	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Plant Systems	**RO Group**	1	**SRO Group**	1				078000A301
078	Instrument Air System							**Record Number**	63
A3.	Ability to monitor automatic operations of the Instrument Air System including:								
A3.01	Air pressure							3.1	3.2

Explanation of Answer

55.41.b(7) AB.CA Basis Document states…"When supplied from SA through the dryers, CA pressure runs approximately 5 psig below SA pressure. CA pressure cycles along with the SA cycle. Thus, CA pressures normally run between 95 and 105 psig.The Emergency Control Air Compressor (ECAC), should auto start if CA pressure drops to 85 psig.

Reference Title

Control Air System Operation

Loss of Control Air

Learning Objectives

CONAIRE008	Identify and describe the Control Room controls, indications, and alarms associated with the Control Air System, including: The Control Room location of Control Air System control bezels and indications. (Licensed Operator & STA only) The function of each Control Air System Control Room control and indication. (Licensed Operator & STA only) The effect each Control Air System control has upon Control Air System components and operation. (Licensed Operator & STA only) The plant conditions or permissives required for Control Air System Control Room controls to perform their intended function. (Licensed Operator & STA only) The setpoints associated with the Control Air System control room alarms. (Licensed Operator & STA only)

Material Required for Examination

Question Source: Facility Exam Bank

Question Modification Method: Concept Used

Question Source Comments: Vision Q42094 changed to normal pressure and auto start pressure, from a what starts the ECAC.

Given the following conditions:

- Unit 2 is operating at 100% power when the following occurs:
- OHA A-7, FIRE PROT FIRE.
- 2RP5 Fire Protection Panel, Zone 148 - Work Control Ops Ready Room lamp illuminates, as does the FIRE lamp for that row of alarms.
- The audible coded fire alarm is broadcast over the plant PA system.

For these conditions, which of the following identifies:
1. What these indications mean.
2. How the 2RP5 Fire Protection Panel is reset when the condition has cleared.

a. An active fire suppression system (water/CO2/Halon) has activated. Reset from the Control Room.

b. An active fire suppression system (water/CO2/Halon) has activated. Reset from the Relay Room.

c. A Fire alarm (smoke or heat) has activated. Reset from the Control Room.

d. A Fire alarm (smoke or heat) has activated. Reset from the Relay Room.

Answer	d	**Exam Level**	R	**Cognitive Level**	Application	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Plant Systems	**RO Group**	2	**SRO Group**	2				086000A402
086	Fire Protection System							**Record Number**	64
A4.	Ability to manually operate and/or monitor in the control room:								
A4.02	Fire detection panels							3.5	3.5

Explanation of Answer

55.41.b(4) 1)Early warning detectors provide the Control Room with early indication of fire, but do not cause a suppression system to actuate. Early warning Smoke Detectors and Fire Detectors installed in the plant are arranged to alarm to the Control Room by zone.a) If a detector on a zone actuates, the zone indicating light on Panel RP5 will illuminate along with the appropriate group "FIRE" light, and the coded fire alarm assigned to the zone will be broadcast over the station PA system.
Once a zone actuates, the alarm remains illuminated until the zone is manually reset from the fire protection panels in the Relay Room.

Reference Title

Fire Protection System Lesson Plan

Learning Objectives

FIRPROE008	Identify and describe the Control Room controls, indications, and alarms associated with the Fire Protection System, including: The Control Room location of Fire Protection System control bezels and indications. (Licensed Operator & STA only) The function of each Fire Protection System Control Room control and indication. (Licensed Operator & STA only) The effect each Fire Protection System control has upon Fire Protection System components and operation. (Licensed Operator & STA only) The plant conditions or permissives required for Fire Protection System Control Room controls to perform their intended function. (Licensed Operator & STA only) The setpoints associated with the Fire Protection System control room alarms. (Licensed Operator & STA only)

Material Required for Examination			
Question Source:	New	**Question Modification Method:**	
Question Source Comments:			

Given the following conditions:

- Unit 1 is operating at 100% power.
- Control room operators are preparing to perform a Containment Pressure Relief IAW S1.OP-SO.CBV-0002, CONTAINMENT PRESSURE-VACUUM RELIEF SYSTEM OPERATION.
- Containment radiation levels are NORMAL for 100% power operation with no failed fuel.

After opening the 1VC5 and 1VC6 to initiate the pressure relief, which choice describes how the respective radiation monitors indication will respond?
1R12A - Containment Gas Effluent
1R41B - Plant Vent Noble Gas Intermediate Range
1R41D - Plant Vent Noble Gas Release Rate

a. 1R12A constant; 1R41B constant; 1R41D rises.

b. 1R12A rises; 1R41B constant; 1R41D constant.

c. 1R12A constant; 1R41B rises; 1R41D constant.

d. 1R12A rises; 1R41B rises; 1R41D rises.

Answer	a	**Exam Level**	R	**Cognitive Level**	Application	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Plant Systems	**RO Group**	1	**SRO Group**	1				103000A409
103	Containment System							**Record Number**	65
A4.	Ability to manually operate and/or monitor in the control room:								
A4.09	Containment vacuum system							3.1*	3.7*

Explanation of Answer

55.41.b(11) 1R12A is sampling containment atmosphere, so it will NOT rise when the pressure relief is started. 1R41B is an intermediate range monitor that normally does not have sample flow through it. It's sample flow will start when the lo range 1R41A monitor nears its high end of monitoring range. It's indication will not change during a pressure relief with NORMAL containment radiation levels. The R41D provides the gaseous effluent release rate (uCi/sec) by combining (product of) the on-range R41A through R41C with plant vent flow (cc/sec). It will rise when the pressure relief is initiated, and also provides automatic termination of release on hi gaseous effluent.

Reference Title

Containment Pressure-Vacuum Relief System Operation

Learning Objectives

CONTMTE007	Identify and describe the Control Room controls, indications, and alarms associated with the Containment and Containment Support Systems, including: The Control Room location of Containment and Containment Support Systems control bezels and indications. (Licensed Operator & STA only) The function of each Containment and Containment Support Systems Control Room control and indication. (Licensed Operator & STA only) The effect each Containment and Containment Support Systems control has upon Containment and Containment Support Systems components and operation. (Licensed Operator & STA only) The plant conditions or permissives required for Containment and Containment Support Systems Control Room controls to perform their intended function. (Licensed Operator & STA only) The setpoints associated with the Containment and Containment Support Systems control room alarms. (Licensed Operator & STA only)

Material Required for Examination

Question Source:	Facility Exam Bank	Question Modification Method:	Direct From Source
Question Source Comments:	Vision Q75012		

Given the following conditions:

- After taking the NRC Initial License Exam in May, you are assigned to an operating crew.
- The NRC issues you your Reactor Operator License on June 25th.
- The LORT Annual Requalification Exam (Segment 3) starts on June 26th.

IAW TQ-AA-106, Licensed Operator Requal Training Program, which of the following describes if you are/are not required to participate in the Licensed Operator Requalification Training Program for Segment 3?

a. You ARE required to take the Annual Requal exam AND Segment 3 training.

b. You are NOT required to take the Annual Requal exam nor attend Segment 3 training.

c. You are NOT required to take the Annual Requal Exam but must still attend Segment 3 training.

d. You ARE required to take the Annual Requal exam but do not have to attend Segment 3 training.

Answer	c	**Exam Level**	R	**Cognitive Level**	Memory	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Generic Knowledge and Abilities			**RO Group**	1	**SRO Group**	1		194001G104
GENERIC								**Record Number**	66

2.1	Conduct Of Operations		
2.1.4	Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, “no-solo” operation, maintenance of active license status, 10CFR55, etc.	3.3	3.8

Explanation of Answer

55.41.b(10) "Trainees who obtain a license in a given year may be exempted from taking the Annual Operating Test that is scheduled to be administered during the first requalification training cycle in which the operator participates. However,
operators who complete one or more training cycles before the scheduled Annual Operating Test shall take the test to ensure that they do not exceed the allowed testing intervals." In addition, while the requirement to to take the annual test is waived, the requiremnt to attend requal is not waived, and the operator must attend segment 3 training.

Reference Title

Licensed Operator Requal Training Program

Learning Objectives

Material Required for Examination

Question Source:	New	**Question Modification Method:**	
Question Source Comments:			

The purpose of the Reactor Coolant Pump (RCP) Thermal Barrier is to protect the RCP...

a. motor windings from overheating.

b. motor bearings from overheating.

c. radial bearing and seals from the heat of the RCS.

d. thrust bearing from damage due to excessive heat.

Answer	c	**Exam Level**	R	**Cognitive Level**	Memory	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Generic Knowledge and Abilities			**RO Group**	1	**SRO Group**	1		194001G128
GENERIC								**Record Number**	67

2.1	Conduct Of Operations		
2.1.28	Knowledge of the purpose and function of major system components and controls.	4.1	4.1

Explanation of Answer

55.41.b.(3) The RCP thermal barrier prevents hot reactor coolant from flowing up the shaft to the radial bearing and RCP seal package when normal seal injection flow is lost. The normal flow of cool seal injection water past the raidial barrier into the RCS normally performs that function. In the event of a loss of seal injection flow, reactor coolant would flow up across the tubes of the heat exchanger(cooled by CCW) and into the pump to provide the coolant and lubricant for the radial bearing and seals

Reference Title

Reactor Coolant Pump Lesson Plan

Learning Objectives

RCPUMPE004

LOR NCT Describe the function of the following components and how their normal and abnormal operation affects the Reactor Coolant Pump:
Impeller
Turning Vane Diffuser
Diffuser Adapter
Thermal Barrier and Heat Exchanger
Pump Radial Bearing
Controlled Leakage Seal Assembly
Lower Motor Radial Bearing
Upper Motor Radial Bearing
Flywheel
Anti-Reverse Rotation Device
Oil Lift Pump
Motor Space Heaters

Material Required for Examination

Question Source:	Facility Exam Bank	**Question Modification Method:**	Editorially Modified
Question Source Comments:	Vision Q43989 made pschometrically balanced.		

Which of the following conditions will REQUIRE the suspension of fuel movement in the Unit 2 Rx vessel?

a. Chemistry reports Rx Cavity boron concentration is 2499 ppm.

b. A NEO reports BOTH 100' elevation containment airlock doors are open.

c. Containment Radiation Monitor 2R12A fails causing a Containment Ventilation Isolation signal.

d. BOTH Unit 2 Control Room Ventilation intake duct radiation monitors 2R1B-1 and 1R1B-2 are declared inoperable when intake damper CAA40 shuts, and the PO depresses Fire Outside Control Room.

