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

KEY

Applicant Information

Name:	Region: I
Date: 12/18/2006	Facility: Salem 1 & 2
License Level: RO	Reactor Type: W
Start Time:	Finish Time:

KEY

Instructions

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

Applicant Certification

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

KEY

______________ Applicant's Signature

Results

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

Which of the following choices identifies the relationship between a Rx trip and a Turbine trip?

a. A Turbine trip will ONLY cause a Rx trip if power is < P-9.

b. A Rx trip will ONLY cause a Turbine trip if Rx power is >P-9.

c. A Turbine trip will ALWAYS cause a Rx trip to prevent lifting the PZR safeties.

d. A Rx trip will ALWAYS cause a Turbine trip to prevent an uncontrolled cooldown of the RCS.

Answer:	d	**Exam Level**	R	**Cognitive Level**	Memory	**Facility:**	Salem 1 & 2
Exam Date:	12/11/2006						
Tier:	Emergency and Abnormal Plant Evolutions	**RO Group**	1	**SRO Group**	1		000007K103
007	Reactor Trip					**Record Number**	1

EK1.	Knowledge of the operational implications of the following concepts as they apply to Reactor Trip:		
EK1.03	Reasons for closing the main turbine governor valve and the main turbine stop valve after a reactor trip	3.7	4.0

Explanation of Answer: A is incorrect because the power level needs to be ABOVE P-9 for a turbine trip to cause a Rx trip. B is incorrect because a Rx trip ALWAYS causes a turbine trip. C is incorrect because a turbine trip <P-9 will not cause a Rx trip. D is correct because a Rx trip always causes a turbine trip and the reason is IAW the Bases Document

Reference Title

Rx Trip or Safety Injection Bases Document

Learning Objectives

TRP001E022	Describe the basis for each Step, Caution, Note, and Continuous Action Summary item in EOP-TRIP-1

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

The crew has diagnosed a Pressurizer (PZR) Vapor Space Accident.

The following is the procedural flowpath followed:

-EOP-TRIP-1, REACTOR TRIP OR SAFETY INJECTION
-EOP-LOCA-1, LOSS OF REACTOR COOLANT
-EOP-LOCA-2, POST LOCA COOLDOWN AND DEPRESSURIZATION

90 minutes after the reactor trip, performing the COOLDOWN and DEPRESSURIZATION of the plant per LOCA-2 will result in …

a. a stable flowrate out the vapor space leak, and lowering PZR level.

b. a reduction in flowrate out the vapor space leak, and increasing PZR level.

c. a reduction in flowrate out the vapor space leak, and PZR level offscale high.

d. a stable flowrate out the vapor space leak, and stable PZR level.

Answer:	c	Exam Level	R	Cognitive Level	Application	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Emergency and Abnormal Plant Evolutions	RO Group	1	SRO Group	1				000008K102
008	Pressurizer Vapor Space Accident							Record Number	2

AK1. Knowledge of the operational implications of the following concepts as they apply to Pressurizer Vapor Space Accident:

AK1.02 Change in leak rate with change in pressure | 3.1 | 3.7

Explanation of Answer: Distracters a and d are incorrect because the whole purpose of the depressurization is to reduce the break flow by reducing pressure. C is correct and ~~d~~ is incorrect because PZR level will be off-scale high.

Reference Title

Post Loca Cooldown and Depressurization b [illegible] 12/8/06

Learning Objectives

LOCA02E002	Describe the plant response to EOP actions taken during POST LOCA COOLDOWN AND DEPRESSURIZATION

Material Required for Examination:

Question Source: Other Facility **Question Modification Method:** Editorially Modified

Question Source Comments: Byron 12/10/2003 NRC Exam. Modified procedure titles and correct answer. EOP-LOCA-2 Step 15 CAS states to stop depressurization at PZR level of 25% (33% adverse)

Given the following conditions:

- Unit 2 is operating at 100% power.
- RCS Tavg is 573 degrees.
- The unit experiences a SBLOCA.
- RCS pressure has dropped from NOP to 1825 psig.
- Using trended data, the highest CET has dropped from 614 degrees to 560 degrees.

Subcooling has gone from _____ to _____.

a. 39; 64

b. 64; 39

c. 81;93

d. 93; 81

Answer	a	**Exam Level**	R	**Cognitive Level**	Application	**Facility:**	Salem 1 & 2	**Exam Date:**	12/11/2006
Tier:	Emergency and Abnormal Plant Evolutions	**RO Group**	1	**SRO Group**	1				000009K102
009	Small Break LOCA							**Record Number**	3

EK1.	Knowledge of the operational implications of the following concepts as they apply to Small Break LOCA:		
EK1.02	Use of steam tables	3.5	4.2

Explanation of Answer: Saturation temperature at NOP (2250 psia) is 653 deg. Highest CET in stem is 614. 653-614=39. Saturation temp at 1840 psia is 624 deg. Highest CET in stem is 560. 624-560=64. 81 degrees is subcooling if use TAVG of 573 instead of highest CET. 93 is if use 2235 psig for current pressure and current temp of 560.

Reference Title
Steam Tables

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: Steam Tables

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

Question Source Comments:

Given the following conditions:

- Salem Unit 2 is operating at 100% power.
- 22 RHR pp is C/T.
- A catastrophic failure of RCS loop 21 cold leg piping occurs.
- RCS pressure is 35 psig.
- Initial RWST level was 41.1 feet.

Given the RWST tank curve from S2.OP-TM.ZZ-0002 TANK CAPACITY DATA, which of the following choices identifies the time available until the swap to Cold Leg Recirc will be required?

a. Between 30-31 minutes.

b. Between 22-23 minutes.

c. Between 18-19 minutes.

d. Between 13-14 minutes.

Answer	c	Exam Level	R	Cognitive Level	Application	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Emergency and Abnormal Plant Evolutions	RO Group	1	SRO Group	1				000011K202
011	Large Break LOCA							Record Number	4

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

Explanation of Answer

Explanation: Question stem describes the design basis LOCA, but with power. With the RCS at 35 psig, all available ECCS pumps will be injecting at their maximum rate. The flow rates used are: Charging pumps 2x550= 1100gpm; SI 2x650= 1300; RHR 1x4600= 4600; and Containment Spray pump flow of 2x2600=5200 So, 1100+1300+4600+5200= 12,200gpm 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 18.33 minutes. Distracter a is the time it would take to pump in the entire RWST volume with available pumps I/S. Distracter b is the time if one CS pump is used. Distracter d is if 16,800 gpm used (all pumps I/S)

Reference Title

TANK CAPACITY DATA

2-EOP-LOCA-3

Learning Objectives

ECCS00E008	Identify and describe the Control Room controls, indications, and alarms associated with the Emergency Core Cooling System, including: (Licensed Operator & STA only) The Control Room location of Emergency Core Cooling System control bezels and indications. The function of each Emergency Core Cooling System Control Room control and indication. The effect each Emergency Core Cooling System control has upon Containment Spray System components and operation. The plant conditions or permissives required for Emergency Core Cooling System Control Room controls to perform their intended function.
LCA3U1E004	Determine the indications that are monitored to ensure proper system/component operation for each step in 2-EOP-LOCA-3

Material Required for Examination	S2.OP-TM.ZZ-0002, PAGE 28, RWST TANK CURVE		
Question Source:	Facility Exam Bank	Question Modification Method:	Significantly Modified
Question Source Comments:	Changed conditions in the stem to make a distracter correct and the old correct answer wrong.		

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

b. An ATWT. The Rx should have tripped on 1/4 RC Loops Lo Flow <90%.

c. 22 RCP shaft has sheared.

d. 22 RCP shaft has seized.

Answer	c	Exam Level	R	Cognitive Level	Comprehension	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Emergency and Abnormal Plant Evolutions	RO Group	1	SRO Group	1				000015A111
015	Reactor Coolant Pump Malfunctions							Record Number	5
AA1.	Ability to operate and / or monitor the following as they apply to Reactor Coolant Pump Malfunctions:								
AA1.11	RCP on/off and run indicators							2.5	2.4

Explanation of Answer

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 b 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 a is incorrect because there are 3 low pressure flow taps, and 1 common high pressure flow tap. Distracter d is incorrect because a seized RCP shaft would cause its supply breaker to trip on ovewrcurrent. The indication in the stem is that the breaker is closed. C 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

Reactor Coolant System

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)

Material Required for Examination

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

Question Source Comments:

Given the following conditions:

- Unit 2 is operating at 40% steady state power.
- 23 CVCS Pp I/S
- 21,23 CC Pps I/S
- 21,24 SW Pps I/S
- 22 SW Pp in AUTO

A loss of 2A 460 Volt Vital Bus occurs. One minute after the loss of bus, with NO OPERATOR ACTION, which of the following will be observed?

a. Letdown Isolation.

b. PZR level dropping ~1% / minute.

c. Control Console alarm, 21 (22) CC HDR PRESSURE LO.

d. OHA B13 21 SW HDR PRESS LO, and/or OHA B-14, 22 SW HDR PRESS LO.

Answer	b	**Exam Level**	R	**Cognitive Level**	Comprehension	**Facility:**	Salem 1 & 2	**Exam Date:**	12/11/2006
Tier:	Emergency and Abnormal Plant Evolutions	**RO Group**	1	**SRO Group**	1				000022G421

022	Loss of Reactor Coolant Makeup	**Record Number**	6

2.4	Emergency Procedures / Plan

2.4.21	Knowledge of the parameters and logic used to assess the status of safety functions including: 1. Reactivity control 2. Core cooling and heat removal 3. Reactor coolant system integrity 4. Containment conditions 5. Radioactivity release control.	3.7	4.3

Explanation of Answer

Reactor coolant integrity is maintained, however, the RCS is leaking ~ 5 gpm up the shaft of each RCP. This is occurring since the only operating charging pump lost power, and no operator action is taken. Uncorrected this will lead to a loss of PZR level. A is incorrect because the 23 charging pump is powered from 2A 460V bus but does NOT have any UV trip associated with it. The breaker remains closed, with no power supplied. The interlock for isolating letdown requires ALL 3 charging pump breakers to be open, it does not operate on no flow. B is correct because with no letdown isolation and normal letdown flow of 75 gpm, with no charging flow PZR level will drop. A thumbrule is 75 gallons per percent in the PZR at NOT. C is incorrect because 21 CC pump is powered from 2A 4KV bus, and is not affected. D is incorrect because no SW pumps will be lost.

Reference Title

No. 1&2 Units-CVCS No. 1CV4&2CV4 Letdown Oriface Isolation Valves

Learning Objectives

CVCS00E006	LOR NCT Outline the interlocks associated with the following Chemical and Volume Control System components: a) VCT Isolation Valves, CV40 and CV41 b) Letdown Isolation Valves, CV2 and CV277 c) Letdown Orifice Isolation Valves, CV3, CV4 and CV5 d) Centrifugal Charging Pumps
AB460VE001	Describe the operation of the following system as applied to S1/S2.OP-AB.460-0001/2/3: a) 460/230V Vital Bus Distribution

Material Required for Examination

Question Source:	Facility Exam Bank	**Question Modification Method:**	Direct From Source
Question Source Comments:	VISION Q71244		

Which of the following describes the situation in which the GREATEST amount of reactivity will be added during the first 5 minutes of boration while performing a Rapid Boration IAW S2.OP-SO.CVC-0008, RAPID BORATION?

The Rapid Boration flowpath is aligned....

a. via the Boric Acid Blender @ EOL.

b. via 2CV175 with the Rx core @ BOL.

c. via 2CV174 with the Rx core at 10,000 MWD/MTU.

d. from the RWST with the Rx core at 15,000 MWD/MTU.

Answer	b	Exam Level	R	Cognitive Level	Application	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Emergency and Abnormal Plant Evolutions	RO Group	2	SRO Group	2				000024K102
024	Emergency Boration							Record Number	7

AK1. Knowledge of the operational implications of the following concepts as they apply to Emergency Boration:

AK1.02 Relationship between boron addition and reactor power | 3.6 | 3.9

Explanation of Answer: WRITTEN FOR CYCLE 15 Choices a,b,and c all have the same flow rate for boron injection during a rapid boration. The difference in those 3 choices is time in core life which affects Boron worth and RCS boron concentration. The RWST distracter is incorrect because the boron concentration is much less than the BAST's, which is the source of the other 3 methods. Using the REM figures, the differential boron worth is -6.325, -6.725, and -6.9 pcm/ppm respectiveley for a,b, and c. C is correct because it has the highest reactivity worth for the same boron flow rate.

Reference Title

Figures

Figures

Learning Objectives

CVCS00E015

LOR NCT Given plant conditions, relate the Chemical and Volume Control System with the following,
Pressurizer Level Control System
RCS Temperature Control
Main Turbine/Generator
Reactor Coolant Pump seal injection flows
Automatic Control Rod Control
VCT Makeup
Nuclear Instrumentation
Emergency Core Cooling System
Residual Heat Removal System
Component Cooling Water System
Pressurizer Pressure Control System
Pressurizer including Pressure Relief Tank
Waste Gas
Waste Liquid
Service Water
4 Kv Vital AC System
480 V Vital AC System
240 V Vital AC System
125 VDC System

Material Required for Examination: S2.RE-RA.ZZ-0012(Q) FIGURES

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

Question Source Comments: Developed using 2R15 cycle data

Given the following conditions:

- Unit 1 is operating at 100% power.
- A PZR safety valve fails open.

Assuming the safety valve remains open and a Rx trip is performed, which of the following describes RCP operation strategy, and what is the bases for that strategy?

Trip ALL RCPs at RCS pressure of 1350 psig if...

a. ANY ECCS pump is supplying at least 100 gpm ECCS flow. This prevents depletion of RCS inventory which might lead to severe core uncovery if the RCPs were tripped later in the accident.

b. ANY Charging or SI pump is supplying at least 100 gpm ECCS flow. This prevents depletion of RCS inventory which might lead to severe core uncovery if the RCPs were tripped later in the accident.

c. ANY ECCS pump is supplying at least 100 gpm ECCS flow. This prevents formation of a stagnant water volume in the upper head region which may flash and form a steam bubble during subsequent cooldown and depressurization.

d. ANY Charging or SI pump is supplying at least 100 gpm ECCS flow. This prevents formation of a stagnant water volume in the upper head region which may flash and form a steam bubble during subsequent cooldown and depressurization.

Answer	b	Exam Level	R	Cognitive Level	Memory	Facility:	Salem 1 & 2	Exam Date:	12/11/2006

Tier:	Emergency and Abnormal Plant Evolutions	RO Group	1	SRO Group	1	000027G418
027	Pressurizer Pressure Control Malfunction			Record Number	8	
2.4	Emergency Procedures / Plan					
2.4.18	Knowledge of the specific bases for EOPs.				2.7	3.6

Explanation of Answer: With a safety valve open, operators are directed to trip the Rx and go to TRIP-1. Distracters A and C are incorrect because ANY ECCS pump would include the RHR pumps. If the RHR pump is injecting, RCS pressure is less than 300 psig and the LOCA is Large, and the RCP trip criteria are for pump protection only.

Reference Title
Rx Trip or Safety Injection
Westinghouse ERG- RCP Trip/Restart
Pressurizer Pressure Malfunction

Learning Objectives	
TAA000E015	Assess TAA conditions that affect heat removal rates

Material Required for Examination	

Question Source:	New	Question Modification Method:	

Question Source Comments:	

Given the following conditions:

- Unit 2 is operating at 100%.
- Reactor Trip Breaker "A" and Reactor Trip Bypass Breaker "B" are racked in and shut.
- Reactor Trip Breaker "B" is open.
- A feedwater problem has developed, and the CRS directs the RO to trip the reactor.
- The RO depresses the OPEN pushbuttons for the Rx Trip Breakers, but the Rx does NOT trip.

Assuming no automatic trip demand has been generated, and the RO has NOT attempted a trip by any other means, which of the following conditions prevented the Rx from tripping?

a. Reactor Trip Breaker "A" shunt trip coil did not energize.

b. Reactor Trip Breaker "A" UV coil did not de-energize.

c. Reactor Trip Bypass Breaker "B" shunt coil did not energize.

d. Reactor Trip Bypass Breaker "B" UV coil did not de-energize.

Answer	a	**Exam Level**	R	**Cognitive Level**	Comprehension	**Facility:**	Salem 1 & 2	**Exam Date:**	12/11/2006
Tier:	Emergency and Abnormal Plant Evolutions	**RO Group**	1	**SRO Group**	1				000029K206
029	Anticipated Transient Without Scram							**Record Number**	9
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

The control console PB are only control-function for the Reactor Trip Breakers. The Reactor Trip Bypass breakers are indicate only on 2CC2. The Reactor Trip breaker CC2 PB ONLY energizes the shunt coil of its specific breaker. The correct answer is "a" because the shunt coil for Reactor Trip Breaker "A" did not energize to open the breaker. Opening EITHER of the 2 breakers in series will remove power to the control rods and cause a rx Trip. Distracter "b" is incorrect because the UV coil is not expected to de-energize when the breaker bezel PB is depressed. Distracters "c" and "d" are incorrect because the Bypass Breakers do not have a control function from the 2CC2, only breaker position indication.

Reference Title

Reactor Protection System Reactor Trip Signals

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: New **Question Modification Method:**

Question Source Comments:

Given the following conditions:

- Unit 2 was tripped from 100% power 20 minutes ago.
- 2N35 indicates 2.0E-9 amps.
- 2N36 indicates 5.0E-11 amps.

Which of the following choices identifies the condition present, and any action(s) performed as a result of this condition?

Intermediate Range NI's are indicating...

a. that N35 is under compensated. Manually reset the Source Range NI channels.

b. that N36 is over compensated. Manually reset the Source Range NI channels.

c. correctly. Ensure P-6 is blocked when the second IR NI channel goes below 7E-11 amps.

d. correctly. Ensure Source Range channels reset when the second IR NI channel goes below 7E-11 amps.

Answer	a	Exam Level	R	Cognitive Level	Memory	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Emergency and Abnormal Plant Evolutions		RO Group	2	SRO Group	2			000032A101
032	Loss of Source Range Nuclear Instrumentation							Record Number	10

AA1.	Ability to operate and / or monitor the following as they apply to Loss of Source Range Nuclear Instrumentation:		
AA1.01	Manual restoration of power	3.1*	3.4*

Explanation of Answer

After a Rx trip, power will lower at -1/3 decade per minute following the prompt drop in power. 20 minutes times -1/3 dpm = 6.6 decades. 100% power is 5E-5 amps, so in 20 minutes power should have dropped at least to 5E-11 amps. With the N35 channel reading 2E-9 it is over 2 decades above where it should be. This points to under compensation as the problem, since more of the gamma pulses are being seen by the detector which are not "screened out." Distracter B is incorrect because if the N36 were overcompensated it would read low off scale and N35 would read normally. Distracters C and D are incorrect because the N35 is under compensated.

Reference Title

Reactor Trip Response

Learning Objectives

EXCOREE010	*State the setpoints, coincidence, blocks and permissives for automatic actuations associated with the Excore Nuclear Instrumentation System including the following:* Source Range High Flux Reactor Trip Intermediate Range High Flux Reactor Trip Power Range (Low) High Flux Reactor Trip Power Range (High) High Flux Reactor Trip Power Range High Rate Reactor Trip

Material Required for Examination			
Question Source:	Facility Exam Bank	Question Modification Method:	Significantly Modified
Question Source Comments:	Modified VISION Q61948 to different detector having different compensation problem, making a distracter right and the previously correct answer wrong.		

Given the following conditions:

- Unit 1 is performing a Rx startup.
- With Rx power at 1E-5 amps, 1N35 fails low.

Which of the following, if any, is the Power Range NI reading which will confirm Rx power is actually at 1E-5 amps?

a. 1%.

b. 20%.

c. 100%.

d. Power is below the Power Range.

Answer	b	Exam Level	R	Cognitive Level	Application	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Emergency and Abnormal Plant Evolutions	RO Group	2	SRO Group	2				000033A201
033	Loss of Intermediate Range Nuclear Instrumentation							Record Number	11

AA2.	Ability to determine and interpret the following as they apply to Loss of Intermediate Range Nuclear Instrumentation:		
AA2.01	Equivalency between source-range, intermediate-range, and power-range channel readings	3.0	3.5

Explanation of Answer

The Intermediate Range NI indication is expected to overlap with the Power Range between 4 and 6 E-6 Amps (IOP-3 PAGE 27). Also AV–1026R drawing shows PR overlap at 5E-6. Since power in stem is given as 1E-5 amps, the only answer is 20%. 1% would be ~5-6E-6. 100% would be 5E-4. Power is not below the power range.

Reference Title

Hot Standby to Minimum Load

Nuclear Detector Range

Learning Objectives

EXCOREE009	Identify and describe the Control Room controls, indications, and alarms associated with the Excore Nuclear Instrumentation System, including: The Control Room location of Excore Nuclear Instrumentation System control bezels and indications. The function of each Excore Nuclear Instrumentation System Control Room control and indication. The effect each Excore Nuclear Instrumentation System control has upon Excore Nuclear Instrumentation System components and operation. The plant conditions or permissives required for Excore Nuclear Instrumentation System Control Room controls to perform their intended function. The setpoints associated with the Excore Nuclear Instrumentation System control room alarms.

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

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

a. To minimize S/G mass loss during a SGTR with a Main Steamline Break.

b. To prevent high alarm on 2R40, RAD MON CONDENSATE PRCS FILTER, from isolating the Condensate Polisher.

c. To prevent backfeeding contamination from 21 S/G to any other S/G through the unaffected S/G's blowdown lines.

d. To prevent the spread of contamination from a Steam Generator Tube Rupture (SGTR) on 21 S/G to secondary systems.

Answer	d	Exam Level	R	Cognitive Level	Memory	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Emergency and Abnormal Plant Evolutions	RO Group	1	SRO Group	1				000038K303
038	Steam Generator Tube Rupture							Record Number	12
EK3.	Knowledge of the reasons for the following responses as they apply to Steam Generator Tube Rupture:								
EK3.03	Automatic actions associated with high radioactivity in S/G sample lines							3.6*	4.0

Explanation of Answer

d is correct because 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. a is incorrect because S/G mass lost to blowdown is negligible when compared to MSLB. C is incorrect because the S/G each have its own blowdown line, so backfeeding contamination is not possible through the blowdown lines. b is incorrect because the polisher does not receive an isolation signal from 2R40

Reference Title

Steam Generator Tube Rupture

Steam Generator Blowdown Operation

Learning Objectives

ABSG01E001 Describe the operation of the following system as applied to S1/S2.OP-AB.SG-0001:
a) ADFWCS response to increasing SG level.
b) CVCS make-up response to decreasing pressurizer level.
c) Radiation Monitor response to increasing SG activity.

Material Required for Examination	
Question Source:	Facility Exam Bank
Question Modification Method:	Direct From Source
Question Source Comments:	VISION Q78166

Given the following conditions:

- Unit 2 is operating at 85% power.
- Rx power is rising slowly.
- RCS Tave is dropping slowly.
- Containment pressure is 0.1 psig and steady.

Which of the following is causing these indications, and what actions are required?

a. A Main Turbine Governer Valve is slowly failing open, trip the RX IAW S2.OP-AB.STM-0001.

b. A RCS leak > 10 gpm in the letdown piping OUTSIDE containment, isolate letdown IAW S2.OP-AB.RC-0001, REACTOR COOLANT SYSTEM LEAK.

c. A normal dilution of 100 gallons was set as 1,000 gallons in the Primary Water Flow Register and performed. Initiate a rapid Boration IAW S2.OP-SO.CVC-0006, RAPID BORATION.

d. An inadvertent boration is occurring, place the CVCS Make-up Control in MANUAL and close malfunctioning valves IAW S2.OP-SO.CVC-000, BORON CONCENTRATION CONTROL.

Answer	a	Exam Level	R	Cognitive Level	Comprehension	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Emergency and Abnormal Plant Evolutions	RO Group	1	SRO Group	1				000040A202
040	Steam Line Rupture							Record Number	13

AA2.	Ability to determine and interpret the following as they apply to Steam Line Rupture:		
AA2.02	Conditions requiring a reactor trip	4.6	4.7

Explanation of Answer: A is correct because it would cause all the indications in the stem. The AB.STm states at steps 3.3-3.6 that if an EH problem is causing turbine load to be improperly controlled with the turbine >49% power, the trip the reactor. Distracter b is incorrect because an RCS leak would not cause a power change or Tave change, even though it would cause charging flow to rise. Distracter c is incorrect because the dilution would not cause Tave to lower, it would rise along with Rx power and charging flow as PZR level rose. Distracter d is incorrect because a boration would cause Rx power to lower and charging flow to lower as PZR programmed level dropped while lowering RCS temperature.

Reference Title

Excessive Steam Flow

Learning Objectives

ABSTM1E004

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: New **Question Modification Method:**

Question Source Comments:

Given the following conditions:

- Unit 2 is operating at 100% power when a simultaneous loss of BOTH SGFPs occurs.
- The Rx is manually tripped.

Which of the following choices describes how AFW flow will be controlled after the Immediate Actions of EOP-TRIP-1 are performed, and why?

a. Manually reduce total AFW flow to no less than 22E4 lbm/hr to prevent an excessive RCS cooldown.

b. Ensure total AFW flow is no less than 44E4 lbm/hr to prevent an un-needed transfer to FRHS-1, RESPONSE TO LOSS OF SECONDARY HEAT SINK.

c. The Pressure Overide Defeat PBs will be required to be depressed since runout flow cannot be prevented to the SGs when they shrink and depressurize following the loss of feed.

d. AFW flow from 2 MDAFW pumps is sufficient for decay heat removal following ANY Rx trip, so only 23 AFW pump flow should be reduced to zero by idling the 23 AFW pump to prevent over feeding the SGs.

Answer	a	**Exam Level**	R	**Cognitive Level**	Memory	**Facility:**	Salem 1 & 2	**Exam Date:**	12/11/2006
Tier:	Emergency and Abnormal Plant Evolutions	**RO Group**	1	**SRO Group**	1				000054K303
054	Loss of Main Feedwater							**Record Number**	14

AK3. Knowledge of the reasons for the following responses as they apply to Loss of Main Feedwater:

AK3.03 Manual control of AFW flow control valves | 3.8 | 4.1

Explanation of Answer: EOP-TRIP-2, Reactor Trip Response checks AFW flow > 22E4 lbm/hr to ensure an adequate heat sink is maintained. 23 AFW pump speed is reduced to idle. The Basis Document for TRIP-2 references C0542, which provides direction to throttle AFW flow to minimize cooldown from excessive feedwater flow. A is correct because it contains both parts of the requirement in TRIP-2. Distracter b is incorrect because 44E4 lbm/hr is the AFW flow required in FRSM-1, and will over cool the RCS. Transfer to FRHS-1 would only be required with AFW flow < 22E4 and NR levels all less than 9%. Distracter c is incorrect because the setting of the AF21 valves prevents runout. Distracter d is incorrect because leaving 2 MDAFW pumps running at 95% AF21 open (normal setting) would cause an excessive cooldown.