Answer	d	**Exam Level**	R	**Cognitive Level**	Application	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Generic Knowledge and Abilities	**RO Group**	1	**SRO Group**	1				194001G140
GENERIC								**Record Number**	68

2.1	Conduct Of Operations		
2.1.40	Knowledge of refueling administrative requirements.	2.8	3.9

Explanation of Answer: 55.41.b(11) D is correct because operation in the recirculation mode requires suspension of fuel movement. (SO.CAV P&L 3.6.3 When aligned to FIRE OUTSIDE CONTROL AREA (Recirculation Mode),Core Alterations and movement of irradiated fuel is NOT permitted (T/S Bases 3/4.7.6)). C is incorrect because Containment Radiation monitors are not required to be operable for Mode 6 or Fuel Movement or Core Alts per Tech Specs. B is incorrect because the airlock doors are only required to be CAPABLE of being shut, and can be open (S2.OP-ST.CAN-0007, page 8) A is incorrect because the COLR limit for boron concentration is 2139

Reference Title
Salem Tech Specs
Refueling Operations-Containment Closure
Control Area Ventilation Operation

Learning Objectives	
REFUELE012	Discuss the procedural requirements associated with the Refueling System, including an explanation of major precaution and limitations in the Refueling System procedures. (Licensed Operator & Non-licensed Operator only)

Material Required for Examination			
Question Source:	Facility Exam Bank	**Question Modification Method:**	Direct From Source
Question Source Comments:	Vision Q110741		

Given the following conditions:

- Unit 2 is operating at 100% power.
- Excess Letdown is in service due to a problem with the control circuit for 2CV35, VCT 3 WAY INLET V.
- I&C troubleshooting is in progress on 2CV35.
- The RO reports that 2CV35 has just swapped to the Flow to HUT position.
- The RO also reports that during the pre-job brief for the 2CV35 troubleshooting, it was stated that 2CV35 actual position would NOT be affected during the troubleshooting.

Which of the following identifies the FIRST action the crew should take?

a. Enter S2.OP-AB.CVC-0001, Loss of Charging to address the unanticipated CVCS system lineup.

b. Contact the WCC to initiate a tagout for 2CV35 since the troubleshooting needs to have the valve deactivated.

c. Contact the I&C Supervisor and stop work on 2CV35 based on being outside of Procedures, Parameters or Processes (OOPS) IAW HU-AA-101, Human Error Prevention.

d. Have the RO place 2CV35 in the Auto position to maintain status control since the Component Off Normal and Off Normal Tagged report does not reflect the valves current position.

Answer	c	**Exam Level**	R	**Cognitive Level**	Memory	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Generic Knowledge and Abilities	**RO Group**	1	**SRO Group**	1				194001G220
GENERIC								**Record Number**	69
2.2	Equipment Control								
2.2.20	Knowledge of the process for managing troubleshooting activities.							2.6	3.8

Explanation of Answer

55.41.b(10) Excess letdown does not flow through 2CV35, so its movement will not affect RCS letdown, and AB.CVC-1 will not address the valve movement. While a tagout may need to be initiated, it would not happen without stopping the job in progress and finding out in more detail from I&C how the valve could be affected and what the blocking points needed to be. OOPS requires stopping the job because they system realignment was stated to NOT going to occur during troublshooting. Status control does not require movement of components just to align with the Off Normal, the Off Normal position would be updated in SAP.

Reference Title

HUMAN PERFORMANCE TOOLS AND VERIFICATION PRACTICES

Learning Objectives

MISCAP007	LOR NCT Given a set of plant conditions and SH.OP-AP.ZZ-0103, Component Configuration Control, evaluate the plant conditions and summarize the actions necessary to properly maintain the status of systems and components, IAW SH.OP-AP.ZZ-0103.

Material Required for Examination

Question Source:	Facility Exam Bank	**Question Modification Method:**	Direct From Source
Question Source Comments:	Vision Q125707		

The Unit 2 control room recieves a call from the Rad Waste Operator, who states that an isolated Gas Decay Tank in hold-up has lowered in pressure from 90 psig to 40 psig over the last 2 hours, and continues to lower slowly.

Which of the following would provide confirmation that tank pressure has lowered (vs instrument failure), and why is confirmation important?

a. Display the 2R41D trend reading on 2RP1. An unapproved release may be in progress.

b. Display the 2R41D trend reading on 2RP1. An unmonitored release may be in progress.

c. Direct Rad Pro to locally retrieve trend data for the Area Monitor closest to the GDT area. An unapproved release may be in progress.

d. Direct Rad Pro to locally retrieve trend data for the Area Monitor closest to the GDT area. An unmonitored release may be in progress.

Answer	a	**Exam Level**	R	**Cognitive Level**	Application	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Generic Knowledge and Abilities	**RO Group**	1	**SRO Group**	1		194001G244		
GENERIC								**Record Number**	70

2.2	Equipment Control		
2.2.44	Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.	4.2	4.4

Explanation of Answer: 55.41.b(11) R41 (plant vent) monitors in control room have trend function which can display historical data. If a release is in progress its being monitored, but is unapproved. Area Monitors do not have trend functions locally, but some of them are trended on P-250 computer (R4 and R34)in control room.

Reference Title

Radiation Monitoring System Operation

Learning Objectives

RMS000E007	Identify and describe the Control Room controls, indications, and alarms associated with the Radiation Monitoring System, including: The Control Room location of Radiation Monitoring System control bezels and indications. (Licensed Operator & STA only) The function of each Radiation Monitoring System Control Room control and indication. (Licensed Operator & STA only) The effect each Radiation Monitoring System control has upon Radiation Monitoring System components and operation. (Licensed Operator & STA only) The plant conditions or permissives required for Radiation Monitoring System Control Room controls to perform their intended function.

Material Required for Examination

Question Source: New | **Question Modification Method:**

Question Source Comments:

Which of the following describes when rising radiation levels on 2R19A, STM GEN BLOWDOWN RAD MONITOR, will automatically close the 21GB4, SG B/D OUTLET ISOL VALVE, and why?

2R19A in _______ will close the 21GB4 _____.

a. Warning, to minimize the spread of contamination from a Steam Generator Tube Rupture (SGTR) on 21 Steam Generator to secondary systems.

b. Alarm, to minimize the spread of contamination from a Steam Generator Tube Rupture (SGTR) on 21 Steam Generator to secondary systems.

c. Warning, to prevent backfeeding contamination from 21 Steam Generator to any other Steam Generator through the unaffected Steam Generators blowdown lines.

d. Alarm, to prevent backfeeding contamination from 21 Steam Generator to any other Steam Generator through the unaffected Steam Generators blowdown lines.

Answer	b	**Exam Level**	R	**Cognitive Level**	Memory	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Generic Knowledge and Abilities	**RO Group**	1	**SRO Group**	1				194001G314
GENERIC								**Record Number**	71

2.3	Radiation Control		
2.3.14	Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.	3.4	3.8

Explanation of Answer

55.41.b(11) B is correct because the GB4 is automatically shut on the Hi Rad Alarm, whereas the 21-24GB10s, 21-24185s, and 2GB50 shut on Hi Rad Warning. Isolating the blowdown path from the S/G to the condenser will prevent the spread of contamination, and also will prevent any type of release from the main condenser to atmosphere. Each S/G has its own blowdown line, so backfeeding contamination is not possible through the blowdown lines

Reference Title

Radiation Monitoring System Operation

Learning Objectives

ABSG01E003	Describe, in general terms, the actions taken in S1/S2.OP-AB.SG-0001 and the bases for the actions in accordance with the Technical Bases Document.

Material Required for Examination			
Question Source:	Facility Exam Bank	**Question Modification Method:**	Significantly Modified
Question Source Comments:	Q78166 modified to make a 2 and 2 vs just a why do we isolate blowdown		

Which of the following Area Radiation Monitors is checked in 2-EOP-LOCA-1, Loss of Reactor Coolant, when determining if a LOCA outside Unit 2 containment is occurring?

a. 2R47, Electrical Penetration.

b. 2R34, Mechanical Penetration 100'.

c. 2R5, Fuel Handling Building-Spent Fuel.

d. 2R10A, Personnel Hatch Containment 100'.

Answer	b	**Exam Level**	R	**Cognitive Level**	Memory	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Generic Knowledge and Abilities	**RO Group**	1	**SRO Group**	1				194001G315
GENERIC								**Record Number**	72

2.3	Radiation Control		
2.3.15	Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.	2.9	3.1

Explanation of Answer: 55.41.b(11) LOCA-1, step 16, checks for radiation outside containment by looking at 2R4, charging pump area, 2R41D plant vent process, 2R34, 1R3 Radio Chem lab area, 1R6A Sampling room, 1R20B counting room.

Reference Title

Loss of Reactor Coolant

S2.OP-SO.RM-0001

Learning Objectives

RMS000E006	NCT Identify and describe the local controls, indications, and alarms associated with the Radiation Monitoring System, including: The location of Radiation Monitoring System local controls and indications. (Licensed Operator & Non-licensed Operator only) The function of Radiation Monitoring System local controls and indications. (Licensed Operator & Non-licensed Operator only) The plant conditions or permissives required for Radiation Monitoring System local controls to perform their intended function. (Licensed Operator & Non-licensed Operator only)
LOCA01E007	Identify possible radioactivity release paths for a Loss of Coolant Accident, and describe how the actions of 2-EOP-LOCA-1 minimize the potential for a release
LOCA06E003	Identify possible radioactivity release paths for a LOCA OUTSIDE CONTAINMENT

Material Required for Examination

Question Source:	Facility Exam Bank	**Question Modification Method:**	Editorially Modified

Question Source Comments: Vision Q125697, replaced distracter with 2R10A (used to assist in determining LOCA is ocurring IN cont, not outside.

Given the following:

- The unit has been tripped and Safety Injection initiated due to a LOCA.
- The STA observes a PURPLE path displayed by SPDS for the CORE COOLING Status Tree, with no other RED or PURPLE paths on SPDS.
- The SMM is blinking dashes on both channels.
- SPDS is displaying question marks for all CET's.
- Plant Computer CET indication shows all CET's between 50-60°F.
- Local CET Display is unavailable.

Which of the following identifies how RCS saturation temperature will be determined IAW 2-EOP-CFST-1, Critical Safety Function Status Trees?

Wide Range RCS Thot and…

a. RCS pressure (PI-403 or PI-405) will be used in conjunction with Steam Tables.

b. RCS pressure (PI-403 or PI-405) will be used in conjunction with CFST Subcooling Tables.

c. PZR pressure channels (PI-455A, 456, 457 or 474A) will be used in conjunction with Steam Tables.

d. PZR pressure channels (PI-455A, 456, 457 or 474A) will be used in conjunction with CFST Subcooling Tables.

Answer	b	**Exam Level**	R	**Cognitive Level**	Memory	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Generic Knowledge and Abilities	**RO Group**	1	**SRO Group**	1				194001G403
GENERIC								**Record Number**	73
2.4	Emergency Procedures / Plan								
2.4.3	Ability to identify post-accident instrumentation.							3.7	3.9

Explanation of Answer: 55.41.b(7,10) EOP-CFST Basis Document, page 2 of 18, "The SMM should be used to determine RCS subcooling. If the SMM is inoperable, then calculate and log RCS subcooling on Table D. The value of T-Sat is obtained by using Table A for Normal Containment or Table B for Adverse Containment." Table D footnote states..."*RCS temperature- Use CET's (WR Thot RTD's if CET's are not available)."

Reference Title
Critical Safety Function Status Trees

Learning Objectives	
LOCA01E008	Determine the indications that are monitored to ensure proper system/component operation for each step in 2-EOP-LOCA-1

Material Required for Examination			
Question Source:	Facility Exam Bank	**Question Modification Method:**	Significantly Modified
Question Source Comments:	Vision Q81104. Used stem and expanded answer required from generic "what to do next" to how to determine RCS saturation temp.		