Reference Title

Reactor Trip Response

Learning Objectives

TRP002E002	Describe the plant response to actions taken in the following EOP step sequence(s): 3, 4.1, 5, 6, 8, 10, 16, 19, 20, and 24.1

Material Required for Examination:

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

Question Source Comments:

Given the following conditions:

- Unit 2 has lost all off site power.
- 2A EDG failed to start.
- 2B EDG tripped on overcrank.
- 2C EDG started but its output breaker tripped on 2C Vital bus differential current.
- After isolating SW to the Turbine Building at Step 19 of EOP-LOPA-1, Loss of All AC Power, 2A EDG is successfully started and its output breaker is shut.

Which of the following describes the next action(s) to be performed, and why?

a. Start 21 or 22 SW pump to provide cooling to 2A EDG.

b. Start 25 or 26 SW pump to provide cooling to 2A EDG.

c. Close 22 and 24 SW20 valves, NUC HDR ISO VLVS to prevent water hammer to the 2A EDG, start ONE SW pump, and throttle open the 22SW20 to repressurize the nuc header prior to putting full flow to the header.

d. Close 22 and 24 SW20 valves, NUC HDR ISO VLVS to prevent water hammer to the 2A EDG, start ALL available SW pumps, and throttle open the 22SW20 to repressurize the nuc header prior to putting full flow to the header.

Answer	a	Exam Level	R	Cognitive Level	Application	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Emergency and Abnormal Plant Evolutions	RO Group	1	SRO Group	1				000055A106
055	Station Blackout							Record Number	15
EA1.	Ability to operate and / or monitor the following as they apply to Station Blackout:								
EA1.06	Restoration of power with one ED/G							4.1	4.5

Explanation of Answer

The overiding concern after starting a DG in LOPA-1 is in getting cooling water flow to the EDG. The stem of the question states that SW has just been isolated to the TGA. The operator needs to know where in LOPA-1 this occurs in relation to WHEN the EDG gets started and power is restored to the bus. In the case presented above, power is restored PRIOR to step 42, and the CAS 14 states to close EDG output breaker and start a SW pump. The operator must also know which SW pumps would be available from 2A EDG. The 2 longer distracters would be the steps performed if the restoration of power from the single EDG happened later in the procedure.

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:	New	Question Modification Method:	
Question Source Comments:			

Given the following conditions:

- Unit 2 is operating normally at 100% power.
- 23 charging pump is in service.
- 21 and 24 SW pumps are in service.
- A loss of the 500KV switchyard occurs.

Which of the following contains ONLY equipment that will be running as determined by the 2RP4 status lights?

a. 2 ECAC, 21 CFCU, 22 AFP.

b. 24 SW pump, 21 CC pump, 22 Chiller.

c. 25 CFCU, 23 Rx Nozzle Support Fan, 2 ECAC.

d. 23 SW pump, 21 Aux Bldg Supply Fan, 22 Rx Shield Vent fan.

Answer	b	Exam Level	R	Cognitive Level	Memory	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Emergency and Abnormal Plant Evolutions	RO Group	1	SRO Group	1				000056A202
056	Loss of Off-Site Power							Record Number	16

AA2. Ability to determine and interpret the following as they apply to Loss of Off-Site Power:

AA2.02 ESF load sequencer status lights 3.5* 3.6*

Explanation of Answer

The ESF load sequencer will load all vital busses on their respective EDG when all off-site power is lost. This is a MODE II Blackout signal, and the SEC Sequencer number 2 will be used. In this Mode, the vital busses are stripped, the EDGs start and energized the vital busses, and loads are sequenced on to the bus to prevent overloading the EDG if all loads were to start simultaneously. The correct answer b contains ONLY equipment that will be started and indicated on 2RP4 as having their breaker closed. All the distracters contain 1 component which is NOT loaded during a Blackout. For Distracter a it is the 21 CFCU. For Distracter c it is 25 CFCU. For Distracter d it is the 23 SW pump. The 24 SW pump is selected as the "Lead" pump, and will start on B vital bus. The 23 pump will only start if the 23 pump does not start. (Dwg 203668 contains the tables of loads sequenced on and the SW pump start logic.)

Reference Title

Salem 500-4KV Electrical Distribution Simplified One-Line

Safeguards Emergency Loading Sequence Logic Diagram

Service Water Pump Operation

Learning Objectives

SEC000E005	LOR NCT List the equipment actuated by the Safeguards Equipment Control System in each Mode Operation

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

Given the following conditions:

- 1A EDG is set up for normal standby operation.
- 125VDC breaker 1ADC2AX26, 1A DIESEL GENERATOR UNIT TRIPS, fails and trips open.
- The Emergency supply breaker to the 1A EDG unit trips has NOT been closed in yet.

The EDG will mechanically...

a. start, but will NOT be capable of flashing its field, due to not having a PMG on the shaft.

b. NOT start from ANY start signal, since the DUTR must be energized to allow the EDG to start.

c. start ONLY if the Fire Bypass Switches are placed in BYPASS, which allows local starting of the EDG.

d. start ONLY from a SEC signal, since the SEC start circuitry is independent of local or control room start circuitry.

Answer	b	Exam Level	R	Cognitive Level	Application	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Emergency and Abnormal Plant Evolutions		RO Group	1	SRO Group	1			000058K301
058	Loss of DC Power							Record Number	17
AK3.	Knowledge of the reasons for the following responses as they apply to Loss of DC Power:								
AK3.01	Use of dc control power by D/Gs							3.4*	3.7

Explanation of Answer: The EDG start circuitry can only start the EDG when there are NO trip signals present. The DUTR relay has to be ENERGIZED to allow the EDG to start. When a trip signal is present, the DUTR DEENERGIZES and trips the EDG if running, or prevents a start if it is secured. The control power has a normal and emergency source, but the stem states the emergency supply breaker has not been closed in yet. The EDG will not start under any circumstances.

Reference Title

1A & 2A EDG Unit Trip and Breaker Failure Protection

No. 1 Unit 1AADC Distribution Cabinet

1A & 2A EDG Alarms

Learning Objectives

DCELECE014	Given a DC Electrical System failure, predict the effect of the DC Electrical System failure on the following: (License Operator and STA only) Emergency Diesel Generators Components using DC control power

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

Given the following conditions:

- Unit 2 is in MODE 6, with core reload in progress.
- Containment Purge is in service.
- Water level over the Rx vessel flange is > 23'.
- The Spent Fuel Pool Gate Valve is open.

Which of the following identifies a condition which would require IMMEDIATE suspension of irradiated fuel movement in containment IAW Technical Specifications?

a. BOTH the inner and outer 100' Airlock doors are opened.

b. A valid 2R5 alarm is received in the Spent Fuel Handling Building.

c. 22 SG secondary side manway is opened, and the entire Main Steam line is C/T, vented, and drained.

d. The Containment Coordinator reports that the Equipment Hatch inside door is being held in place with only 3 bolts, but a 4th bolt is available in containment.

Answer	c	Exam Level	R	Cognitive Level	Memory	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Emergency and Abnormal Plant Evolutions	RO Group	2	SRO Group	2				000069A101
069	Loss of Containment Integrity							Record Number	18
AA1.	Ability to operate and / or monitor the following as they apply to Loss of Containment Integrity:								
AA1.01	Isolation valves, dampers, and electropneumatic devices.							3.5	3.7

Explanation of Answer

Distracter a is incorrect because TSAS 3.9.4.b states that a minimum of one door in each airlock must be CAPABLE of being closed. Distracter b is incorrect because a valid area radiation alarm in the FHB requires suspension of fuel movement in the FHB only, not containment. (S2.OP-IO.ZZ-0010 PAGE 4) C is correct because the secondary side of the SG, when open in containment, will provide a direct path to the outside if the steam line is vented, drained and C/T, because a drain line in the outer penetration area MUST be open to C/T the steam line. Distracter d is incorrect because IAW TSAS 3.9.4.a the equipment hatch door only needs to be CAPABLE of being closed and secured with 4 bolts, which it can be if it is already closed and secured with 3 bolts with a 4th bolt in containment.

Reference Title

Technical Specifications

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

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

Which of the following Process Radiation Monitors reaching its alarm setpoint would require operator action to isolate its process path?

a. 2R18- Liquid Waste Disposal.

b. 2R31- Letdown Line-Failed Fuel.

c. 2R41D- Plant Vent Release Rate.

d. 2R17B- Component Cooling Header 22.

Answer: b **Exam Level:** R **Cognitive Level:** Memory **Facility:** Salem 1 & 2 **Exam Date:** 12/11/2006

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 2 000076K201

076 High Reactor Coolant Activity **Record Number:** 19

AK2. Knowledge of the interrelations between High Reactor Coolant Activity and the following:

AK2.01 Process radiation monitors 2.6 3.0

Explanation of Answer: All of the distracters have an automatic function that acts to close the pathway from the process to the atmosphere EXCEPT the R31, which does not. With no automatic isolation of the letdown line, RCS water will end up in the VCT, and be pumped from the charging pumps back through the regen HX to the RCS.

Reference Title

Abnormal Radiation

Learning Objectives

ABRAD1E001	Describe the operation of radiation monitors as applied to S2.OP-AB.RAD-0001(Q): a) Radiation monitor response to high radiation; including actions that occur as a result of the channel in warning or alarm.
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: New **Question Modification Method:**

Question Source Comments:

Given the following conditions:

- Unit 2 experienced a Rx trip and safety injection following a small pipe rupture in containment.
- The crew is performing a cooldown and depressurization IAW EOP-LOCA-2, Post LOCA Cooldown and Depressurization.

Which of the following describes how the cooldown will be performed, and why?

a. Dump steam at MAXIMUM rate with the 21-24MS10 Atmospheric Relief Valves. This will minimize the amount of RCS inventory loss.

b. Operate the Main Steam Dumps in MS PRESSURE CONTROL - AUTO mode and reduce temperature in discrete steps. This will allow the simultaneous reduction of RCS pressure which leads to an overall faster method of reducing break flow.

c. Dump steam using 21-24MS10 at a rate to ensure RCS subcooling remains greater than 20 degrees. This will prevent an unwanted transition to FRCC-2, Response to Degraded Core Cooling, which would lead to a RCP start which would raise RCS inventory loss rate.

d. Operate the Main Steam Dumps in MS PRESSURE CONTROL - MANUAL mode, dump steam at rate not to exceed 100 degree / hr cooldown rate. This will prevent entry into FRTS-1, Response To Imminent Thermal Shock, which would require an 8 hour soak and raise the amount of RCS inventory loss.

Answer	d	Exam Level	R	Cognitive Level	Memory	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Emergency and Abnormal Plant Evolutions		RO Group	2	SRO Group	2			00WE03K101
E03	LOCA Cooldown and Depressurization							Record Number	20

EK1.	Knowledge of the operational implications of the following concepts as they apply to LOCA Cooldown and Depressurization:		
EK1.1	Components, capacity, and function of emergency systems.	3.4	4.0

Explanation of Answer

D is correct because at step 10 of LOCA-2, 100 degree limit is stated. The basis is also stated to prevent entry into FRTS. The preferred method of heat removal is stea dumps, and they are available. A and C are wrong because of the rate of temp reduction. B is wrong because the MS dumps are not operated in that manner, and a continual cooldown is prefferred over step changes, which add stresses to RCS components.

Reference Title
Post LOCA Cooldown and Depressurization

Learning Objectives	
LOCA02E001	Describe the EOP mitigation strategy during POST LOCA COOLDOWN AND DEPRESSURIZATION.

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

Given the following conditions:

- Unit 2 is operating at 25% during a power ascension following a refueling outage.
- 21 charging pump is in service.
- An RCS leak is identified, and operators enter S2.OP-AB.RC-0001, Reactor Coolant System Leak.
- After reducing letdown flow to minimum, and raising charging flow to maximum, the following indications are present:
  - PZR level is dropping ~ 0.1% every 45 seconds.
  - 22 RHR sump pump run alarm is locked in.

Which of the following choices describes the proper course of action for these conditions, and why?

a. Trip the Main Turbine and enter S2.OP-AB.TRB-001, Turbine Trip Below P-9. Safety Injection will only be required if VCT level can NOT be maintained above 4%.

b. Trip the Rx and initiate Safety Injection. The leak rate exceeds the CVCS system make-up capacity. Action will be taken in LOCA-6, LOCA Outside Containment to isolate the leak from the RHR system.

c. Isolate letdown to establish rising PZR level and place the CVCS Make-Up system in MANUAL to raise VCT level. This will ensure PZR level can be maintained above 11% and preclude having to initiate a Safety Injection.

d. Place a second centrifugal charging pump in service. Continue in AB.RC-0001 to perform leak identification and isolation steps. Initiate a shutdown to comply with the actions of TSAS 3.4.7.2 Reactor Coolant System Operational Leakage.

Answer	b	Exam Level	R	Cognitive Level	Application	Facility:	Salem 1 & 2	Exam Date:	12/11/2006

Tier:	Emergency and Abnormal Plant Evolutions	RO Group	1	SRO Group	1	00WE04A202
E04	LOCA Outside Containment				Record Number	21

EA2.	Ability to determine and interpret the following as they apply to LOCA Outside Containment:		
EA2.2	Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.	3.6	4.2

Explanation of Answer: RCS leakage in excess of ONE centrifugal charging pump with minimum letdown, will require a Rx trip and SI. The RHR pump sump runs indicate the leak is in RHR.