During performance of Emergency Operating Procedures (EOPs), some Cautions apply to the remainder of the EOP in progress.

Which of the following describes how these Continuous Cautions are identified?

a. Shaded box prior to the steps affected.

b. Shaded box prior to step 1 of the procedure.

c. Double-bordered box prior to the steps affected.

d. Single bordered box prior to step 1 of the procedure.

Answer	c	**Exam Level**	R	**Cognitive Level**	Memory	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Generic Knowledge and Abilities			**RO Group**	1	**SRO Group**	1		194001G419
GENERIC								**Record Number**	74

2.4	Emergency Procedures / Plan		
2.4.19	Knowledge of EOP layout, symbols, and icons.	3.4	4.1

Explanation of Answer: 55.41.b(10)4. "A Continuous Caution applies from the point at which it is encountered through the remainder of the EOP. The Continuous Caution symbol has double borders. The use of Continuous Cautions should be minimized."

Reference Title

EMERGENCY/ABNORMAL OPERATING PROCEDURE PROGRAM

Learning Objectives

TRP001E001 — Describe the EOP usage rules associated with the following in accordance with SC.OP-AP.ZZ-0102(Q):

A._Immediate Actions
B._Continuous Action Summaries
C._Communications
D._Log Keeping
E._Application of Notes and Cautions
F._Transitions
G._Adverse Containment

Material Required for Examination			
Question Source:	Facility Exam Bank	**Question Modification Method:**	Editorially Modified
Question Source Comments:	Vision Q81103 Replaced implausible distracter		

In which one of the following Unit 2 procedures are the Emergency Diesel Generator FIRE EMERGENCY BYPASS Keylock switches directed to be placed in BYPASS?

a. S2.OP-AB.FIRE-0002, Fire Damage Mitigation.

b. S2.OP-AB.CR-0001, Control Room Evacuation.

c. S2.OP-AB.FIRE-0001, Control Room Fire Response.

d. S2.OP-AB.CR-0002, Control Room Evacuation Due to Fire in the Control Room, Relay Room, 460/230V Switchgear Room, or 4KV Switchgear Room.

Answer	d	**Exam Level**	R	**Cognitive Level**	Memory	**Facility:**	Salem 1 & 2
Exam Date:	12/3/2012						
Tier:	Generic Knowledge and Abilities	**RO Group**	1	**SRO Group**	1		194001G427
GENERIC		**Record Number**	75				
2.4	Emergency Procedures / Plan						
2.4.27	Knowledge of "fire in the plant" procedures.	3.4	3.9				

Explanation of Answer

55.41.b(10) AB.CR-2, Attachment 4, Reactor Operator, pages 15, 19, and 22 place the keylock switches in bypass. The 3 distracters do not contain steps to perform this action. The bypass switches remove the SEC control from the EDG control, and are only operated when the control room has been evacuated due to a fire and SEC operation may be aberrant.

Reference Title

Control Room Evacuation Due to Fire in the Control Room, Relay Room, 460/230V Switchgear Room, or 4KV Switchgear Room.

Learning Objectives

ABCR02E003
a) Determine the appropriate abnormal procedure.
b) Describe the plant response to actions taken in the abnormal procedure.
c) Describe the final plant condition that is established by the abnormal procedure.

Material Required for Examination

Question Source: Facility Exam Bank

Question Modification Method: Direct From Source

Question Source Comments:

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

Applicant Information

Name:	Region: I
Date: 12/3/2012	Facility: Salem 1 & 2
License Level: SRO	Reactor Type: W
Start Time:	Finish Time:

Instructions

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. To pass the examination you must achieve a final grade of at least 80.00 percent overall, with 70.00 percent or better on the SRO-only items if given in conjunction with the RO exam; SRO-only exams given alone require a final grade of 80.00 percent to pass. You have 8 hours to complete the combined examination, and 3 hours if you are only taking the SRO portion.

Applicant Certification

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

____
Applicant's Signature

Results

RO/SRO-Only/Total Examination Values	____ / ____ / ____	Points
Applicant's Score	____ / ____ / ____	Points
Applicant's Grade	____ / ____ / ____	Percent

Senior Reactor Operator Answer Key

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

Given the following conditions:

- Unit 1 has performed a rapid downpower IAW S1.OP-AB.LOAD-0001, Rapid Load Reduction, due to severe Circulating Water System grassing.
- 3 Circulators remain in service, one on each "A" waterbox.
- With Rx power at 8%, the RO notices that over borating has caused RCS Tavg to drift below the Minimum Temperature for Criticality, and begins continuously withdrawing control rods to raise Tavg.
- As control rods continue to be withdrawn, the CRS notices Rx power has risen to 22% and continues to rise.

Which of the following identifies the proper response to this condition, and why?

Direct the RO to...

a. trip the Rx. Multiple RPS functions have failed to trip the Rx and adequate DNBR cannot be assured.

b. trip the Rx. A control grade interlock has failed to prevent approaching a Rx trip setpoint.

c. immediately stop rod motion. If RCS Tavg is >541°F, adjust Steam Dumps to lower Rx power.

d. immediately insert control rods while maintaining RCS Tavg >535°F.

Answer	b	**Exam Level**	S	**Cognitive Level**	Comprehension	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Emergency and Abnormal Plant Evolutions		**RO Group**	2	**SRO Group**	2			000001A203
001	Continuous Rod Withdrawal							**Record Number**	1
AA2.	Ability to determine and interpret the following as they apply to Continuous Rod Withdrawal:								
AA2.03	Proper actions to be taken if automatic safety functions have not taken place							4.5	4.8

Explanation of Answer

55.43(2) This was the precursor to the Salem April 1994 event, in which continuous rod withdrawl was performed trying to raise tavg afet overborating and rod insertion. In the stem, Rx rods are still withdrawing at 22% power, when Control Grade interlock C-1 should have acted at 20% power to block all outward rod movement. A is incorrect because it is not a multiple failure, either P-10 (10% Rx power) failed to reinstate the low power trips, or the C-1 failed to stop rod motion. C and D are incorrect in that as the Rx is rapidly approaching an automatic Rx trip setpoint, stopping rod motion and attempting to recove r the plant is non-conservative and incorrect.

Reference Title

RPS Nuclear Instrumentation System Permissives and Blocks

Learning Objectives

RXPROTE019 — Identify and describe the Control Room controls, indications, and alarms associated with the Reactor Protection System, including: (Licensed Operator and STA Only)
a) The Control Room location of Reactor Protection System control bezels and indications
b) The function of each Reactor Protection System Control Room control and indication
c) The effect each Reactor Protection System control has upon Reactor Protection System components and operation
d) The plant conditions or permissives required for Reactor Protection System Control Room controls to perform their intended function
e) The setpoints associated with the Reactor Protection System control room alarms

Material Required for Examination			
Question Source:	Facility Exam Bank	**Question Modification Method:**	Concept Used

Question Source Comments:

Given the following conditions:

- Unit 1 was operating at 100% power, EOL.
- A 70 gpm RCS leak in containment was identified.
- Operators tripped the Rx and initiated a Safety Injection IAW S1.OP-AB.RC-0001, Reactor Coolant System Leak.
- While performing EOP-TRIP-1, Rx Trip or Safety Injection, Step 23, PZR PORV STATUS, the RO reports that PZR PORV 2PR1 indicates open.
- The RO reports 2PR1 will not manually shut.
- The RO reports that 2PR6 PZR PORV Block Valve will not shut.

Which of the following identifies how the CRS should proceed?

a. Continue in TRIP-1. The combined size of the PZR PORV and RCS leaks will allow a transition to TRIP-3, Safety Injection Termination.

b. Continue in TRIP-1. A transition to LOCA-1, Loss of Reactor Coolant will be made based on rising containment radiation monitors.

c. Transition to LOCA-1. A transition to TRIP-3 will be made after PZR PORV status is checked again in LOCA-1.

d. Transition to LOCA-1. SI Termination criteria cannot be met with either PORV not fully shut.

Answer	c	Exam Level	S	Cognitive Level	Application	Facility:	Salem 1 & 2	Exam Date:	12/3/2012

Tier:	Emergency and Abnormal Plant Evolutions	RO Group	1	SRO Group	1	000007G120

007	Reactor Trip	Record Number	2

2.1	Conduct Of Operations

2.1.20	Ability to interpret and execute procedure steps.	4.6	4.6

Explanation of Answer

55.43.(5) A 100 gpm leak would require SI as stated in the stem, but charging pump flow would be able to maintain both PZR level and RCS pressure. If a PORV is open in TRIP-1 and it or its block valve cannot be shut, a transition is made to LOCA-1 because the EOP doesn't know how big the PORV leak is. Once in LOCA-1, there are steps redundant to those performed in TRIP-1 to attempt to close the PORV or its block valve, then it continues(there is no transition to another procedure based on PORV/block valve status.) Immediately after the PORV Status step in LOCA-1, SI Flow Reduction criteria are checked. All conditions should be met which are: Subcooling >0°F, AFW flow/Adequate SG NR level, RCS pressure stable or rising, PZR level >11%. With RCS pressure given in stem of 2235 psig and stable, this would indicate the PORV is only slightly open vs full open. A is incorrect because while a transition WILL be made to TRIP-3, it is not made in TRIP-1, it is made in LOCA-1. B is incorrect because while a transition to LOCA-1 might be made based on rising containment radiation levels, the transition must be made now. D is incorrect because the small size of the RCS leak combined with the small size of the PORV opening(as shown by normal RCS pressure), SI termination criteria can be met.

Reference Title
Rx Trip or Safety Injection
Loss of Reactor Coolant

Learning Objectives	
TRP001E009	select which (if any) transition should be made from a given procedure, in accordance with SC.OP-AP.ZZ-0102(Q)

Material Required for Examination	

Question Source:	New	Question Modification Method:	

Question Source Comments:

Given the following conditions:

- Unit 1 was tripped from 100% power when a SBLOCA occurred.
- While performing EOP-TRIP-1, 11 SG was identified as being faulted.
- All actions have been completed to isolate 11 SG.
- After transitioning to EOP-LOCA-1, the crew is performing the Faulted SG Evaluation steps.
- 11 SG pressure is 740 psig and lowering.
- 12-14 SG pressures are all 960 psig and lowering very slowly.

Which of the following identifies how the CRS should respond, and why?

a. Transition first to EOP-LOSC-1, then LOSC-2 because all SGs are now faulted. Faulted SGs require isolation because they may be masking other accidents (or their severity) in progress.

b. Continue in LOCA-1 since the ECCS injection is cooling the RCS and causing the unisolated SG pressures to lower. Going to LOSC-1 would only perform steps which have already been performed.

c. Continue in LOCA-1 since the ECCS injection is cooling the RCS and causing the unisolated SG pressures to lower. The additional subcooling provided will allow an earlier transition to TRIP-3 during SI Flow Reduction steps.

d. Transition directly to EOP-LOSC-2 because all SGs are now faulted. Steps to determine if SI termination can be performed will be adversely affected due to the lowering RCS pressure from the SG fault(s), and cause unecessary procedure performance.