Reference Title
LOCA Outside Containment
Reactor Coolant System Leak

Learning Objectives	
LOCA06E007	Determine a discrete path through the EOP for LOCA OUTSIDE CONTAINMENT
ABRC01E006	In accordance with NRC IN 92-36, "Intersystem LOCA Outside Containment": A. Describe an Intersystem LOCA (ISLOCA) B. Describe the conditions under which an ISLOCA is expected to occur C. State which ISLOCA category is of greatest concern, and the two primary reasons for this concern

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

Given the following conditions:

- Unit 2 reactor has tripped.
- Multiple SGs are faulted.
- ALL AFW pumps have tripped or failed to start.
- RCS Bleed and Feed has been initiated.
- All SG WR levels have dropped to 7% WR.
- CETs are STABLE.
- Containment pressure is 0.5 psig.

Electricians have reported that a spare breaker of the same rating has been installed in 21 AFP cubicle and is ready to be shut. The SM concurs with starting 21 AFP.

When establishing flow to available SGs, which of the following describes the AFW feed strategy to restore SG levels?

Initiate AFW flow...

a. at maximum rate until WR level is greater than 15%, then feed at desired rate to recover levels into the NR.

b. at maximum rate until WR level is greater than 11%, then feed at desired rate to recover levels into the NR.

c. at 1.0 - 5.0 E4 lbm/hr until WR level is greater than 15%, then feed at desired rate to recover levels into the NR.

d. at 1.0 - 5.0 E4 lbm/hr until WR level is greater than 11%, then feed at desired rate to recover levels into the NR.

Answer	d	Exam Level	R	Cognitive Level	Comprehension	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Emergency and Abnormal Plant Evolutions	RO Group	1	SRO Group	1				00WE05A101
E05	Loss of Secondary Heat Sink							Record Number	22
EA1.	Ability to operate and / or monitor the following as they apply to Loss of Secondary Heat Sink:								
EA1.1	Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.							4.1	4.0

Explanation of Answer

The concern with feeding hot SGs is thermal shocking the tubes with cold (70 deg) AFW. The FRP will check the status of the RCS (CET's rising?) to see if the AFW flow is effective at reducing RCS temp during natural circ. It will also check SG WR level for enough inventory to effectively heat up the cold AFW in the downcomer region before it hits the tube sheet. In the conditions given in the stem, the CETs are NOT rising, and WR level is <11%, so feed flow must be initiated with caution, at a rate of 1-5E4 lbm/hr, to prevent shocking the SG tubes until WR level is recovered. This is identified as 11% in the FRHS. 15% is the adverse number used if Containment pressure is >4 psig, which it is not. Distracter b is incorrect because adverse containment numbers are not in effect. Distracters c and d ares incorrect because if substantial level is not present in SG, then thermal shocking IS a concern and feeding will NOT be at maximum rate.

Reference Title

Loss of Heat Sink Functional Recovery

Learning Objectives

FRHS00E009	Determine the indications that are monitored to ensure proper system/component operation for each step in 2-EOP-FRHS-1 thru 5

Material Required for Examination			
Question Source:	Facility Exam Bank	**Question Modification Method:**	Significantly Modified
Question Source Comments:	VISION Q50399 modified from how much and why feeding, to how much until to better match K/A.		

Given the following conditions:

- Unit 2 was operating at 100% power.
- A small break LOCA occurred.
- The reactor has tripped and SI has been initiated.
- Numerous ECCS components did not start/reposition as required.
- FRCC-2, "Response to Degraded Core Cooling", is entered.

You have been directed to place SI Valves in Safeguards position using Table A, Safeguards Valve Alignment.

Which ONE (1) of the following sets of valves should have automatically opened upon receipt of an SI signal?

a. 2CV40 AND 2CV41, VCT ISOLATION VALVES.

b. 2CV68 AND 2CV69, CHARGING LINE ISO VALVES.

c. 21SJ40 AND 22SJ40, HOT LEG INJECTION VALVES.

d. 2SJ12 and 2SJ13, BIT OUTLET ISOLATION VALVES.

Answer	d	Exam Level	R	Cognitive Level	Memory	Facility:	Salem 1 & 2
Exam Date:	12/11/2006						

Tier:	Emergency and Abnormal Plant Evolutions	RO Group	2	SRO Group	2

00WE06K201

E06	Degraded Core Cooling	Record Number	23
EK2.	Knowledge of the interrelations between Degraded Core Cooling and the following:		
EK2.1	Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.	3.6	3.8

Explanation of Answer: Distracter a is incorrect because CV40 and 41 close on a SI signal. Distracter b is incorrect because they are automatically closed. Distracter c is incorrect because the hot leg injection valves are manually opened to establish hot leg injection/recirc. D is correct because the BIT outlet valves are normally closed at power and receive an open signal on SI.

Reference Title

Response to Degraded Core Cooling

Learning Objectives

FRCC00E005	Determine the indications that are monitored to ensure proper system/component operation for each step in the following: A. EOP-CFST-1, Figure 2 B. 2-EOP-FRCC-1 C. 2-EOP-FRCC-2 D. 2-EOP-FRCC-3

Material Required for Examination			
Question Source:	Other Facility	Question Modification Method:	Editorially Modified
Question Source Comments:	Robinson 2, 9/27/2004 NRC Exam		

Which of the following is the reason why the PZR PORVs are closed regardless of PZR pressure during performance of FRCC-3, Response To Saturated Core Cooling?

a. To terminate the unwarranted flow of RCS inventory.

b. Open PORVs are the only way to reach saturation with a constant RCS temperature.

c. Exit from FRCC-3 can only be obtained with the PORVs closed and PZR pressure rising.

d. The PORVs are the only source of pressure reduction NOT addressed by higher priority FRPs.

Answer	a	Exam Level	R	Cognitive Level	Memory	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Emergency and Abnormal Plant Evolutions	RO Group	2	SRO Group	2				00WE07K303
E07	Saturated Core Cooling							Record Number	24
EK3.	Knowledge of the reasons for the following responses as they apply to Saturated Core Cooling:								
EK3.3	Manipulation of controls required to obtain desired operating results during abnormal, and emergency situations.							3.8	3.6

Explanation of Answer: A is correct. The Core Cooling Bases document for Step 6 tells operators to close any PORVs that are open. There is no pressure check. The distracters are all false.

Reference Title

Response to Saturated Core Cooling Conditions

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

Material Required for Examination:

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

Question Source Comments:

Given the following conditions:

- Unit 1 initiated a Rx trip and SI due to a LBLOCA.
- 11 RHR pump is C/T.
- Operators have transitioned to EOP-LOCA-5, Loss of Emergency Recirculation, when the 12SJ44, RHR Pump Suction From Containment Sump, failed to open when required.

Which of the following choices identifies a possible reason why 12SJ44 has not opened?

a. Containment sump level is <62%.

b. 12SJ49, RHR CL INJECTION, is open.

c. The AUTO ARM PB was not depressed first.

d. The 12RH4, RHR PUMP SUCTION from RWST is not closed.

Answer	d	Exam Level	R	Cognitive Level	Memory	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Emergency and Abnormal Plant Evolutions	RO Group	1	SRO Group	1				00WE11K201
E11	Loss of Emergency Coolant Recirculation							Record Number	25
EK2.	Knowledge of the interrelations between Loss of Emergency Coolant Recirculation and the following:								
EK2.1	Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.							3.6	3.9

Explanation of Answer: A is incorrect because Containment sumplevel is a procedural requirement. The only way to open the 12SJ44 is to press the open PB. The stem states that it did not open when required. Required can only mean when it was demanded to open.Unit 1 does not have the AUTO swapover feature. B is incorrect because there is no interlock between the SJ44 and the SJ49. The procedure closes the SJ49 before opening the SJ44, but no interlock. C is incorrect because there is no auto swapover on Unit 1. D is correct because the RHR suction from the RWST has to be sHUT before the RHR suction from the containiment sump can be opened.

Reference Title

Loss of Emergency Recirculation

Transfer to Cold Leg Recirculation

Learning Objectives

LOCA05E005	Determine the indications that are monitored to ensure proper system/component operation for each step in LOSS OF EMERGENCY RECIRCULATION
LOCA05E002	Describe the plant response to EOP actions taken for LOSS OF EMERGENCY RECIRCULATION

Material Required for Examination			
Question Source:	Other Facility	Question Modification Method:	Significantly Modified
Question Source Comments:	Prairee Island 4/23/2004 NRC exam		

During the performance of LOSC-2, Multiple Steam Generator Depressurization, the following plant condition exists:

- Cooldown rate of the RCS is greater than 100F/hour.

How is the control room crew directed to control feedwater flow?

Feedwater flow is...

a. maximized to all S/Gs until narrow range level in any SG is >9%.

b. maintained at least 22E4 lbm/hr total until any SG narrow range is >9%.

c. terminated to all but a single intact S/G, which is fed at no less than 1E4 lbm/hr.

d. reduced to no less than 1E4 lbm/hr to each S/G with narrow range level less than 9%.

Answer	d	Exam Level	R	Cognitive Level	Application	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Emergency and Abnormal Plant Evolutions	RO Group	1	SRO Group	1				00WE12A102
E12	Uncontrolled Depressurization of all Steam Generators							Record Number	26

EA1.	Ability to operate and / or monitor the following as they apply to Uncontrolled Depressurization of all Steam Generators:		
EA1.2	Operating behavior characteristics of the facility.	3.6	3.7

Explanation of Answer: Distracters a and b are incorrect because the feed rate is minimized to keep the tubes wet while minimizing the RCS cooldown. Distracter c is incorrect because ALL SGs are fed at nlt 1E4. D is correct because with a cooldown rate > 100 degrees per hour, feed flow is reduce to nlt 1E4 lbm/hr (minimum measurable feed flow indication corresponding to 25 gpm)

Reference Title

Multiple Steam Generator Depressurization

Learning Objectives

LOSC02E003	Determine the indications that are monitored to ensure the proper operation of systems and components in 2-EOP-LOSC-2

Material Required for Examination			
Question Source:	Other Facility	Question Modification Method:	Editorially Modified
Question Source Comments:	Indian Point NRC Exam 3/10/2003, modified to Salem procedure title and AFW flow units.		

Given the following conditions:

- Unit 2 has been tripped due to a secondary system malfunction.
- Operators are performing actions in EOP-TRIP-2, Reactor Trip Response.
- The CRS elects to enter FRHS-2, Steam Generator Overpressure, for a YELLOW PATH on the Heat Sink Status Tree.

Which of the following contains ONLY conditions which would allow steam release from the affected SG after entering FRHS-2?

a. Affected SG pressure is 1130 psig; NR level is 93%.

b. Affected SG pressure is 1140 psig; NR level is 77%.

c. Affected SG pressure is 1110 psig; NR level is 100%.

d. Affected SG pressure is 1090 psig; NR level is 68%.

Answer	b	Exam Level	R	Cognitive Level	Memory	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Emergency and Abnormal Plant Evolutions	RO Group	2	SRO Group	2				00WE13A202
E13	Steam Generator Overpressure							Record Number	27

EA2. Ability to determine and interpret the following as they apply to Steam Generator Overpressure:

EA2.2 Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments. 3.0 3.4

Explanation of Answer: Distracter a is incorrect because the NR level is >92%. Step 4 of FRHS-2 asks if NR level is less than 92%, at which point a transition to FRHS-3 High Steam Generator Level would be made. B is correct because pressure is above 1125 psig, the highest SG safety valve setpoint, and NR level is below 92%. Distracter c is incorrect because pressure is less than 1125 psig, and NR level is >92% and a transition would be made back to procedure in effect. Distracter d is incorrect because pressure is less than 1125 psig.

Reference Title

Steam Generator Overpressure

Learning Objectives

FRHS00E009 Determine the indications that are monitored to ensure proper system/component operation for each step in 2-EOP-FRHS-1 thru 5

Material Required for Examination:

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

Question Source Comments:

Given the following conditions:

- Unit One is in Mode 5, and has just started a heatup to NOT/NOP.
- 11 CVCS HUT level is 70%.
- 13 CVCS HUT level is 12% and in service.

Using the attached tank curves and assuming 50,000 gallons will be letdown from the RCS to the CVCS HUT's, what will be the final level of the CVCS HUTs when the RCS heatup to NOT/NOP is complete?

	11 CVCS HUT	13 CVCS HUT
a.	70%	84%
b.	70%	100%
c.	80%	90%
d.	80%	74%

Answer a **Exam Level** R **Cognitive Level** Application **Facility:** Salem 1 & 2 **Exam Date:** 12/11/2006

Tier: Plant Systems **RO Group** 2 **SRO Group** 2 002000K407

002 Reactor Coolant System **Record Number** 28

K4. Knowledge of Reactor Coolant System design feature(s) and or interlock(s) which provide for the following:

K4.07 Contraction and expansion during heatup and cooldown 3.1 3.5

Explanation of Answer

Distracter b will be the result if x and y axis are reversed. Distracter C will be the result if x and y axis are reversed and stop filling I/S tank at high alarm setpoint of 90% (provided on graph). Distracter D is a balanced distracter to make 2 80% choices and 2 choices without multiples of 10 in the second part of the choice. IOP-2 step 5.1.14 states that 50,000 gallons of capacity is required for RCS heatup. Initial conditions of 12%=8,000 gallons, +50,000 gallons = 58,000 gallons = 84% from tank curve for CVCS HUT.

Reference Title

Tank Curves

Cold Shutdown to Hot Standby

Learning Objectives

IOP002E002	detemine if precautions and/or prerequisites are met to perform a plant heatup.
CVCS00E013	LOR NCT Discuss the procedural requirements associated with the Chemical and Volume Control System, including an explanation of major precaution and limitations in the Chemical and Volume Control System procedures.

Material Required for Examination S2.OP-TM.ZZ-0002, Rev. 7, Page 7 of 33 CVCS HUT curve

Question Source: Facility Exam Bank **Question Modification Method:** Direct From Source

Question Source Comments: VISION Q50410

Given the following conditions:

- A steam generator tube rupture occurred on Unit 2 from 80% power.
- The control room operators have tripped the plant and initiated an SI.
- Operators are performing a controlled cooldown of the RCS per EOP-SGTR-1 and are NOT at the target temperature.

The following indicated parameters are present:

- Ruptured SG pressure is 985 psig and stable.
- Ruptured SG level is 42% and rising.
- PZR level is 5% and lowering.
- All RCP's are operating.
- RCS pressure is 1300 psig and lowering slowly.
- RCS subcooling is 25°F and rising.
- High alarms are standing on R15 and the affected R19.

Select the proper crew action for the given conditions.

a. Verify ECCS flow established and trip the RCP's.

b. Maintain operation of the RCP's and continue the RCS cooldown per EOP-SGTR-1.

c. Trip the RCP's and transition to EOP-SGTR-3, SGTR with LOCA-Subcooled Recovery.

d. Immediately stop the cooldown and depressurization, and return to EOP-SGTR-1 step 1.

Answer	b	Exam Level	R	Cognitive Level	Memory	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Plant Systems	RO Group	1	SRO Group	1				003000G420
003	Reactor Coolant Pump System							Record Number	29
2.4	Emergency Procedures / Plan								
2.4.20	Knowledge of operational implications of EOP warnings, cautions, and notes.							3.3	4.0

Explanation of Answer

Distracter a is incorrect because the stem states that the cooldown depressurization is in progress, so the SGTR-1 CAS item to trip RCPs is NOT in effect. Distracter d is incorrect because the cooldown is required to be continued to target temperature. The standing rad monitor signals are due to the ruptured generator, not from intact SGs. Distracter c is incorrect because subcooling is adequate per the stem, so transition to SGTR-3 won't be necessary.

Reference Title

Steam Generator Tube Rupture

Steam Generator Tube Rupture Basis Document

Learning Objectives

SGTR01E008	Describe special background information for the following steps in 2-EOP-SGTR-1: Steps 6, 6.1, 7.3, 8, 9, 13, 15, 15.3, 16.3, 17.2, 21, 25, and 27

Material Required for Examination			
Question Source:	Facility Exam Bank	Question Modification Method:	Direct From Source
Question Source Comments:	VISION Q57357		

Given the following conditions:

- Unit 1 is in MODE 3, at NOP/NOT.
- 11 and 12 CCW pumps are in service.
- 13 CCW pump is O/S and in AUTO.
- A Bus differential fault occurs on 1C 4KV Vital bus.

Which of the following identifies the CCW pumps which will be running 1 minute after the vital bus fault?

a. 11 and 12.

b. 11 and 13.

c. 12 and 13.

d. 11, 12, and 13.

Answer	a	Exam Level	R	Cognitive Level	Memory	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Plant Systems	RO Group	1	SRO Group	1				003000K202
003	Reactor Coolant Pump System							Record Number	30
K2.	Knowledge of bus power supplies to the following:								
K2.02	CCW pumps							2.5*	2.6*

Explanation of Answer: 11-13 CCW pumps are powered from A,B, and C 4kv vital busses respectively. A bus differential will cause the bus to remain deenergized. A single 4KV vital bus being deenergized will not affect the other 2 powered vital busses. 11 and 12 pumps will remain running. Distracters are incorrect because 13 CCW pump has no power.

Reference Title

No. 1 Unit 4160V Vital Busses One Line

Learning Objectives

CCW000E005	NCT State the power supply to the following Component Cooling Water System components: Component Cooling Water Pumps CC-16s, CC-136, CC-187, CC-118, CC-190, CC-131 Radiation Monitors

Material Required for Examination

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

Question Source Comments:

Given the following conditions:

- Unit 1 is operating at 100% power.
- Operators are transferring CVCS Letdown from 1CV4, LETDOWN ORIFICE ISOLATION VALVE, to 1CV3, LETDOWN ORIFICE ISOLATION VALVE, IAW S1.OP-SO.CVC-0001, CHARGING, LETDOWN, AND SEAL INJECTION.
- 1CV18, LETDOWN PRESSURE CONTROL VALVE, is in MANUAL.

As the 1CV3 is opened, which of the following identifies how the 1CV18 will need to be adjusted, and how the 1CC71, LTDWN HX CC CONT V will respond as letdown flow changes?

The 1CV18 will be...

a. throttled OPEN as pressure rises, and the 1CC71 modulates open in response to higher temperature.

b. throttled CLOSED as pressure rises, and the 1CC71 modulates open in response to higher temperature.

c. throttled OPEN as pressure lowers, and the 1CC71 modulates closed in response to lower temperature.

d. throttled CLOSED as pressure lowers, and the 1CC71 modulates closed in response to lower temperature.

Answer	a	Exam Level	R	Cognitive Level	Application	Facility:	Salem 1 & 2	Exam Date:	12/11/2006

Tier:	Plant Systems	RO Group	1	SRO Group	1	004000A405
004	Chemical and Volume Control System				Record Number	31
A4.	Ability to manually operate and/or monitor in the control room:					
A4.05	Letdown pressure and temperature control valves				3.6	3.1

Explanation of Answer

A is correct because as more system (RCS) pressure is felt as the 2nd orifice is opened, the Pressure Control Valve must be opened to reduce pressure to maintain at NOP of 300 psig. As the letdown flow rises, the temperature control valve must modulate open to maintain setpoint temperature of 100 degrees. The distracters are all incorrect combination of directions for valve movement and system pressure and temperature changes.

Reference Title

CVCS System

Component Cooling System

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
CVCS00E008	LOR Identify and describe the Control Room controls, indications, and alarms associated with the Chemical and Volume Control System, including: a) The Control Room location of Chemical and Volume Control System control bezels and indications (N/A NEO) b) The function of each Chemical and Volume Control System Control Room control and indication (N/A NEO) c) The effect each Chemical and Volume Control System control has upon Chemical and Volume Control System components and operation (N/A NEO) d) The plant conditions or permissives required for Chemical and Volume Control System Control Room controls to perform their intended function e) The setpoints associated with the Chemical and Volume Control System control room alarms.

Material Required for Examination

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

Question Source Comments:

Given the following conditions:

- Unit 1 has experienced a LBLOCA.
- 11 RHR pp is C/T.
- While attempting to transfer to Cold Leg Recirc IAW 1-EOP-LOCA-3, neither the 11SJ113 SI CHG PUMP X-OVER VALVE, NOR the 12SJ113 SI CHG PUMP X-OVER VALVE can be opened.

What effect will this have on the operation of the CVCS pumps?

a. 11 and 12 CVCS pumps will continue to operate with their discharge aligned to the four RCS hot legs.

b. 11 and 12 CVCS pumps will continue to operate with their discharge aligned to the four RCS cold legs.

c. One of the CVCS pumps will have to be stopped to prevent runout of the only operating RHR pump.

d. BOTH CVCS pump will have to be stopped since their suction cannot be aligned to the discharge of 12 RHR pump.

Answer	b	Exam Level	R	Cognitive Level:	Application	Facility:	Salem 1 & 2	Exam Date:	12/11/2006

Tier:	Plant Systems	RO Group	1	SRO Group	1	005000K104

005	Residual Heat Removal System	Record Number	32

K1.	Knowledge of the physical connections and/or cause-effect relationships between Residual Heat Removal System and the following:

K1.04	CVCS	2.9	3.1

Explanation of Answer	B is correct because 12SJ45 is open between 12 RHR pump and the CVCS pump suction. A is incorrect because the CVCS pumps are aligned to the cold legs. C is incorrect because the affected SJ49 will be shut to prevent runout. (step 22 LOCA-3) D is incorrect because of the same reason that B is correct.

Reference Title
Transfer to Cold Leg Recirculation

Learning Objectives	
LCA3U1E004	Determine the indications that are monitored to ensure proper system/component operation for each step in 2-EOP-LOCA-3

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

Given the following conditions:

- Salem Unit 2 has experienced a LBLOCA.
- All equipment functioned properly EXCEPT 21 RHR pump, which seized 8 hours after SI was initiated. It will take 3 days to repair.
- After consultation with the TSC, 21SJ45 RHR TO SI PMPS STOP VALVE was closed, and no other operational action related to 21 RHR pump trip has been taken.

Which of the following identifies the lineup which will be present AFTER the transfer to Hot Leg Recirc is complete?

a. Containment spray header flow through 21CS36 RHR CS STOP VALVE, Hot Leg Recirc from 21 SI pump through 21SJ40 SJ HDR STOP VALVE, Cold Leg Recirc through 22 RHR pump and 22SJ49 RHR DISCH TO COLD LEGS.

b. Containment spray header flow through 22CS36 RHR CS STOP VALVE, Hot Leg Recirc from 21 SI pump through 21SJ40 and 22SJ40, Cold Leg Recirc through 22 RHR pump and 21SJ49 RHR DISCH TO COLD LEGS.

c. NO flow through EITHER containment spray headers, Hot Leg Recirc from 22 SI pump through 22SJ40 SJ HDR STOP VALVE, Cold Leg Recirc through 22 RHR pump and 22SJ49 RHR DISCH TO COLD LEGS.

d. NO flow through EITHER containment spray headers, Hot Leg Recirc from 22 SI pump through 21SJ40 and 22SJ40, Cold Leg Recirc through 22 RHR pump and 21SJ49.

Answer	c	Exam Level	R	Cognitive Level	Application	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Plant Systems	RO Group	1	SRO Group	1				005000K112
005	Residual Heat Removal System							Record Number	33

K1.	Knowledge of the physical connections and/or cause-effect relationships between Residual Heat Removal System and the following:		
K1.12	Safeguard pumps	3.1	3.4

Explanation of Answer

There will be no recirc flow through 21SJ49 because the RH19's were shut in LOCA-3
Also, there will be no spray flow through either spray header because 21CS36 is closed since 21 RHR pump isn't running, and 21CS36 was never opened.
The SI Hot Leg recirc is only through one SI pump, 22, and its associated SJ40.

Reference Title

Transfer to Hot Leg Recirculation

Transfer to Cold Leg Recirculation

Learning Objectives

LOCA04E003	Determine the indications that are monitored to ensure proper system/component operation for each step in 2-EOP-LOCA-4
LCA3U2E004	Determine the indications that are monitored to ensure proper system/component operation for each step in 2-EOP-LOCA-3

Material Required for Examination			
Question Source:	Facility Exam Bank	Question Modification Method:	Editorially Modified
Question Source Comments:	VISION Q55343		

Given the following conditions:

- Unit 2 has experienced a LOCA.
- The reactor was tripped and MANUAL SI initiated sucessfully 10 minutes ago.
- Containment pressure peaked at, and is stable at, 3.9 psig.
- Operators have just entered EOP-LOCA-1, Loss of Reactor Coolant.

Which of the following choices identifies an effect if ECCS injection flow were lost by inadvertent closing of the 2SJ69, RHR SUCT FROM RWST?

a. Adverse containment conditions will exist. The criteria for SI flow reduction become less restrictive.

b. An AUTOMATIC SI will be initiated from the High Containment pressure signal at 4 psig since the initial SI was MANUALLY intiated.

c. Containment equipment is subjected to a harsher environment. A higher level of instrument error causes indicated subcooling to lower.

d. Control air to the Containment will be isolated when the 21/22CA330 CONT SUP INLET VALVES close on the Phase A isolation signal.

Answer	c	Exam Level	R	Cognitive Level	Comprehension	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Plant Systems	RO Group	1	SRO Group	1				006000K303
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:								
K3.03	Containment							4.2	4.4

Explanation of Answer

A is incorrect because the requirement to use different numbers when containment pressure is > 4 psig (Adverse Containment) only is there because of greater instrument inaccuracies. The criteria haven't changed, only the numbers associated with a certain amount. For instance the level requirement for PZR level is 11% normally to allow transition to TRIP for SI flow reduction, and 19% Adverse. The MASS is the same for those 2 numbers, just the uncertainty has changed to require a higher reading to ensure that required MASS is present in the PZR. Also, 19%(adverse) is MORE restrictive than 11% (normal) is you look at it strictly from an absolute value point of view. B is incorrect because there is no second SI, manual or automatic following SI initiation. C is correct because the Adverse Containment values are AUTOMATICALLY inserted into the SMM. D is incorrect because the Phase A isolation would have been reset following SI reset, and an automatic Phase A isolation on the Hi containment pressure will not occur.

Reference Title

Loss of Coolant Accident

Learning Objectives

CONTMTE008	State the setpoints, coincidence, blocks and permissives for automatic actuations associated with the Containment and Containment Support Systems including: (Licensed Operator & STA only) Containment Isolation Actuation Containment Ventilation Isolation Actuation

Material Required for Examination

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

Question Source Comments:

The Pressurizer Relief Tank (PRT) rupture disks are designed to rupture at ______ psig and their discharge is ________ .

a. 10; hard piped to the containment sump.

b. 100; hard piped to the containment sump.

c. 10; released directly to the containment atmosphere.

d. 100; released directly to the containment atmosphere.