Answer	b	Exam Level	S	Cognitive Level	Application	Facility:	Salem 1 & 2	Exam Date:	12/3/2012
Tier:	Emergency and Abnormal Plant Evolutions	RO Group	1	SRO Group	1				000009G244
009	Small Break LOCA							Record Number	3
2.2	Equipment Control								
2.2.44	Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.							4.2	4.4

Explanation of Answer

55.43(5) The procedure transition to LOSC-1 is made before LOCA-1 when in TRIP-1. The stem syas all actions have been completed to isolated the faulted SG. These actions would have to be performed in LOSC-1. The transition out of LOSC-1 is SGTR-1 if there is a rupture or to LOCA-1. The first step in LOCA-1 is to check for faulted SG's that have not been isolated. C is incorrect because RCS pressure will still be lowering (not stable or rising) based on the faulted SG. A is incorrect because the other SGs are not faulted, they are reacting to the cool ECCS water being pumped into the RCS, but the reason is plausible. D is incorrect because there is no direct transition to LOSC-2, you first have to enter LOSC-1.

Reference Title

Reactor Trip or Safety Injection

Loss of Reactor Coolant

Loss of Secondary Coolant

Learning Objectives

LOCA01E009	Describe the bases for each step, caution, note, and continuous action summary item in 2-EOP-LOCA-1
LOSC01E004	A. Determine a discrete path through the EOP. B. Determine an appropriate transition out of the EOP

Material Required for Examination			
Question Source:	New	Question Modification Method:	
Question Source Comments:			

Given the following conditions:

- Unit 2 is in MODE 5 after refueling the Rx.
- The Rx was shutdown 20 days ago.
- 21 RHR loop providing shutdown cooling, and 22 RHR loop aligned for ECCS.
- 21 RHR HX inlet temperature is 105°F and stable.
- RCS pressure is 205 psig.
- 21 RHR pump begins cavitating due to a valve being mispositioned during a tagging release.
- The CRS enters S2.OP-AB.RHR-0001, Loss of RHR, and stops 21 RHR pump.
- During performance of S2.OP-AB.RHR-0001, the crew has time for normal restoration and local venting of the RHR system.
- 1 hour after the initial loss of RHR, 22 RHR pump is started.
- The RO reports RHR flow is oscillating between 1,500-3,000 gpm.
- The highest CET temperature is 184°F.

Which of the following describes how the CRS should proceed?

a. Start 21 RHR pump and initiate Attachment 7, Hot Leg Injection.

b. Start 21 RHR pump and initiate Attachment 8, Cold Leg Injection.

c. Stop 22 RHR pump and initiate Attachment 7, Hot Leg Injection.

d. Stop 22 RHR pump and initiate Attachment 8, Cold Leg Injection.

Answer	d	Exam Level	S	Cognitive Level	Application	Facility:	Salem 1 & 2	Exam Date:	12/3/2012
Tier:	Emergency and Abnormal Plant Evolutions		RO Group	1	SRO Group	1			000025A205
025	Loss of Residual Heat Removal System							Record Number	4

AA2.	Ability to determine and interpret the following as they apply to Loss of Residual Heat Removal System:		
AA2.05	Limitations on LPI flow and temperature rates of change	3.1*	3.5*

Explanation of Answer

55.43(5) The conditions given in the stem indicate a RCS heatup rate of ~1.3 deg per minute. With the initial temp of 105, this will add 78 degrees and when the RHR pump is started RCS temp will be 183°. RHR flow is required to be stable between 1800-3000 after the venting and starting of the RHR pump to allow exiting the procedure with all other RHR system parameters normal.(page 12) With 1,500 gpm flow swings, this will be answered NO. Step 3.30 states to stop any running RHR pump. Step 3.32 (page 18)states to initiate an alternate method of Decay Heat Removal. With CET temps <200°F, the preferred method is Attachment 8 Cold Leg injection. Attachment 7 Hot leg injection is not preferred due to RCS being intact and <200°F. With the procedure stating to stop all RHR pumps, starting 21 is not directed.

Reference Title

Loss of RHR

Learning Objectives

ABRHR1E004	Describe, in general terms, the actions taken in S2.OP-AB.RHR-0001 and the bases for the actions in accordance with the Technical Bases Document.

Material Required for Examination			
Question Source:	Facility Exam Bank	Question Modification Method:	Concept Used
Question Source Comments:	Vision Q80328 concept used, expanded to make SRO level. Changed 2 distracters to start idle RHR pump.		

Given the following conditions:

- Unit 2 is operating at 100% power.
- The RO reports that the PZR Cold Cal level channel (LI-462) is indicating 0%.

Which of the following describes how the CRS should respond?

a. Enter TSAS 3.3.1.1 Reactor Trip System Instrumentation. This Tech Spec is based on being able to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. Place a single piece of red translucent tape across LI-462.

b. Enter TSAS 3.3.3.7 Accident Monitoring Instrumentation. This Tech Spec is based on ensuring sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. Place an INFO sticker on LI-462.

c. No Tech Spec entry will be made since TSAS 3.3.1.1 and TSAS 3.3.3.7 are applicable to this instrument ONLY in MODES 4-6 and during movement of irradiated fuel. Place a single piece of red translucent tape across LI-462.

d. No Tech Spec entry will be made since TSAS 3.3.1.1 and TSAS 3.3.3.7 are never applicable to this instrument. Place an INFO sticker on LI-462.

Answer	d	**Exam Level**	S	**Cognitive Level**	Memory	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Emergency and Abnormal Plant Evolutions			**RO Group**	2	**SRO Group**	2		000028G243
028	Pressurizer Level Control Malfunction							**Record Number**	5
2.2	Equipment Control								
2.2.43	Knowledge of the process used track inoperable alarms.							3.0	3.3

Explanation of Answer: 55.43(2)(5) The PZR cold cal channel is not included in the Rx Trip Instrumentation or the Accident Monitoring Tech Specs. It is used when RCS temperature is <200°F as directed in S2.OP-IO.ZZ-0006, Hot Standby to Cold Shutdown. A is incorrect because TS 3.3.1.1 is not entered, although the Bases is correct. Also, the red translucent tape is used to identify an inoperable alarm.

Reference Title
Salem Tech Specs
Control Room Instrumentation and Alarms
Hot Standby to Cold Shutdown

Learning Objectives	
PZRP&LE010	Given a situation dealing with Pressurizer Pressure and Level Control System operability, examine the situation and apply the appropriate Technical Specification action. (License Operator and STA only) NCT State the Technical Specification associated with the component, parameters and operation of the Pressurizer Pressure and Level Control System including the Limiting Condition for Operation(s) (LCO) and the applicability of the LCO(s) (Non-licensed Operator)

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following conditions:

- Unit 1 is operating at 100% power.
- The RO reports the following:
  - PZR level slowly lowering.
  - Charging flow slowly rising.
  - PZR pressure 2235 psig and stable.
  - Condenser radiation monitor R15 in WARNING and rising.

Which of the following identifies the procedure which will provide the most effective mitigating actions for these conditions, and why?

a. S2.OP-AB.SG-0001, Steam Generator Tube Leak. Actions provided in procedure will reduce the spread of contamination.

b. S2.OP-AB.RC-0001, Reactor Coolant System Leak. This procedure checks in a prioritized manner for all sources of RCS leakage.

c. S2.OP-AB.SG-0001, Steam Generator Tube Leak. Actions provided in procedure will ensure that that any affected SG tube will not rupture.

d. S2.OP-AB.RC-0001, Reactor Coolant System Leak. This procedure checks first to ensure the RCS leakage is controllable with normal CVCS make-up capability and provides direction for initiating Safety Injection directly if it is not.

Answer	a	**Exam Level**	S	**Cognitive Level**	Application	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Emergency and Abnormal Plant Evolutions	**RO Group**	2	**SRO Group**	2				000037A204
037	Steam Generator Tube Leak							**Record Number**	6

AA2.	Ability to determine and interpret the following as they apply to Steam Generator Tube Leak:		
AA2.04	Comparison of RCS fluid inputs and outputs, to detect leaks	3.4	3.7

Explanation of Answer

55.43(5) C is incorrect because while the actions to shutdown the reactor and cooldown/depressurize will lessen the likelihood that the tube will not rupture, it does not ensure that it will not. A is correct because after the procedure checks that PZR level can be maintained stable or rising, actions are taken to minimize the spread of contamination by isolating the SG to the effect that it can be isolated. AB.RC is not the most effective procedure because it will require additional time to get through the procedure before being directed to go to AB.SG, lengthening the time before actions are taken in AB.SG to minimize the spread of contamination.

Reference Title
Steam Generator Tube Leak
Reactor Coolant System Leak

Learning Objectives	
ABSG01E004	a) Determine the appropriate abnormal procedure. b) Describe the plant response to actions taken in the abnormal procedure. c) Describe the final plant condition that is established by the abnormal procedure.

Material Required for Examination			
Question Source:	Facility Exam Bank	**Question Modification Method:**	Significantly Modified
Question Source Comments:	Vision Q39610. Originally had 4 procedure choices with no why part. Changed to 2 and 2 with why incorporated.		

Given the following conditions:

- Unit 1 is operating at 100% power.
- 12 SG NR Channel I has been removed from service while undergoing a Channel Calibration IAW S1.IC-CC.RCP-0045, 1LT-529 #12 Steam Generator Level Protection Channel I.
- 12 SG NR Channel IV fails high.
- Control rods begin stepping in at 72 spm.

Which of the following describes how the CRS should respond?

a. Enter EOP-TRIP-1 Reactor Trip or Safety Injection, and attempt to trip the Rx from the control room. If unsuccessful, enter FRSM-1 Response to Nuclear Power Generation and dispatch an operator to locally trip the Rx.

b. Enter EOP-TRIP-1 and attempt to trip the Rx and Main Turbine from the control room. If unsuccessful, dispatch an operator to locally trip the Main Turbine, then enter FRSM-1 and dispatch an operator to locally trip the Rx.

c. Enter FRSM-1 directly, and ensure that the Main Turbine is tripped. AMSAC will start ONLY the MDAFW pumps when SG NR level is < 5% for > 25 seconds.

d. Enter FRSM-1 directly, and ensure AFW flow >44E4 lbm/hr is established. If not already running, AMSAC will start ALL AFW pumps when SG NR level is < 13% for > 25 seconds.

Answer	a	Exam Level	S	Cognitive Level	Application	Facility:	Salem 1 & 2	Exam Date:	12/3/2012
Tier:	Emergency and Abnormal Plant Evolutions	RO Group	1	SRO Group	1				000054A205
054	Loss of Main Feedwater							Record Number	7
AA2.	Ability to determine and interpret the following as they apply to Loss of Main Feedwater:								
AA2.05	Status of MFW pumps, regulating and stop valves							3.5	3.7

Explanation of Answer: A P-14 signal is generated by 2/3 NR level channels on 12 SG being >67%, The P-14 signal trips the Main Turbine (which is indicated by 72 spm rod insertion) trips the Main Feed pumps and shuts the BF19s and 40s and 13s. (FW Isolation signal) The reactor should have tripped on the Main Turbine trip, but has not as evidenced by control rods stepping in on the Main Turbine load reject vs being on the bottom. The FRSM distracters are both incorrect because FRSM-1 is not entered directly, even though the actions of D are correct with the correct setpoints. C has incorrect action and setpoint. B is incorrect because the Main Turbine has already tripped.