Answer	d	**Exam Level**	R	**Cognitive Level**	Memory	**Facility:**	Salem 1 & 2	**Exam Date:**	12/11/2006
Tier:	Plant Systems	**RO Group**	1	**SRO Group**	1				007000K101
007	Pressurizer Relief Tank/Quench Tank System							**Record Number**	35

K1.	Knowledge of the physical connections and/or cause-effect relationships between Pressurizer Relief Tank/Quench Tank System and the following:		
K1.01	Containment system	2.9	3.1

Explanation of Answer: The PRT rupture disks are not hard piped. They discharge directly to the containment atmosphere at a rupture pressure of 100 psig.

Reference Title

Reactor Coolant

Learning Objectives

PZRPRTE009	NCT State the setpoints, coincidence, blocks and permissives for automatic actuations associated with the Pressurizer and Pressurizer Relief Tank.

Material Required for Examination

Question Source: Other Facility

Question Modification Method: Concept Used

Question Source Comments: Davis Besse 5/10/2004 NRC Exam.

During normal power operations, which of the following describes how Pressurizer Relief Tank (PRT) temperature is reduced if required IAW S2.OP-SO.PZR-0003, Pressurizer Relief Tank operation?

a. Ensure open 2NT25 PRT N2 SUPPLY, then open 2PR14 PRT DRN to establish a drain path. Start a Primary Water Pump and verify 2WR80 and 2WR82 PW TO CONTMT STOP VALVES cycle to maintain 54-87% during draining/cooling.

b. Open the 2WR80 and 2WR82, start a Primary Water Pump, and fill the PRT to the Hi level alarm (89%). Secure the Primary water pump and open the 2PR14 until the lo level alarm is reached (56%), then close the PR14. Repeat sequence as needed.

c. Open the 2PR14 to start the RCDT pumps, and open the 2PR15, PRT VENT TO RCDT VENT HDR, which will start the gravity feed through the normally open 2WR80 and 2WR82.

d. Establish gravity feed and bleed from the PWST by opening 2WR80 and 2WR82, and opening 2PR14.

Answer	d	Exam Level	R	Cognitive Level	Memory	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Plant Systems	RO Group	1	SRO Group	1				007000K401
007	Pressurizer Relief Tank/Quench Tank System							Record Number	36

K4.	Knowledge of Pressurizer Relief Tank/Quench Tank System design feature(s) and or interlock(s) which provide for the following:		
K4.01	Quench tank cooling	2.6	2.9

Explanation of Answer: B is correct because gravity feed and bleed of the tank is performed at Step 5.3, Reducing PRT Temperature by Feed and Bleed. A is incorrect because a Primary Water pump is not required, and the possibility of water hammer is pointed out in NOTE prior to step 5.3.1. C is incorrect because the WR80 and 82 are normally closed valves. D is incorrect because WR80 and 82 do not cycle on PRT level.

Reference Title
Pressurizer Relief Tank Operation
Reactor Coolant
Chemical and Volume Control Primary Water Recovery

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:			

Given the following conditions:

- Unit 1 is operating at 100% power.
- Pressurizer level is dropping slowly.
- CCW Surge tank level is rising slowly.
- Radiation Monitor R17A, CCW Process Radiation Monitor is rising.

Which of the following identifies the component which is the source of in-leakage to the CCW system, and what action(s) will prevent the release of radiation to the atmosphere?

a. RHR Heat Exchanger; 1R41D will swap Aux Bldg Exh ventilation to HEPA plus Charcoal in service.

b. RCP Thermal barrier heat exchanger; 2CC149 Surge Tank Vent Valve will auto close on rising radiation.

c. RCP seal water return heat exchanger; 2CC149 Surge Tank Vent Valve will auto close on rising radiation.

d. Spent fuel pool cooling heat exchanger; 1R41D will swap Aux Bldg Exh ventilation to HEPA plus Charcoal in service.

Answer	b	Exam Level	R	Cognitive Level	Application	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Plant Systems	RO Group	1	SRO Group	1				008000A204
008	Component Cooling Water System							Record Number	37

A2.	Ability to (a) predict the impacts of the following on the Component Cooling Water System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:		
A2.04	PRMS alarm	3.3	3.5

Explanation of Answer

A is incorrect because at 100% power, the RHR HXs will be isolated from CCW by the normally closed CC16 valves. Also, the R41D does NOT auto swap ABV, the AB.RAD has operators place it in that condition. B is correct because the Thermal Barrier is exposed to full seal injection pressure, any leak in the thermal barrier would be into the CC system. C is incorrect because CCW pressure is higher than seal return pressure, and any leakage would be out of the CCW system. D is incorrect for the same reason.

Reference Title

Abnormal Radiation

Learning Objectives

ABRAD1E001	Describe the operation of radiation monitors as applied to S2.OP-AB.RAD-0001(Q): a) Radiation monitor response to high radiation; including actions that occur as a result of the channel in warning or alarm.

Material Required for Examination

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

Question Source Comments:

Which of the following choices describes an evolution which will require the GREATEST magnitude (in percent from normal) of correction signal for the PZR Master Pressure Controller to return PZR pressure to normal?

a. A 1,000 gallon continuous dilution at 90% power @ EOL.

b. PZR goes solid after inadvertent SI with ALL RCPs tripped.

c. A single RCP trips while in MODE 3 with rod control deenergized.

d. A single Main Turbine Governor Valve fails shut in one second at 100% power.

Answer	d	Exam Level	R	Cognitive Level	Comprehension	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Plant Systems			RO Group	1	SRO Group	1		010000K602
010	Pressurizer Pressure Control System							Record Number	38

K6.	Knowledge of the of the effect of a loss or malfunction on the following will have on the Pressurizer Pressure Control System:		
K6.02	PZR	3.2	3.5

Explanation of Answer

A is incorrect because at EOL, with very little boron in the core, diluting has a much smaller effect on RCS temp/power/pressure. A 1,000 gallon dilution at the normal flowrate of 62 gpm will take 16 minutes to inject. Using figures 13 and 101 of the REM, the total amount of reactivity added by the 1,000 gallon dilution is 13 pcm. B is incorrect because the SI will isolate control air to the containment. With no air available to operate the spray valves, AND no motive force for the sprays, (RCPs tripped), the PZR pressure control system can NOT return pressure to normal, even though it will have a 100% pressure reduction signal applied to it. C is incorrect because the pressure pertubation will be assuaged by the other 3 RCP's. D is correct because the instantaneous (1 sec) downpower will cause a 25% load rejection, heatup, insurge, and pressure rise. The PZR pressure control system will have to respond with a large signal to limit the rise.

Reference Title

Reactor Enginneering Manual

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:			

Given the following conditions:

- Unit 2 is in MODE 3, NOT/NOP.
- The North 13KV bus section 6 becomes deenergized, and remains deenergized.

With NO operator action, which of the following identifies ONLY the Unit 2 PZR heaters which remain available for PZR pressure control?

a. 21 Backup heaters ONLY.

b. 22 Backup heaters ONLY.

c. 21 Backup heaters and Control Group heaters ONLY.

d. 22 Backup heaters and Control Group heaters ONLY.

Answer	b	Exam Level	R	Cognitive Level	Application	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Plant Systems			RO Group	2	SRO Group	2		011000K202
011	Pressurizer Level Control System							Record Number	39
K2.	Knowledge of bus power supplies to the following:								
K2.02	PZR heaters							3.1	3.2

Explanation of Answer: Using drawing 203000-SIMP, when 13KV north ring bus is deenergized, the power to 22 SPT is lost. The Unit Main Generator is not online, so there is no alternate source of power to the F and G 4KV group busses. G bus supplies power to the control group and 21 B/U group of PZR heaters (dwg 601398). This leaves only the 22 B/U heaters powered from E 4KV group bus (601397) available for pressure control. 21 B/U heaters does have a manually transferable power supply to a vital bus, but the question stem specifically says with no operator action. The distracters are wrong because they contain the incorrect heater groups.

Reference Title

Salem 500KV-4KV Electrical Distribution Simplified Oneline

No. 2 Unit Aux Building Penetration Area 2EP-4KV PZR htr. Bus Oneline

No. 2 Unit Aux Building Penetration Area 2GP-4KV PZR htr. Bus Oneline

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:	New	Question Modification Method:	
Question Source Comments:			

Given the following conditions:

- Unit 1 is operating at 100% power.
- SSPS testing is in progress IAW S1.IC-ST.SSP-009, Solid State Protection System Train B Functional Test.
- Rx Trip BYPASS breaker B is racked in and SHUT.
- Rx Trip breaker B is racked in and SHUT.
- OHA A-42, SSPS TRN B TRBL is in alarm as expected.

The 48VDC power supply from B Vital bus to SSPS Train B Logic Cabinet becomes deenergized.

Which of the following describes the impact of this power supply becoming deenergized while in this configuration?

a. The Rx will trip when the UV coils for BOTH Rx Trip Breaker B AND Rx Trip BYPASS breaker become deenergized.

b. The Rx will trip when the shunt trip coils become deenergized for BOTH Rx Trip Breaker B AND Rx Trip BYPASS breaker.

c. The Rx will NOT trip because the 48VDC supplied to the UV coils is powered from the SSPS Output Bay, not the Logic Bay.

d. The Rx will NOT trip because the loss of the auctioneered Logic Bay 48VDC power supply to trip the Rx has not occurred.

Answer	d	Exam Level	R	Cognitive Level	Application	Facility:	Salem 1 & 2	Exam Date:	12/11/2006

Tier:	Plant Systems	RO Group	1	SRO Group	1	012000A207
012	Reactor Protection System				Record Number	40

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:		
A2.07	Loss of dc control power	3.2*	3.7

Explanation of Answer: A is incorrect because the UV coils will remain energized from the redundant 48VDC power supply for RTB B. The bypass breaker receives its UV voltage from the other Train. B is incorrect because the shunt trip coils are energize to operate. C is incorrect because the UV is powered from the logic bay. D is correct because you have to lose both 48VDC power supplies from B and C vital busses to lose power to the UV coils which will trip both the B train breakers.

Reference Title

Reactor Protection System Reactor Trip Signals

Solid State Reactor Protection Train B DC PowerDistribution

Learning Objectives

Material Required for Examination:

Question Source: New

Question Modification Method:

Question Source Comments:

Given the following conditions:

- Unit 1 is performing a reactor startup.
- Initial power PRIOR to the startup was 100 cps.
- Rx power is now 4200 cps in the Source Range.

After performing a rod pull, rod motion continues when the RAISE pushbutton is released.

IAW Salem UFSAR, how is the plant protected from this event with NO operator action, and what will be the effect on the margin to Departure from Nucleate Boiling (DNB)?

a. The Source Range High Neutron Flux Trip will ensure a large margin to DNB is maintained.

b. The Power Range High Neutron Flux Trip (low setting) will ensure a large margin to DNB is maintained.

c. The Source Range High Neutron Flux Trip will terminate the event, but NOT before a minimal margin to DNB is reached.

d. The Power Range High Neutron Flux Trip (low setting) will terminate the event, but NOT before a minimal margin to DNB is reached.

Answer	b	Exam Level	R	Cognitive Level	Memory	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Plant Systems	RO Group	1	SRO Group	1				012000K501
012	Reactor Protection System							Record Number	41
K5.	Knowledge of the operational implications of the following concepts as they apply to the Reactor Protection System:								
K5.01	DNB							3.3*	3.8

Explanation of Answer

Salem UFSAR section 15.2.1.2.4 (assumptions for uncontrolled RCCA withdrawal from subcritical accident) states: "Reactor trip is assumed to be initiated by power range neutron flux (low setting)." 15.2.1.3 states: "There is a large margin to DNB during the transient since the rod surface heat flux remains below the design value, and there is a high degree of subcooling at all times in the core."

Reference Title

Salem Updated Final Safety Analysis Report

Learning Objectives

TAA000E015	Assess TAA conditions that affect heat removal rates

Material Required for Examination

Question Source: Facility Exam Bank

Question Modification Method: Significantly Modified

Question Source Comments: Modified VISION Q48702. Replaced 2 of the distracters with 2 new distracters, (PR Flux trips to SR Flux Trips)

Given the following conditions:

- Unit 1 is operating at 100% power.
- PT-948C, Containment Pressure detector Channel II, fails LOW.

Which of the following describes the Safety Injection and Containment Spray actuation coincidences PRIOR to taking any action?

a. Safety Injection- 2/2; Containment Spray- 1/3

b. Safety Injection- 2/3; Containment Spray- 2/3

c. Safety Injection- 2/2; Containment Spray- 2/3

d. Safety Injection- 1/3; Containment Spray- 1/3

Answer	c	**Exam Level**	R	**Cognitive Level**	Application	**Facility:**	Salem 1 & 2	**Exam Date:**	12/11/2006
Tier:	Plant Systems			**RO Group**	1	**SRO Group**	1		013000K502
013	Engineered Safety Features Actuation System							**Record Number**	42

K5.	Knowledge of the operational implications of the following concepts as they apply to the Engineered Safety Features Actuation System:		
K5.02	Safety system logic and reliability	2.9	3.3

Explanation of Answer

Containment Spray actuation is normally 2/4 Containment Pressure detectors reading 15 psig. These bistables are energize to actuate. If ONE fails LOW, it will still require TWO channels to ACTUATE out of the THREE remaining channels. Safety Injection is normally 2/3 channels reading 4 psig, deenergize to actuate. Channel II indication feeds Bistable Switches 948C for Spray Initiation, AND bistable switch 948F for Safety Injection initiation.

Reference Title

Reactor Prot. & Process Cont. Systems Safety Injection Interconnections

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

Given the following conditions:

- Unit 2 is operating at 100% power.
- A Control Bank control rod drops partially into the core.
- The reactor does NOT trip.