Reference Title
Reactor Trip or Safety Injection
Response to Nuclear Power Generation

Learning Objectives	
TRP001E009	select which (if any) transition should be made from a given procedure, in accordance with SC.OP-AP.ZZ-0102(Q)
TRP001E011	Identify Entry Conditions for EOP-TRIP-1, in accordance with EOP-TRIP-1.

Material Required for Examination			
Question Source:	New	Question Modification Method:	
Question Source Comments:			

Given the following conditions:

- Unit 2 is at 40% power performing a shutdown.
- 4 SW Bay is isolated due to a leak on the 25SW3, 25 SW Pump Discharge Isolation Valve.
- Operators are performing the shutdown to comply with TSAS 3.7.4 because difficulties arose during the leak repair of the 25SW3.
- 2A EDG is supplying 2A 4KV vital bus for a scheduled surveillance.
- 21 and 23 SW pumps are in service.

Which of the following describes how the control room crew will respond if 2A EDG output breaker trips on Bus Differential, and 23 SW pump trips 1 minute later on over current when supplying SW flow at runout conditions?

a. Trip the Main Turbine to reduce Rx power and heat input to the RCS, and enter S2.OP-AB.TRB-0001, Turbine Trip <P-9.

b. Enter S2.OP-AB.SW-0001, Loss of Service Water Header Pressure. Trip the Main Turbine, and reduce Rx power <5% in order to place AFW in service.

c. Enter S2.OP-AB.SW-0005, Loss of All Service Water. Trip the Rx, confirm the trip, and stop RCPs to limit the heat input to the CCW system, and preserve RCP seal packages.

d. Trip the Rx and go to TRIP-1, Reactor Trip or Safety Injection. After exiting TRIP-2, Reactor Trip Response, enter S2.OP-AB.SW-0005 to perform compensatory actions for no service water pumps operating.

Answer	c	Exam Level	S	Cognitive Level	Application	Facility:	Salem 1 & 2	Exam Date:	12/3/2012
Tier:	Emergency and Abnormal Plant Evolutions	RO Group	1	SRO Group	1				000062G406
062	Loss of Nuclear Service Water							Record Number	8
2.4	Emergency Procedures / Plan								
2.4.6	Knowledge of EOP mitigation strategies.							3.7	4.7

Explanation of Answer

55.43(5) A is incorrect because even if the actions are (some of those) performed in AB.SW-5, the next procedure entry to AB.TRB is incorrect since the Rx is tripped in AB.SW-5. B is incorrect because AB.SW-1 doesn't perform those actions. C is correct. D is incorrect because AB.SW-5 should be entered before exiting the TRIP series.

Reference Title
Loss of All Service Water
Loss of Service Water Header Pressure

Learning Objectives	
ABSW04E005	Given a set of initial plant conditions: A. Determine the appropriate abnormal procedure. B. Describe the plant response to actions taken in the abnormal procedure C. Describe the final plant condition that is established by the abnormal procedure

Material Required for Examination			
Question Source:	Facility Exam Bank	Question Modification Method:	Editorially Modified
Question Source Comments:	Vision 88855. Added procedure entry.		

Given the following conditions:

- Unit 2 has experienced a MSLB at the Mixing Bottle.
- All attempts at MSLI have failed, and 21-24MS167s remain open.
- Operators have just completed SI termination steps in EOP-LOSC-2, Multiple Steam Generator Depressurization, and PZR level is being maintained stable.
- AFW flow to each SG is 1.0E4 lbm/hr.
- The RO reports rising pressure in 22 SG.

Which of the following describes how the CRS should proceed, and why?

a. Transition to EOP-LOSC-1 and stop RCPs if RCS pressure is <1350 psig, since RCPs cannot be stopped in LOSC-2.

b. Transition to EOP-LOSC-1, Loss of Secondary Coolant, since one SG is now available for subsequent recovery actions.

c. Remain in EOP-LOSC-2 until positive control can be established over the cooldown after the remaining Steam Generators have fully depressurized, then transition to EOP-LOSC-1.

d. Remain in EOP-LOSC-2 since returning to EOP-LOSC-1 will require a transition to EOP-LOCA-1 upon completion and will complicate recovery after the remaining SGs have fully depressurized.

Answer	b	**Exam Level**	S	**Cognitive Level**	Memory	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Emergency and Abnormal Plant Evolutions	**RO Group**	1	**SRO Group**	1				00WE12A201
E12	Uncontrolled Depressurization of all Steam Generators							**Record Number**	9

EA2.	Ability to determine and interpret the following as they apply to Uncontrolled Depressurization of all Steam Generators:		
EA2.1	Facility conditions and selection of appropriate procedures during abnormal and emergency operations.	3.2	4.0

Explanation of Answer: 55.43(5) LOSC-2 CAS states that upon a pressure rise in any SG except when performing SI termination in Steps 8-20, GO TO EOP- LOSC-1. The stem states that it is after Step 20. LOSC-1 Basis Document, page 7, states that.."Any cooldown operations that are performed as subsequent recovery actions will require at least one nonfaulted SG."

Reference Title

Multiple Steam Generator Depressurization

Loss of Secondary Coolant

Learning Objectives

LOSC02E005	A. Determine a discrete path through the EOP. B. Determine an appropriate transition out of the EOP

Material Required for Examination

Question Source:	Facility Exam Bank	**Question Modification Method:**	Concept Used
Question Source Comments:	Vision Q57956 concept used, and added the "why" to question.		

Given the following conditions:

- Salem Unit 2 was performing a Rx shutdown due to indications of failed fuel after chemistry reported reactor coolant activity to be 500 uCi/gm dose equivalent I-131.
- During the shutdown, the RO reports lowering PZR pressure and level.
- With 21 charging pump in service, PZR level continues to rapidly lower, and the RO reports containment pressure is also rising.
- The RO trips the Rx and initiates a Safety Injection, and all equipment responds as expected for the SI.
- The SM declares a Site Area Emergency.
- RCS pressure continues to lower, and 30 minutes later is 35 psig.

Which of the following identifies a condition which would require notification of the NRC, and the correct time for that notification?

a. The wind direction shifts from 0° to 180°. 15 minutes.

b. Containment radiation level exceeds 2,000 R / hr. 60 minutes.

c. Containment sump level indicated on 2CC1 has remained stable at 46%. 15 minutes.

d. The control room must be evacuated and operators cannot access the Auxiliary Building due to high radiation. 60 minutes.

Answer	b	**Exam Level**	S	**Cognitive Level**	Application	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Emergency and Abnormal Plant Evolutions	**RO Group**	2	**SRO Group**	2				00WE16G430
E16	High Containment Radiation							**Record Number**	10
2.4	Emergency Procedures / Plan								
2.4.30	Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator.							2.7	4.1

Explanation of Answer

55.43(1,2) The 15 minute times in the question are from the 15 minute notifications required during emergencies to the states. The NRC is not required to be notified for 60 minutes. The second knowledge part of the question is what would cause a notification to be required, i.e. a more sever E plan classification is made. The radiation levels > 2,000 R/hr adds 2 points from the containment barrier. The stem states that the SM declared a SAE, which would have been under FB4.L and RB2.L each of which is 5 points. The 2 additional points would put the unit in a GE. The wind shift while in a would not require a notification because no PAR would have been made for the SAE. The containment sump level is NOT expected, with containment pressure at 35 psig, the entire contents of the RCS are on the floor and level would have risen, as is seen for LBLOCAs. The 15 minute time for this condition is wrong, however. The CR evac and inability to establish control of the plant in 15 minutes is a SAE, and the plant is already in a SAE.

Reference Title

Salem ECG

Learning Objectives

Material Required for Examination	SRO 10 Salem ECG
Question Source: New	**Question Modification Method:**

Question Source Comments:

Given the following condition:

- Unit 1 is in MODE 5 during a plant startup.
- 11 RHR loop is in service.
- 12 RHR loop is aligned for ECCS.
- 13 RCP is in service.
- 11 charging pump is in service.
- RCS Tavg is 175°.
- RCS pressure is 310 psig.
- PZR level is 60%.

When placing the second RCP in service, RCS pressure momentarily rises to 390 psig.

Which of the following describes the RHR system response, and how the CRS should proceed?

a. The 1RH3, RHR SAF RLF VLV TO CONTAINMENT SUMP opens. Enter S2.OP-AB.PZR-0001, PZR Pressure Malfunction, and ensure that any PZR PORV that opened in response to the RCS pressure has shut.

b. The 1RH2, RHR COMMON SUCT MOV automatically shuts. Enter S2.OP-AB.PZR-0001 and ensure that any PZR PORV that opened in response to the RCS pressure has shut.

c. The 1RH3 opens. Enter S2.OP-AB.LOCA-0001, Shutdown LOCA, and isolate letdown to minimize RCS inventory loss.

d. The 1RH2 automatically shuts. Enter S2.OP-AB.LOCA-0001 and isolate letdown to minimize RCS inventory loss.

Answer	a	Exam Level	S	Cognitive Level	Application	Facility:	Salem 1 & 2	Exam Date:	12/3/2012
Tier:	Plant Systems	RO Group	1	SRO Group	1				005000A202
005	Residual Heat Removal System							Record Number	11

A2.	Ability to (a) predict the impacts of the following on the Residual Heat Removal System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:		
A2.02	Pressure transient protection during cold shutdown	3.5	3.7

Explanation of Answer

55.43(5) With RHR in service both the PZR PORVs and the 1RH3 will open at their 375 psig setpoints. The RH3 opening will not be apprent to the control room, but the PORV opening will. AB.PZR, Attachment 3, will ensure the PORV has shut. The 1RH2 has an OPENING interlock that requires RCS pressure to be <375 psig, then a keyswitch opens the valve. There is no automatic closure associated with this valve on high pressure. AB.LOCA is used in MODE 3 and MODE 4 with the accumulators isolated, and with the unit in MODE 5 as described in the stem, would not be entered.

Reference Title

PZR Pressure Malfunction

Shutdown LOCA

Residual Heat Removal

Learning Objectives

RHR000E004 LOR NCT Describe the function of the following components and how their normal and abnormal operation affects the Residual Heat Removal System:
a) RHR Pumps
b) Refueling Water Storage Tank
c) Heat Exchangers
d) Motor Operated Valves
   i) RH1 and RH2, Inlet Isolation Valves
   ii) RH4, Pump Suction Isolation Valves

iii) SJ44, Containment Sump Isolation Valves
iv) SJ69, RWST to RHR Suction
v) RH29, Miniflow Recirc. Valves
vi) RH19, Loop Isolation Valves
vii) SJ45, RHR to SI or Charging/SI Pump Suction
viii) SJ113, CCP-SIP Suction Cross-Connect Valves
ix) SJ49 Outlet Isolation Valve
x) RH26, RHR Hot Leg Isolation
xi) CS36, Spray Recirculation from RHR Valve

e) Air-Operated Valves
i) RH18, RHR HX Outlet Valves
ii) RH20, RHR HX Bypass Valve

f) Other System Valves
i) RH3, RCS to RHR Inlet Relief Valve
ii) RH25, RHR to RCS Hot Leg Relief Valve
iii) SJ48, RHR to RCS Cold Leg Relief Valves
iv) RH12, RHR HX Bypass Valve
v) RH17, RHR to CVCS Letdown
vi) RH21, RHR to RWST

g) Containment Sump Anti-Vortex Baffle
h) Orifices

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Given the following conditions:

- Unit 2 is operating at 9% power, performing a power ascension prior to rolling the Main Turbine.
- OHA F-17 IR FLUX HI is received in the control room.
- The Bistable for 2N36 on 2RP4 is lit.
- The reactor remains at power.

Which choice identifies what has happened, and what action(s) is/are required to be performed?

a. An ATWT has occurred, attempt to trip the reactor manually. Verify the turbine is tripped, initiate rod insertion and go to FRSM-1 if the reactor does NOT trip.

b. This alarm is expected at ~10% power. BLOCK both intermediate range channels by depressing the BLOCK INTERMEDIATE RANGE A and B PBs IAW OHA F-17 ARP.

c. A failure of the IR high flux trip block has occurred. Lower Rx power to less than 5% and depress BLOCK INTERMEDIATE RANGE B pushbutton to block Train B IAW S2.OP-IO.ZZ-0004 POWER OPERATION.

d. This alarm is NOT expected for this power level. The reactor is NOT expected to trip since the IR High Flux trip setpoint is 25% current equivalent power. Place the power ascension on hold IAW S2.OP-IO.ZZ-0003 HOTSTANDBY TO MINIMUM LOAD.

Answer	a	Exam Level	S	Cognitive Level	Application	Facility:	Salem 1 & 2	Exam Date:	12/3/2012
Tier:	Plant Systems	RO Group	1	SRO Group	1				012000A201
012	Reactor Protection System							Record Number	12

A2.	Ability to (a) predict the impacts of the following on the Reactor Protection System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:		
A2.01	Faulty bistable operation	3.1	3.6

Explanation of Answer

55.43(5)B is correct because with the RP4 coincidence made up, there is a valid demand for a reactor trip and an ATWT exists. The actions are correct for an ATWT IAW EOP-TRIP-1. Distracter C is incorrect because at 9% power, the IR block cannot have been blocked, and failure of the block would prevent a Rx trip. In any event, the Rx would be tripped not power lowered. Distracter B is incorrect because the alarm is NOT expected. Distracter D is incorrect because the reactor has a trip signal demanded.

Reference Title
Reactor Trip or Safety Injection
Overhead Annunciator Window F

Learning Objectives	
RXPROTE012	LOR State the setpoints, coincidence, blocks and permissives for all Reactor Trips and Safety Injections actuations (Licensed Operator and STA Only) NCT List all Reactor Trips and Safety Injections (Non-Licensed Operator)

Material Required for Examination			
Question Source:	Facility Exam Bank	Question Modification Method:	Direct From Source
Question Source Comments:	Vision Q80969 Used on "H" Salem NRC SRO Exam (4 NRC exams ago.)		

Given the following conditions:

- Unit 2 is at 100% power.
- "LOSS OF TRIPPING CAPABILITY" Alarm is received for "A" Reactor Trip Breaker (RTB).

Which of the following describes the effect on RTB "A" from this condition and how should the CRS proceed?

Reactor Trip Breaker A....

a. will NOT trip from an Automatic Safety Injection signal since the UV coil is unavailable. Restore UV coil capability within 48 hours or be in Hot Standby within the next six hours.

b. will trip from ALL Reactor Trip initiation signals EXCEPT a Manual Safety Injection since the Shunt Coil cannot be energized. Restore Shunt Trip capability within 48 hours or be in Hot Standby within the next six hours.

c. will NOT trip from an Automatic Safety Injection signal since the UV coil is unavailable. Place Reactor Trip Bypass Breaker A in service and open Reactor Trip Breaker A within one hour or be in Hot Standby within the next six hours.

d. will trip from ALL Reactor Trip initiation signals EXCEPT a Manual Safety Injection since the Shunt Coil cannot be energized. Place Reactor Trip Bypass Breaker A in service and open Reactor Trip Breaker A within one hour or be in Hot Stanby within the next six hours.

Answer	b	**Exam Level**	S	**Cognitive Level**	Memory	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Plant Systems	**RO Group**	1	**SRO Group**	1				013000A205
013	Engineered Safety Features Actuation System							**Record Number**	13

A2.	Ability to (a) predict the impacts of the following on the Engineered Safety Features Actuation System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:		
A2.05	Loss of dc control power	3.7	4.2

Explanation of Answer

55.43(2,5) The RTB Shunt Coils are energized to trip. The RTBs also have a deenergize to actuate UV coil. Both of these open the RTB. The Loss of Tripping Capability alarm indicates that 125VDC power has been lost to the Shunt Coil. As shown on drawing 221051, the only Reactor Trip which goes SOLELY to the Shunt trip coil is a Manual Safety Injection. A is incorrect because an auto SI will trip the Rx, but the action is correct. C is incorrect because of A above and the action is incorrect. B is correct because of above and TSAS 3.3.1.1, Table 3.3-1, Functional Unit 21, page 3/4 3-4, in Modes 1 and 2, directs Action 14. Action 14 states that with either the UV or shunt trip unavailable, restore it to Operable within 48 hours or be in HSB within 6 hours.

Reference Title

Salem Tech Specs

Control Console 2CC2

Learning Objectives

ESF000E021	State the setpoints for automatic actuations associated with the Engineered Safety Features

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments: Vision Q77831. Added Tech Spec part to make SRO level

Given the following conditions:

- Unit 1 is performing a Reactor startup IAW S1.OP-IO.ZZ-0003 Hot Standby to Minimum Load.
- All Shutdown Bank control rods have been fully withdrawn.
- Control Bank A is fully withdrawn.
- As Control Bank B is withdraws past 20 steps, the RO reports OHA E-48, ROD BOTTOM has just alarmed and remains locked in.
- No other alarms are received.

Which of the following describes how the system is operating, and how the CRS should proceed?

a. The Rod Bottom Bistable causes OHA E-48 to alarm as each control bank is withdrawn past 20 steps and is expected. The CRS should direct the reset of the Non-Urgent Failure to reset the alarm, then continue the startup.

b. The Rod Bottom Bistable causes OHA E-48 to alarm as each control bank is withdrawn past 20 steps and is expected. The CRS should direct the RO to depress the STARTUP pushbutton on 1CC2 to reset the alarm, and continue the startup.

c. The Rod Bottom Bistable cleared when Control Bank A was withdrawn past 20 steps and OHA E-48 is unexpected at this time. The CRS should enter S1.OP-AB.ROD-0002 Dropped Rod and direct the opening of the Reactor Trip Breakers to terminate the Rx startup.

d. The Rod Bottom Bistable cleared when Control Bank A was withdrawn past 20 steps and OHA E-48 is unexpected at this time. The CRS should place the startup on hold and initiate S1.OP-AB.ROD-0002, Dropped Rod, or S1.OP-AB.ROD-0004, Rod Position Indication Failure, to determine what malfunction has occurred.

Answer	d	**Exam Level**	S	**Cognitive Level**	Comprehension	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Plant Systems		**RO Group**	2	**SRO Group**	2			016000G237
016	Non-Nuclear Instrumentation System						**Record Number**		14
2.2	Equipment Control								
2.2.37	Ability to determine operability and/or availability of safety related equipment.							3.6	4.6

Explanation of Answer

The Rod Bottom alarm CLEARS when CB A is withdrawn past 20 steps. This is because a Rod Bottom Bistable Bypass for each of the other three control banks B,C,D bypass the alarm for their respective bank when all rods in that group are below 35 steps. This means that the alarm was CLEAR when it alarmed, it did not reflash. Since no other alarms occurred, the CRS should place the startup on hold and enter AB.ROD-2 (which will direct entry into AB.ROD-4) or enter AB.ROD-4 directly to investigate the failure. There is a 4 hour window for having to terminate the startup (IOP-3, step 5.2.19) so opening the trip breakers is not required.

Reference Title

Rod Control and Position Indicating Systems Lesson Plan

Dropped Rod

Rod Position Indication Failure

Learning Objectives

RODS00E006 — NCT Describe the function of the following components and how their normal and abnormal operation affects the Rod Control and Position Indication Systems:
Rod Cluster Control Assembly (RCCA)
Control Rod Drive Mechanism (CRDM)
Rod Drive MG Sets
Reactor Trip and Trip Bypass breakers
Reactor Control Unit
Power Cabinets

Logic Cabinet components: Pulser Master Cycler Slave Cyclers Bank Overlap Unit h. DC Hold Cabinet i. Rod Position Indicator (RPI) Coils j. Signal Conditioning Modules k. PulseûtoûAnalog (P to A) Converters l. Rod Bottom Bistables m. Rod Insertion Limit Comparator n. Step Counters

Material Required for Examination	

Question Source:	Facility Exam Bank	**Question Modification Method:**	Concept Used

Question Source Comments:	Vision Q60249 expanded from rod bottom bistable question to whether alarm is expected and what CRS should do.

Given the following conditions:

- Unit 2 is operating at 100% power.
- Technicians are performing a sensor calibration of Containment Pressure Channel IV IAW S2.IC-SC.RCP-0066, 2PT-948A CONTAINMENT PRESSURE PROTECTION CHANNEL IV.
- All bistables and test switches are in their proper alignment for the calibration.
- While I&C Technicians are performing the calibration, the control room recieves OHA C-16, PHASE B CNTMT ISOL ACT.

Which of the following identifies what has happened, and how the CRS should respond to this alarm?

a. Phase B isolation valves will shut ONLY. The isolation will not be able to be reset until I&C returns their bistables and test switches to normal. Trip the Rx, stop all RCPs, and GO TO EOP-TRIP-1.

b. Phase B isolation valves will shut and Containment Spray valves will open. Attempt to reset and open the Phase B isolation valves. If unable to reset Phase B, GO TO S2.OP-AB.RCP-0001, RCP Abnormality.

c. Phase B isolation valves will shut, Containment Spray valves will open, and both Containment Spray pumps will start. Attempt to reset and open the Phase B isolation valves. If unable to reset Phase B, trip the Rx and GO TO EOP-TRIP-1 Reactor Trip or Safety Injection.

d. This is an expected alarm during performance of the test because the Channel IV Containment Spray bistables were previoously tripped, and the Phase B Isolation signal is not expected to cause any components to change state. Verify Phase B isolation valves and Containment Spray valves and pumps have not been affected.

Answer	b	Exam Level	S	Cognitive Level	Application	Facility:	Salem 1 & 2	Exam Date:	12/3/2012
Tier:	Plant Systems	RO Group	1	SRO Group	1				026000G450
026	Containment Spray System							Record Number	15
2.4	Emergency Procedures / Plan								
2.4.50	Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.							4.2	4.0

Explanation of Answer

55.43(5) This alarm is not expected to occur during the sensor cal, as it requires 2/4 containment pressure channels to see 15 psig. The ARP says if cont pressure is <15 psig, attempt to reset and open Phase B isoaltion valves, and if unsuccessful go to AB.RCP based on the loss of CCW to the RCPs. The CS valves will reposition, but the CS pumps will not start.