Which of the following describes a condition that will result if the reactor is left in this configuration?

a. The flux in the area of the dropped rod will be depressed and cause Iodine production to rise. Conversely, other areas of the core will see higher flux generation, causing Xenon burnout rate to lower. As iodine levels in the lower flux areas of the core build up, reversal of the initial situation occurs, and a Xenon oscillation is occurring.

b. The Heat Flux Hot Channel factor will rise to a value in excess of the allowed linear heat generation at elevations in the core furthest from the dropped rod. This will cause the Departure from Nucleate Boiling Ratio to drop less than required, and some localized steam blanketing can occur at the extreme edges of the core x-y elevations.

c. The depressed flux in the area of the dropped rod causes Xenon to build in at a higher rate. Conversely, the higher flux in other areas of the core causes Xenon burnout rate to rise. As iodine levels in the higher flux areas of the core build up, reversal of the initial situation occurs, and a Xenon oscillation is occurring.

d. The radial flux tilt will cause certain areas of the core to burn out faster than others, leading to a condition in which an extended coastdown to refueling would be required due to the inability to maintain AFD within the target band without reducing power to less than 90%.

Answer	c	Exam Level	R	Cognitive Level	Comprehension	Facility:	Salem 1 & 2
Exam Date:	12/11/2006						

Tier:	Plant Systems	RO Group	2	SRO Group	2
015000K511					

015	Nuclear Instrumentation System	Record Number	43

K5.	Knowledge of the operational implications of the following concepts as they apply to the Nuclear Instrumentation System:
K5.11	Axial flux imbalance, including long-range effects — 3.3 / 3.7

Explanation of Answer

A is incorrect because it describes the opposite of how flux will be affected. B is incorrect because the concern with hot channel factor is a radial tilt causing a much high flux at certain areas of the core. C is correct because Large thermal reactors with little flux coupling between regions may experience spatial power oscillations because of the non-uniform presence of xenon-135. The mechanism is described in the following four steps. (1) An initial lack of symmetry in the core power distribution (for example, individual control rod movement or misalignment) causes an imbalance in fission rates within the reactor core, and therefore, in the iodine-135 buildup and the xenon-135 absorption. (2) In the high-flux region, xenon-135 burnout allows the flux to increase further, while in the low-flux region, the increase in xenon-135 causes a further reduction in flux. The iodine concentration increases where the flux is high and decreases where the flux is low. (3) As soon as the iodine-135 levels build up sufficiently, decay to xenon reverses the initial situation. Flux decreases in this area, and the former low-flux region increases in power. (4) Repetition of these patterns can lead to xenon oscillations moving about the core with periods on the order of about 15 hours. With little change in overall power level, these oscillations can change the local power levels by a factor of three or more. D is incorrect because it concerns radial flux tilts and coastdown factors.

Reference Title

Tecnhincal Specifications-Bases

Salem UFSAR- Rod Cluster Control Assembly Misalignment

Learning Objectives

RXOPERE019	Explain the effects of control rod motion, boration, and dilution on reactor power.

Material Required for Examination	

Question Source:	New	Question Modification Method:	

Question Source Comments:	

Given the following conditions:

- A small break LOCA has occurred on Unit 1.
- The RVLIS Summary Page is displaying DYNAMIC RANGE.
- During the subsequent cooldown and depressurization, DYNAMIC RANGE indication remained constant at 80% with RCPs running.

Which of the following describes DYNAMIC RANGE response and reason for that response during the cooldown and depressurization?

a. It was lower than 80%. Lowering pressure and rising density creates a RVLIS indication error.

b. It was higher than 80%. Lowering pressure and rising density creates a RVLIS indication error.

c. It was accurate at 80%. RVLIS DYNAMIC RANGE is pressure and temperature compensated.

d. It was accurate at 80%. D/P is an accurate measure of void content, without compensation.

Answer	c	Exam Level	R	Cognitive Level	Comprehension	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Plant Systems			RO Group	2	SRO Group	2		016000A302
016	Non-Nuclear Instrumentation System							Record Number	44
A3.	Ability to monitor automatic operations of the Non-Nuclear Instrumentation System including:								
A3.02	Relationship between meter readings and actual parameter value							2.9*	2.9*

Explanation of Answer	RVLIS is pressure and temperature compensated for accuracy. FSAR Section 5.6.5 page 5.6-6.

Reference Title

Final Safety Analysis Report

Learning Objectives

RVLIS0E008	Identify and describe the Control Room controls, indications, and alarms associated with the Reactor Vessel Level Instrumentation System, including: a) The Control Room location of Reactor Vessel Level Instrumentation System control bezels and indications (N/A NEO) b) The function of each Reactor Vessel Level Instrumentation System Control Room control and indication (N/A NEO) c) The effect each Reactor Vessel Level Instrumentation System control has upon Reactor Vessel Level Instrumentation System components and operation (N/A NEO) d) The plant conditions or permissives required for Reactor Vessel Level Instrumentation System Control Room controls to perform their intended function e) The setpoints associated with the Reactor Vessel Level Instrumentation System control room alarms

Material Required for Examination			
Question Source:	Facility Exam Bank	Question Modification Method:	Direct From Source
Question Source Comments:	VISION Q60901		

Given the following conditions:

- Unit 1 has experienced a Large Break LOCA from 100% power operation.
- Before operators can respond, containment pressure rises to 10 psig.
- Off-site power remains available.

Assuming ALL automatic actions occur as expected, which of the following describes CFCU operation BEFORE operators take any MANUAL actions?

CFCUs running in HIGH speed...

a. receive a simultaneous HIGH speed STOP signal and a LOW speed START signal, all airflow will be directed through the ROUGHING filters ONLY.

b. receive a HIGH speed STOP signal, followed 20 seconds later by a LOW speed START signal, all airflow will be directed through the HEPA filters ONLY.

c. remain in HIGH speed with all airflow directed through the Roughing Filters, all other available CFCUs IMMEDIATELY start in LOW speed with all airflow through the HEPA Filters ONLY.

d. remain in HIGH speed until their respective Vital Bus EDG is up to speed, then receive a HIGH speed STOP signal, followed 5 seconds later by a LOW speed START signal, all airflow will be directed through theHEPA AND ROUGHING filters.

Answer	b	Exam Level	R	Cognitive Level	Memory	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Plant Systems	RO Group	1	SRO Group	1				022000K402
022	Containment Cooling System							Record Number	45

K4.	Knowledge of Containment Cooling System design feature(s) and or interlock(s) which provide for the following:		
K4.02	Correlation of fan speed and flowpath changes with containment pressure	3.1*	3.4*

Explanation of Answer

The stem states that no operator manual action has been taken, and all automatic actions occur as expected. There will be an AUTOMATIC Safety Injection when containment pressure reaches 4 psig, and the stem states cont pressure is 10 psig. This will start the SEC MODE 1 sequencer. In MODE 1, most equipment is immediately loaded onto its vital busses. However, the CFCUs normally operate in HIGH speed. In order to protect the motors when shifting to LOW speed, a 20 second time delay is incorporated into the automatic LOW speed start signal. (See dwg 203673 C-1). The normal air flow path in HIGH speed is through the rouging filter, when the LOW speed breaker is shut, the HEPA filter is placed in service and the roughing filter is removed from service by automatic damper operation.

Reference Title

No. 1& 2 Units Safeguards Emergency Loading Sequence

No. 1& 2 Units Safeguards Emergency Loading Sequence

Learning Objectives

CONTMTE014	Given plant conditions, relate the Containment and Containment Support Systems with the following: Containment Isolation/Containment Integrity Electrical Penetrations Personnel Access Hatch Service Water System Safeguards Equipment Cabinets

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

Question Source Comments:

In addition to pressing the STOP PB on CC1, which ONE of the following identifies ALL required actions for stopping the Containment Spray Pumps following automatic initiation of Containment Spray (CS)?

a. Reset Safety Injection and reset associated SEC.

b. Reset Containment Isolation Phase B Isolation signal ONLY.

c. Reset Safety Injection Signal, then reset Containment Isolation Phase B.

d. Reset Containment Isolation Phase A, ensure containment pressure less than 14 psig (CS and Phase B initiating signal clear), reset associated SEC

Answer	a	Exam Level	R	Cognitive Level	Memory	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Plant Systems	RO Group	1	SRO Group	1				026000A405
026	Containment Spray System							Record Number	46
A4.	Ability to manually operate and/or monitor in the control room:								
A4.05	Containment spray reset switches							3.5	3.5

Explanation of Answer

A is correct because the SEC controls the start signal for the CS pump. Until it is reset, the SEC blocks manual action for the CS pumps. In order to be reset, the SI signal must first be reset. SI can be reset with a standing SI signal input, it is blocked after initiation until the Rx trip breakers have been cycled. Distracter b is incorrect because the Phase B signal is only an AND logic with SEC start signal for STARTING the pump. The pump can be stopped with a standing Phase B signal. Distracter c is incorrect because for the same reason as b. Distracter d is incorrect because Phase A is not associated with Phase B, and there is no automatic Phase B reset.

Reference Title

Reactor Protection System Safeguards Actuation Signals

Reactor Protection System Reactor Trip Signals

Logic Containment Spray System Containment spray Pumps

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:	Other Facility	Question Modification Method:	Significantly Modified
Question Source Comments:	Beaver Valley-2 2002 NRC RO Exam Question 24, Modified correct answer to reflect Salem logic for Cont Spray pump stop and editorial mod to reflect Salem terminology.		

Given the following conditions:

- A LBLOCA has occurred.
- In response to a RED path on the CORE COOLING Critical Safety Function Status Tree, FRCC-1, "Response to Inadequate Core Cooling", is currently in progress.
- Containment hydrogen concentration is 4.5%.

Which of the following states the action that is to be taken in regards to operation of the hydrogen recombiners?

a. Place ONE hydrogen recombiner in service to reduce the hydrogen concentration.

b. Place BOTH hydrogen recombiners in service to reduce the hydrogen concentration.

c. DO NOT operate the hydrogen recombiners since they could result in ignition of the hydrogen.

d. DO NOT operate the hydrogen recombiners since the hydrogen recombiner system will not be effective at this concentration.

Answer	c	Exam Level	R	Cognitive Level	Memory	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Plant Systems			RO Group	2	SRO Group	2		028000A203
028	Hydrogen Recombiner and Purge Control System							Record Number	47

A2.	Ability to (a) predict the impacts of the following on the Hydrogen Recombiner and Purge Control System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:		
A2.03	The hydrogen air concentration in excess of limit flame propagation or detonation with resulting equipment damage in containment	3.4	4.0

Explanation of Answer	In LOCA-1, step 24 asks if containment H2 concentration is between 0.5-4.0 % to place a SINGLE recombiner in service. The operating procedure specifies to place the recombiners in service BEFORE reaching 4% H2. It also states that only ONE is to be run at a time.

Reference Title
Loss of Coolant Accident
Hydrogen Recombiner Operation

Learning Objectives	

Material Required for Examination			
Question Source:	Other Facility	Question Modification Method:	Editorially Modified
Question Source Comments:	4/27/2004 Ginna NRC Exam		

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 rises; 1R41B rises; 1R41D rises.

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

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

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

Answer	d	Exam Level	R	Cognitive Level	Comprehension	Facility:	Salem 1 & 2
Exam Date:	12/11/2006						
Tier:	Plant Systems	RO Group	2	SRO Group	2		029000A102
029	Containment Purge System					Record Number	48
A1.	Ability to predict and/or monitor changes in parameters associated with operating the Containment Purge System controls including:						
A1.02	Radiation levels					3.4	3.4

Explanation of Answer

A and B are incorrect because 1R12A is sampling containment atmosphere, so it will NOT rise when the pressure relief is started. A and C are incorrect because 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. D is correct because of the above and 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

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

Material Required for Examination

Question Source:	Previous 2 NRC Exams	Question Modification Method:	Direct From Source
Question Source Comments:			

Given the following conditions:

- Unit 2 is in MODE 6.
- Fuel movement is in progress in both the containment and Fuel Handling buildings.
- Operators in containment report lowering Rx cavity level.
- Due to a mis-communication, the Fuel Pool Gate valve is closed with the transfer cart in its way.
- The Fuel Pool Gate valve cannot be closed further than 40 turns closed.

Which of the following choices identifies the condition which will happen FIRST if the leak is in the RHR system, with NO other operator action?

a. Fuel in the Rx vessel will become uncovered.

b. RHR pumps will cavitate and become air bound.

c. Fuel in the Spent Fuel Pool racks will become uncovered.

d. The upper and lower Reactor Cavity will completely drain.

Answer	b	**Exam Level**	R	**Cognitive Level**	Application	**Facility:**	Salem 1 & 2	**Exam Date:**	12/11/2006
Tier:	Plant Systems	**RO Group**	2	**SRO Group**	2				033000K102
033	Spent Fuel Pool Cooling System							**Record Number**	49

K1.	Knowledge of the physical connections and/or cause-effect relationships between Spent Fuel Pool Cooling System and the following:		
K1.02	RHRS	2.5	2.7

Explanation of Answer

B is correct because if the leak is from the RHR system, it will draw the RCS level down into the hot legs and suck air into the RHR piping. D is incorrect because the reactor cavity cannot completely drain if the leak is from RHR since the lower cavity is below the level of the vessel flange (104'). A and d are incorrect because once the level drops below the Rx vessel flange, it will not affect SFP level, since the cavity level connected to the SFP through the open Gate Valve will never go below 104'. The bottom of the SFP is on the 89'6" level. The height of all the fuel racks in the SFP is 185 1/4". (VENDOR DWG 316748) These 2 combined is 104'. The spent fuel assembly fits down inside the rack, so it can never become uncovered if 104' of water is in the SFP.