Reference Title

Overhead Annunciators Window C

Learning Objectives

CSPRAYE008 — Identify and describe the Control Room controls, indications, and alarms associated with the Containment Spray System, including:
The Control Room location of Containment Spray System control bezels and indications. (Licensed Operator & STA only)
The function of each Containment Spray System Control Room control and indication. (Licensed Operator & STA only)
The effect each Containment Spray System control has upon Containment Spray System components and operation. (Licensed Operator & STA only)
The plant conditions or permissives required for Containment Spray System Control Room controls to perform their intended function. (Licensed Operator & STA only)
The setpoints associated with the Containment Spray System control room alarms. (Licensed Operator & STA only)

Material Required for Examination	

Question Source:	New	**Question Modification Method:**	

Question Source Comments:	

Given the following conditions:

- Fuel handling is in progress in the Unit 2 Spent Fuel Pool when a fuel assembly in the Spent Fuel Handling Tool is dropped.
- Gas bubbles are observed in the vicinity of the dropped fuel assembly.
- 2R5, Fuel Handling Building radiation monitor, goes into alarm.

Which of the following describes the effect of this event, and contains actions that will be performed IAW S2.OP-AB.FUEL-0001, Fuel Handling Incident?

a. Auxiliary Building ventilation automatically swaps to place the Charcoal Filter in service to prevent a release to the environment. Ensure the FHB Supply Fan is running.

b. ALL Fuel Handling Crane motion except downward movement is locked out to prevent raising a damaged fuel assembly. Ensure the FHB Truck Bay Roll Up Door is closed.

c. Containment Ventilation Isolation activates to ensure any containment pressure/vacuum relief in service is terminated. Ensure the Fuel Transfer Cart is NOT at the Spent Fuel Pool and close the Gate Valve.

d. Fuel Handling Building ventilation automatically swaps to place the Charcoal Filter in service to prevent a release to the environment. Ensure the FHB Watertight Door remains closed except for normal personnel passage.

Answer	d	**Exam Level**	S	**Cognitive Level**	Memory	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Plant Systems		**RO Group**	2	**SRO Group**	2			034000A201
034	Fuel Handling Equipment System							**Record Number**	16

A2.	Ability to (a) predict the impacts of the following on the Fuel Handling Equipment System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:		
A2.01	Dropped fuel element	3.6	4.4

Explanation of Answer

55.43(7) The 2R5 in alarm swaps the FHB exhaust ventilation to the Charcoal Filter and starts both FHB Exhaust Fans. The normal configuration for FHB ventilation is the single Supply Fan running, and BOTH Exhaust Fans running. A is incorrect because ABV does not auto swap to charcoal filter upon a 2R5 high alarm, nor does the Supply fan get an auto start signal. B is incorrect because while the FH crane only locks out as described with the 2R32A rad monitor on the crane itself goes into alarm, not the 2R5 area monitor, and the action is correct. C is incorrect but plausible because CVI does not actuate, but FHV does discharge to the plant vent. The action in D is correct.

Reference Title

Fuel Handling Incident

Learning Objectives

ABFUEL01E00 2	Describe, in general terms, the actions taken in S2.OP-AB.FUEL-0001(Q) and the bases for the actions.

Material Required for Examination

Question Source:	Facility Exam Bank	**Question Modification Method:**	Concept Used
Question Source Comments:	Vision Q109317. Added why and action to be performed in AB.Fuel-2.		

Given the following conditions:

- Unit 2 is in MODE 3.
- Welding in the Turbine Building has caused an actual deluge actuation to occur in the Turbine Building.
- The control room recieves the following alarms:
  - OHAs A-7 FIRE PROT FIRE
  - OHA A-15 FIRE PUMP 1/2 RUN
  - Coded Fire alarm 2-2-1 TURBINE GEN AREA -88' ELEV

Which of the following identifies how the Fire Protection system has responded, and how should the CRS proceed?

a. ONLY one diesel fire pump has started. Enter S2.OP-AB.FIRE-0001, Control Room Fire Response. Place BOTH Unit 1 and Unit 2 CAV in Fire Outside Control Area.

b. ONLY one diesel fire pump has started. Enter S2.OP-AB.FP-0001, Fire Protection System Malfunction. A Unit shutdown will be required due to the current capability of the Fire Protection system being degraded.

c. BOTH diesel fire pumps have started. Enter S2.OP-AB.FIRE-0001, Control Room Fire Response. Place BOTH Unit 1 and Unit 2 CAV in Fire Outside Control Area.

d. BOTH diesel fire pumps have started. Enter S2.OP-AB.FP-0001, Fire Protection System Malfunction. A Unit shutdown will be required due to the current capability of the Fire Protection system being degraded.

Answer	a	**Exam Level**	S	**Cognitive Level**	Application	**Facility:** Salem 1 & 2	**Exam Date:** 12/3/2012
Tier:	Plant Systems	**RO Group**	2	**SRO Group**	2		086000A203
086	Fire Protection System					**Record Number**	17

A2.	Ability to (a) predict the impacts of the following on the Fire Protection System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:		
A2.03	Inadvertent actuation of the FPS due to circuit failure or welding	2.7	2.9

Explanation of Answer

55.43(5) A deluge valve opening as stated in stem will cause FP system header pressure to lower to the point that #1 Fire pump will start at 85 psig, and will restore header pressure. Each Fire Pump is rated to supply all fire protection needs. The second Fire Pump will NOT start, as its auto start pressure is set at 75 psig. When the deluge occurs, the CRS will not know it is inadvertent. The CRS will respond to the auto start of the pump IAW ARP for A-7 FIRE PROT FIRE and A-15 based on the deluge valve opening. This directs implementation of AB.FIRE, which checks if the CR is affected, then directs placing CR ventilation in fire outside the CR. AB.FP is NOT entered, because there is no indication of a malfunction, but indication of a valid deluge valve actuation. The shutdown distracter is plausible because it is the action required in AB.FP if both normal and backup fire protection systems are unavailable.

Reference Title

Control Room Fire Response

Fire Protection System Lesson Plan

Fire Protection System Malfunction.

Learning Objectives

FIRPROE004 Describe the function and operating characteristics for the following Fire Protection System components:
Fire Barrier Components:
Fire Doors
Fire Dampers
Penetration Seals
Fire Proofing
Marinite Walls

Energy Shields
Protective Wraps and Coatings
b. Fire Detection Devices:
Ionization detector
Thermal detector
Smoke and Fire detectors
c. Fire Protection Subsystems:
Water Supply System
Preaction Deluge System
Wet-Pipe Sprinkler System
Foam System
Carbon Dioxide System
Halon System
d. Fire Header Pressure Switches

Material Required for Examination	
Question Source:	New
Question Modification Method:	
Question Source Comments:	

Given the following conditions:

- Unit 2 has experienced a LOCA.
- 21 RHR pump has been C/T for the last 2 days.
- Containment pressure is 14 psig and rising slowly.
- 22 RHR pump trips while after performing Safeguards Reset actions in EOP-LOCA-1, Loss of Reactor Coolant.
- The CRS transitions to LOCA-5, Loss of Emergency Recirculation.
- The STA reports a valid PURPLE path on Containment Environment with containment pressure at 15 psig and rising slowly, and no higher PURPLE or any RED paths present.

Which of the following describes how the CRS should use Containment Spray pumps?

The CRS should start/stop Containment Spray pumps in.....

a. EOP-FRCE-1, Response to Excessive Containment Pressure, because Containment Spray pumps will not auto start with the SECs reset.

b. EOP-FRCE-1 because at least one Containment Spray pump is required to be running whenever containment pressure is >15 psig.

c. EOP-LOCA-5 to establish minimum required CS flow in order to conserve RWST inventory.

d. EOP-LOCA-5 since FRPs are not in effect when LOCA-5 is in effect.

Answer	c	**Exam Level**	S	**Cognitive Level**	Memory	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Plant Systems			**RO Group**	1	**SRO Group**	1		103000G422
103	Containment System							**Record Number**	18
2.4	Emergency Procedures / Plan								
2.4.22	Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations.							3.6	4.4

Explanation of Answer: 55.43(5) The transition to FRCE-1 is required upon a valid PURPLE path (15 psig containment). A is incorrect because FRCE-1 specifically asks if LOCA-5 is in effect, and if so, direct CS pumps to be operated IAW LOCA-5. B is incorrect because of the above reason, and additionally, using the table found in LOCA-5 as the bases, there are conditions with cont press>15 psig that NO CS pumps will be directed to be started. C is correct. D is incorrect because FRPs are in effect after the transition out of TRIP-1, and there is no direction to suspend FRPs either in LOCA-1 or LOCA-5.

Reference Title

Loss of Emergency Recirculation

Response to Excessive Containment Pressure

Learning Objectives

CONTMTE012	Discuss the procedural requirements associated with the Containment and Containment Support Systems, including an explanation of major precaution and limitations in the Containment and Containment Support Systems procedures
LOCA05E006	Describe the basis for each step, caution, note, and Continuous Action Summary item in LOSS OF EMERGENCY RECIRCULATION
FRCE00E006	Describe the basis for each step, caution, and note in 2-EOP-FRCE-1 thru 3 and EOP-CFST-1, Figure 5

Material Required for Examination

Question Source: Previous 2 NRC Exams **Question Modification Method:** Editorially Modified

Question Source Comments: Vision Q77740 modified to ask what procedure 08-01 NRC Exam

Given the following conditions:

- Unit 2 is in MODE 3 preparing for a startup.
- 21 RDMG set motor AND generator breakers are closed, and BOTH Reactor Trip Breakers A and B are shut for rod control testing.

Which of the following identifies how many Reactor Coolant loops are required to be in operation IAW Salem Tech Spec 3.4.1.2.c, Reactor Coolant System, Hot Standby, and correctly reflects its Bases?

a. Four, because single failure considerations require all loops in operation when rod control is energized.

b. One, because it is sufficient to provide positive pressure control of the RCS with a bubble established in the PZR.

c. Four, because potential energy additions from the secondary system require 4 loops to absorb that energy to prevent exceeding the limits of Appendix G to 10CFR Part 50.

d. One, because it provides adequate flow to ensure mixing, prevent stratification, and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System.

Answer	a	**Exam Level**	S	**Cognitive Level**	Memory	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Generic Knowledge and Abilities		**RO Group**	1	**SRO Group**	1			194001G132
GENERIC								**Record Number**	19
2.1	Conduct Of Operations								
2.1.32	Ability to explain and apply all system limits and precautions.							3.8	4.0

Explanation of Answer: 55.43(2) All of the choices have their reasons pulled from the Bases section of Tech Spec for RCP operation. The conditions in the stem indicate that Rod Control is energized. With rod control energized, 4 RCPs must be in operation. As per the Bases on page B3/4 4-1, it is for single failure criteria. B is incorrect because 4 loops are required. C is incorrect because the secondary system heat concerns are for starting a RCP with RCS cold legs <312°F. D is incorrect because it is one RCP, but has the correct bases for when only one RCP is required.

Reference Title

Salem Tech Specs

Salem Tech Specs Bases

Learning Objectives

RCPUMPE010	LOR Given a situation dealing with Reactor Coolant Pump operability, examine the situation and apply the appropriate Technical Specification action. (License Operator and STA only) NCT State the Technical Specification associated with the component, parameters and operation of the Reactor Coolant Pump including the Limiting Condition for Operation(s) (LCO) and the applicability of the LCO(s) (Non-licensed Operator)

Material Required for Examination

Question Source: Facility Exam Bank

Question Modification Method: Significantly Modified

Question Source Comments: Q27905 Question originally asked why all RCPs had to be running with energized rod control. Modified so that candidate had to determine rod control is energized, then decide how many RCPs had to be running, in addition to bases.

Given the following conditions:

Reactor Engineering (RE) contacts U2 Control Room at 0630 and reports that a shipment of new fuel was scheduled to arrive at 6 a.m.
RE reports the fuel has not arrived and the shipping company is unable to provide a prompt determination of where the shipment is.

Which of the following identifies the most restrictive reporting requirement for this event?

______ is required to be made by_____.

a. A One Hour Report; 0700.

b. A One Hour Report; 0730.

c. A Four Hour Report; 1100.

d. A Four Hour Report; 1130.

Answer	b	Exam Level	S	Cognitive Level	Application	Facility:	Salem 1 & 2	Exam Date:	12/3/2012
Tier:	Generic Knowledge and Abilities	RO Group	1	SRO Group	1				194001G135
GENERIC								Record Number	20
2.1	Conduct Of Operations								
2.1.35	Knowledge of the fuel-handling responsibilities of SROs.							2.2	3.9

Explanation of Answer: 55.43(7) When an expected shipment of SNM that is unaccounted for after the estimated arrival time is a one hour report under RAL 11.9.1.b. The 4 hour distracters are for other RALs associated with fuel also. The time required is from time of discovery per ECG Bases for RAL11.9.1.b

Reference Title
Salem ECG
Salem ECG Technical Bases

Learning Objectives	
ELO_23.b	Given an emergency event condition, describe the typical decisions with priorities that a shift supervisor must make in transitioning from normal operations to coping with an emergency event and implementation of the emergency plan.

Material Required for Examination: SRO 20 Salem ECG

Question Source: Facility Exam Bank | **Question Modification Method:** Significantly Modified

Question Source Comments: Vision Q77417 changed from 4 different reports 1,4,24 and UE, to 1 or 4, and added the required report by time.

Of the following, which identifies who will approve the installation of a Temporary Configuration Change IAW CC-AA-112-1001, Attachment 5, Temporary Configuration Change Package Installation?

a. System Manager(s) for affected system(s).

b. Site Engineering Director.

c. Operations Manager.

d. Shift Manager / CRS.

Answer	d	**Exam Level**	S	**Cognitive Level**	Memory	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Generic Knowledge and Abilities	**RO Group**	1	**SRO Group**	1				194001G211
GENERIC								**Record Number**	21

2.2 Equipment Control

2.2.11 Knowledge of the process for controlling temporary design changes. 2.3 3.3

Explanation of Answer

55.43(3) CC-AA-112 and associated T&RM delineate who approves Temporary Configuration changes. A is incorrect because the System Manager (SM) performs a review of the TCCP. The SM is also responsible for reviewing the limited duration requirements for temporary changes installed per Maintenance Rule (a)(4). SM is responsible for assisting the RE with post installation testing. SM is also responsible for ensuring that TCCP Extended Installation Justification is approved. B is incorrect because the Site Engineering Director (SED) only has overall responsibility for the Temporary Configuration Change Program. C is incorrect because the Ops Manager is not required to approve TCCs.

Reference Title

TEMPORARY CONFIGURATION CHANGES

Temporary Configuration Change Implementation T&RM

Learning Objectives

MISCAPE001	State who must authorize the installation/removal of Temporary Modifications in accordance with NC.NA-AP.ZZ-0013(Q)Control of Temporary Modifications.

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

During normal operations in MODE 1, it is discovered that 2A EDG monthly surveillance was not performed within its 31 day required periodicity.

The required surveillance was last performed 33 days ago.

Which of the following identifies the status of 2A EDG IAW Tech Specs, and why?

2A EDG is ...

a. INOPERABLE because it has exceeded its 31 day surveillance requirement.

b. INOPERABLE since the 24 hour delay time past the 31 day requirement has been exceeded.

c. OPERABLE because the normal surveillance interval plus 25% extension has not been exceeded.

d. OPERABLE because the surveillance can be performed within the 24 hour delay time which starts upon discovery of the missed surveillance.

Answer	c	**Exam Level**	S	**Cognitive Level**	Application	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Generic Knowledge and Abilities	**RO Group**	1	**SRO Group**	1				194001G237
GENERIC								**Record Number**	22
2.2	Equipment Control								
2.2.37	Ability to determine operability and/or availability of safety related equipment.							3.6	4.6

Explanation of Answer

55.43(2) Tech Spec 4.0.2 states..."Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the specified surveillance interval." Since the 25% of 31 days has not been exceeded, the EDG remains OPERABLE, since its surveillance is not required to be performed until 31+7.75 days. A is incorrect because of the 25% time. B is incorrect because the 24 hour delay time is not applicable until after the 25% extension expires, and since it is a 31 day frequency could be allowed to go to 31 days (per Tech Spec 4.0.3) D is incorrect because the 24 hour delay time is N/A first because the allowable time would be longer, and second because it is not applicable yet.

Reference Title

Salem Tech Specs

Learning Objectives

TECHSPE011	Describe the term Surveillance Requirement as it applies to the Technical Specifications
TECHSPE014	Describe the general requirements associated with Specifications 4.0.1 through 4.0.5 relating to implementation of the Technical Specification Surveillance Requirements

Material Required for Examination

Question Source: Facility Exam Bank

Question Modification Method: Concept Used

Question Source Comments: Vision 60376 made into a specific operability question.

Given the following condition:

- Unit 2 was manually tripped to enter a refueling outage at 20:00:00 on January 21st.

If all other requirements are met, which of the following is the EARLIEST time that movement of fuel in the Rx vessel could occur IAW Tech Specs?

a. 0400 on January 25th.

b. 0000 on January 26th.

c. 2000 on January 28th.

d. As soon as physically possible.

Answer	a	**Exam Level**	S	**Cognitive Level**	Application	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Generic Knowledge and Abilities		**RO Group**	1	**SRO Group**	1			194001G313
GENERIC							**Record Number**		23

2.3 Radiation Control

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. 3.4 3.8

Explanation of Answer

55.43(7,2) TSAS 3.9.3.a states that for refueling outages between Oct. 15- May 15th, the reactor shall be subcritical for 80 hours. 80 hours from 2000 on January 21 is 0400 on January 25th. Per tech Specs Bases, the minimum requirement for Rx subcriticality prior to movement of irradiated fuel assemblies in the Reactor Pressure Vessel ensures sufficient decay time has elapsed to allow the radioactive decay of the short lived fission products. The 80 hour decay time (LAR S08-01) is consistent with the assumptions used in the fuel handling accident analysis.

Reference Title

Salem Tech Specs

Learning Objectives

REFUELE012	Discuss the procedural requirements associated with the Refueling System, including an explanation of major precaution and limitations in the Refueling System procedures. (Licensed Operator & Non-licensed Operator only)

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Direct From Source

Question Source Comments: Vision Q84937

An explosion and fire at the RAP tank area has resulted in a possible large spill of radioactive water in the area. An Alert has been declared and all required facilities are activated and staffed. The Fire Department has determined that off-site assistance from the local fire department is needed.

IAW S2.OP-AB.FIRE-0001, Control Room Fire Response, which choice identifies who must authorize requesting off-site fire department assistance?

a. Security Duty Supervisor.

b. Nuclear Fire Protection Supervisor.

c. SM / Emergency Duty Officer (EDO).

d. Radiological Assessment Coordinator (RAC).

Answer	c	**Exam Level**	S	**Cognitive Level**	Memory	**Facility:**	Salem 1 & 2	**Exam Date:**	12/3/2012
Tier:	Generic Knowledge and Abilities			**RO Group**	1	**SRO Group**	1		194001G426
GENERIC								**Record Number**	24

2.4 Emergency Procedures / Plan

2.4.26	Knowledge of facility protection requirements, including fire brigade and portable fire fighting equipment usage.	3.1	3.6

Explanation of Answer: 55.43(7)55.43(5) CAS ATT. 1, Fire Dept. Support, Caution prior to step 3.0 in AB.FIRE-1 states, "In the event of a radiological emergency, the Nuclear Fire Protection Supervisor should obtain permission from the EDO/SM prior to calling for off-site assistance." A is plausible because security is required to be notified whenever off-site assistance is requested (CAS 2.0) B is plausible because they will be leading the fire brigade and will be the person to request the off-site assistance through the EDO. D is plausible because during an Emergency the RAC is associated with the radiological aspect of the emergency.

Reference Title

Control Room Fire Response

Learning Objectives

FIRPROE007	Identify and describe the local controls, indications, and alarms associated with the Fire Protection System, including: The location of Fire Protection System local controls and indications. (Licensed Operator & Non-licensed Operator only) The function of Fire Protection System local controls and indications. (Licensed Operator & Non-licensed Operator only) The plant and conditions or permissives required Fire Protection System local controls to perform their intended function. (Licensed Operator only) The setpoints associated with the Fire Protection System local alarms. (Licensed Operator only)

Material Required for Examination

Question Source: Previous 2 NRC Exams | **Question Modification Method:** Direct From Source

Question Source Comments: Salem 08-01 SRO NRC exam (5/17/2010)

Given the following conditions:

- 22 CVCS Monitor Tank was released to the Delaware River earlier this shift.
- The release was secured 2 hours ago during the performance of S2.OP-SO.WL-0002, Radioactive Release from 22 CVCS Monitor Tank.
- During a sample review after the release was secured, the Chemistry Department recognized that a Radioactive Liquid Release to the Delaware River was performed with an isotopic concentration which exceeded the ECG EAL RU1.3 Unusual Event threshold of 2X the ODCM for >60 minutes.

Which of the following describes how this should be addressed?

a. Initiate a non-emergency one hour report for this After-the-Fact event.

b. Notify the NJ DEP within one hour of the report by Chemistry Department that the release rate exceeded the ODCM.

c. Declare an Unusual Event based on exceeding the EAL at the time of the event, then terminate the UE because the EAL threshold is no longer being exceeded.

d. Declare an Unusual Event based on exceeding the EAL at the time of the event, then retract the UE because the EAL threshold was NOT exceeded when the declaration was made.

Answer	a	**Exam Level**	S	**Cognitive Level**	Memory	**Facility:** Salem 1 & 2	**Exam Date:** 12/3/2012
Tier:	Generic Knowledge and Abilities	**RO Group**	1	**SRO Group**	1		194001G429
GENERIC						**Record Number**	25
2.4	Emergency Procedures / Plan						
2.4.29	Knowledge of the emergency plan.					3.1	4.4

Explanation of Answer

55.43.(5) Salem ECG, Introduction and Usage, Section 8.6, Conditions Discovered After-the-Fact, describes an after-the-fact event as an event that exceeded an EAL threshold and was not recognized at the time of occurrence but is identified greater than one hour after the conditions has occurred and the condition no longer exists. The stem identifies that the liquid release was terminated 2 hours ago, which terminates the release exceeding the ODCM. After the Fact events that occur will be assessed and evaluated to ensure that no EAL current applies. An emergency declaration is NOT required and a non-emergency One-Hour Report should be initiated. The NJ DEP notification is required for spills, not discharges normally performed.

Reference Title

Salem ECG

Learning Objectives

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments: Objective from VISION is EPTRAININGE1, Given EP related issues and topics, analyze the issue, classify events and communicate to the States, IAW approved station procedures. Not listed in drop down menu.