Reference Title

Pool Layout for Spent Fuel Storage Racks

Draining the Reactor Coolant System

Learning Objectives

ABFUE2E001	Describe the operation of the following as applied to S2-OP-AB.FUEL-0002(Q): a) Fuel Pool b) Spent Fuel Cooling c) Refueling Cavity d) S/G Nozzle Dam Seals e) Weir Gate Seals

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

Given the following conditions:

- Unit 2 is at 100% power.
- Surveillance testing is in progress at the turbine front standard.
- A combination of equipment failure and human error causes an automatic SI signal to be generated only on RPS Train A.
- Reactor Trip Breaker A (RTB A) fails to open, and is stuck in the shut position.

With NO operator action, which of the following describes how this affects plant operation?

a. The ATWT allows the Rx to remain critical. AMSAC will trip the Main Turbine, the turbine trip causes the Rx to trip

b. The ATWT allows the Rx to remain critical, and the Main Turbine remains on-line.

c. The SI signal causes the Main Turbine to trip, the turbine trip causes the Rx to trip.

d. The SI signal causes the Rx to trip, the Rx trip causes the Main Turbine to trip.

Answer	d	**Exam Level**	R	**Cognitive Level**	Comprehension	**Facility:**	Salem 1 & 2	**Exam Date:**	12/11/2006
Tier:	Plant Systems	**RO Group**	1	**SRO Group**	1				039000A401
039	Main and Reheat Steam System							**Record Number**	50
A4.	Ability to manually operate and/or monitor in the control room:								
A4.01	Main steam supply. valves							2.9*	2.8

Explanation of Answer

The AUTO SI on Train A causes the RPS system to send a trip signal to RTB A, and Bypass breaker B ONLY. A MANUAL SI, on the other hand, sends trip signals to all 4 trip and bypass breakers, which would trip the reactor even with RTB A stuck shut. In these circumstances, the Rx will still trip, because a signal is sent to trip the Main Turbine. The Main Turbine trip >P-9 will cause a Rx trip, which will send a trip signal to BOTH RTBs. The AUTO SI signal on Train A sends a signal to the AUTO STOP OIL section of the Turbine trips. (221065, D-4) This causes Auto Stop Oil to be dumped from the Main Turbine Stop Valves, and they will shut.

Reference Title

Reactor Protection System Reactor Trip Signals

Reactor Protection System Turbine Trips, Runbacks & Gen Protection

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: New **Question Modification Method:**

Question Source Comments:

Given the following conditions:

- Unit 2 is operating at 100% power.
- 21 SGFP must be removed from service to repair a small steam leak.

Which of the following is the MAXIMUM Rx power which will allow 21 SGFP to be removed from service IAW S2.OP-SO.CN-0002, STEAM GENERATOR FEED PUMP OPERATION?

a. 75%.

b. 66%.

c. 55%.

d. 35%.

Answer	c	Exam Level	R	Cognitive Level	Memory	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Plant Systems		RO Group	1	SRO Group	1			059000A103
059	Main Feedwater System							Record Number	51

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

A1.03 Power level restrictions for operation of MFW pumps and valves. 2.7* 2.9*

Explanation of Answer

The procedure, step 5.6.1 states that total feedwater flow must less than or equal to 9E6 lbm/hr. Also, S2.OP-IO.ZZ-0004, POWER OPERATION, step 5.1.12 states the same limit. At 100% power, total design steam flow is ~15.6E6 lbm/hr. (USFAR Section 10.4, each SG 3.9E6) 9E6/15.6E6=57.7% The actual steam pressures at this power level would be higher, so this is a conservative number. For that reason, c is correct because it is the highest number that is less than 57.7%. Distracter a is incorrect because it is the power level during a normal power reduction at which a condensate pump is removed from service. Distracter b is incorrect because it is the power level at which the automatic Main Turbine runback on a single sGFP trip is automatically disarmed. Distracter d is incorrect because it is the power level at which the feed pump is removed from service IAW IOP-4.

Reference Title

Steam Generator Feed Pump Operation

Power Operation

Learning Objectives

CN&FDWE013	LOR NCT Discuss the procedural requirements associated with the Condensate and Feedwater System, including an explanation of major precaution and limitations in the Condensate and Feedwater System procedures
CN&FDWE015	LOR NCT Given plant conditions, relate the Condensate and Feedwater System with the following: Main Condenser Miscellaneous Condensate System Heater Drain System Gland Seal/Gland Exhaust System Condenser Air Removal System Extraction Steam System Turbine Auxiliary Cooling Auxiliary Feedwater System Main Turbine Control Air System Bleed Steam Demineralized Water System Advance Digital Feedwater Control System Reactor Coolant System Chemical Addition System

Material Required for Examination

Question Source: New | Question Modification Method:

Question Source Comments:

Given the following conditions:

- Unit 1 is operating at 85% power.
- 11 charging pump is in service.
- During a manual bus swap prior to clearing and tagging a Station Power Transformer, the 1A 4KV vital bus is inadvertently deenergized, and the SEC loads 1A bus in Mode 2*.
- All other electrical bus transfers expected to occur from the loss of 1A 4KV vital bus are successful.

With NO operator action, which of the following identifies the plant condition 5 minutes after the initial loss of 1A 4KV vital bus?

a. Reactor power is >85%.

b. PZR level is rising at 1% per minute.

c. The Main Turbine will have run back to 60%.

d. PZR Backup heaters have cycled on due to low pressure and remain ON.

Answer	a	Exam Level	R	Cognitive Level	Application	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Plant Systems	RO Group	1	SRO Group	1				061000A204
061	Auxiliary / Emergency Feedwater System							Record Number	52

A2.	Ability to (a) predict the impacts of the following on the Auxiliary / Emergency Feedwater System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:		
A2.04	pump failure or improper operation	3.4	3.8

Explanation of Answer

A is correct because the addition of cold ~70 deg aux feedwater to the S/G's through the AF21 valves(normally have a 98% demand signal but kept closed by Pressure Override until AFP disch. Pressure rises above 1150 psig) will cause a positive reactivity addition due to temperature of the feedwater entering the S/G's lowering. Reactor power will rise to some value greater than 85.0%. B is incorrect because 11 charging pump is powered from "B" vital bus. If the student thinks letdown isolated because the SEC stripped the operating charging pump during its sequence and that letdown isolated because no charging pumps are running, then rising PZR level would be correct after the charging pump restarted. (87 gpm charging, 12 gpm seal leakoff, 0 gpm letdown, and thumbrule of 1% PZR level = 75 gallons at NOT.) C is incorrect because both Main feedwater pumps will continue to operate, and would not cause a MT runback. D is incorrect because no action has occurred which will lower PZR pressure to the point of energizing the Backup heaters.

Reference Title

Loss of 1A 4KV Vital Bus

Learning Objectives

AFW000E004	NCT Describe the function of the following components and how their normal and abnormal operation affects the Auxiliary Feedwater System: Motor-driven Auxiliary Feedwater Pumps Turbine-driven Auxiliary Feedwater Pump Turbine-driven Auxiliary Feedwater Pump Start-Stop Valve (MS132) Turbine-driven Auxiliary Feedwater Pump Trip Valve (MS52) Turbine-driven Auxiliary Feedwater Pump Speed Control Valve (GOV) (MS53) AFW Pump Alternate Suction Header Supply Valves (AF52s) Motor-driven AFW Pump Recirculation Flow Control Valves (AF140) Motor-driven AFW Pump Discharge Flow Control Valves (AF21) Turbine-driven AFW Pump Discharge Flow Control Valves (AF11)
AFW000E005	NCT State the power supply to the following Auxiliary Feedwater System components: Motor-driven AFW Pumps
AFW000E016	Given a Auxiliary Feedwater System failure, predict the effect of the Auxiliary Feedwater System failure on the following: (License Operator and STA only)

Reactor Coolant System Steam Generators

Material Required for Examination	

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

Question Source Comments:	VISION Q78195, changed 1 distracter from "Reactor has tripped" to distracter D.

Given the following conditions:

- Unit 1 was tripped from 100% power due to a steam leak.
- A MSLI was successful in isolating the leak.
- The PO idles 23 AFW pp, and throttles AFW flow in EOP-TRIP-2 to 6E4lbm/hr to each SG.

Which of the following describes how AFW flow will be affected if 21 AFW pump trips with NO operator action?

AFW flow will remain the same to ____ and ____ SGs, and lower to ____ and ____ SGs

a. 21,22 23, 24.

b. 23,24 21, 22.

c. 21,24 22, 23.

d. 22, 23 21, 24.

Answer	a	Exam Level	R	Cognitive Level	Comprehension	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Plant Systems	RO Group	1	SRO Group	1				061000K602
061	Auxiliary / Emergency Feedwater System							Record Number	53

K6.	Knowledge of the of the effect of a loss or malfunction on the following will have on the Auxiliary / Emergency Feedwater System:		
K6.02	Pumps	2.6	2.7

Explanation of Answer

A is correct because 21& 22 SGs are supplied from 22 AFW pump, and 23 and 24 SGs are supplied from 21 AFW pump. D is incorrect because when 23 AFW pump is taken to idle speed it is running at ~1100 rpm, and its discharge pressure will be ~150-300psig, which is insufficient to provide flow other than to its own oil coolers. Since the distracter says regarless of the relationship, it is not always true.Distracter b is incorrect because AFW flow will not rise. Distracter c is incorrect because AFW flow will not lower. Distracter d is incorrect because of the discharge piping being backwards.

Reference Title

Reactor Trip Response

Auxiliary Feed System Operation

Learning Objectives

AFW000E016	Given a Auxiliary Feedwater System failure, predict the effect of the Auxiliary Feedwater System failure on the following: *(License Operator and STA only)* Reactor Coolant System Steam Generators

Material Required for Examination

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

Given the following conditions:

- 2C EDG is operating in parallel with the 500KV grid for a 24 hour endurance run IAW S2.OP-ST.DG-0014, 2C DIESEL GENERATOR ENDURANCE RUN, following a complete overhaul.
- Cumulative run times for all individual EDG load limits are less than 10% of rated.
- While operating at 2525 KW three hours into the test, the operator mistakenly adjusts 2C EDG speed control resulting in MW loading increasing to 2800 KW.

Which choice describes the consequences, if any, of continued EDG operation at this KW load?

Operation for the remaining 21 hours of the test...

a. will not have any adverse effect on 2C EDG.

b. will result in exceeding the 2 hour load limitation for 2C EDG.

c. will result in exceeding the 30 minute load limitation for 2C EDG

d. will result in exceeding the 2000 hour load limitation for 2C EDG.

Answer	b	Exam Level	R	Cognitive Level	Application	Facility	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Plant Systems			RO Group	1	SRO Group	1		062000A101
062	A.C. Electrical Distribution							Record Number	54

A1. Ability to predict and/or monitor changes in parameters associated with operating the A.C. Electrical Distribution controls including:

A1.01 Significance of D/G load limits — 3.4 — 3.8

Explanation of Answer: The EDG load limitations are maximum of: 2600KW continuous, 2750KW for 2000 hours, 2860KW for 2 hours, and 3100KW for 30 minutes.(Found in the ST for 2C Endurance run, CAUTION between steps 5.1.4 and 5.1.5) With the EDG operating at 2800KW for 21 hours, the EDG will exceed the limit of 2 hours for operation between 2750-2860KW, since the stem stated that the cumulative run time for ALL EDG load limits was <10%, which would be 12 minutes for this limit.

Reference Title

2C Diesel Generator Endurance Run

Learning Objectives

EDG000E012	LOR NCT Given internal or external industry operating experience, review the OE and outline a course of action, which could prevent recurrence.

Material Required for Examination:

Question Source: Facility Exam Bank — **Question Modification Method:** Direct From Source

Question Source Comments: VISION Q63757

Given the following conditions:

- 2C EDG is operating in parallel with the 500KV grid for a 24 hour endurance run IAW S2.OP-ST.DG-0014, 2C DIESEL GENERATOR ENDURANCE RUN, following a complete overhaul.
- Cumulative run times for all individual EDG load limits are less than 10% of rated.
- While operating at 2525 KW three hours into the test, the operator mistakenly adjusts 2C EDG speed control resulting in MW loading increasing to 2800 KW.

Which choice describes the consequences, if any, of continued EDG operation at this KW load?

Operation for the remaining 21 hours of the test…

a. will not have any adverse effect on 2C EDG.

b. will result in exceeding the 2 hour load limitation for 2C EDG.

c. will result in exceeding the 30 minute load limitation for 2C EDG

d. will result in exceeding the 2000 hour load limitation for 2C EDG.

Answer	b	**Exam Level**	R	**Cognitive Level**	Application	**Facility:**	Salem 1 & 2	**Exam Date:**	12/11/2006
Tier:	Plant Systems		**RO Group**	1	**SRO Group**	1			062000A101
062	A.C. Electrical Distribution							**Record Number**	55

A1.	Ability to predict and/or monitor changes in parameters associated with operating the A.C. Electrical Distribution controls including:		
A1.01	Significance of D/G load limits	3.4	3.8

Explanation of Answer: The EDG load limitations are maximum of: 2600KW continuous, 2750KW for 2000 hours, 2860KW for 2 hours, and 3100KW for 30 minutes.(Found in the ST for 2C Endurance run, CAUTION between steps 5.1.4 and 5.1.5) With the EDG operating at 2800KW for 21 hours, the EDG will exceed the limit of 2 hours for operation between 2750-2860KW, since the stem stated that the cumulative run time for ALL EDG load limits was <10%, which would be 12 minutes for this limit.

Reference Title

2C Diesel Generator Endurance Run

Learning Objectives

EDG000E012	LOR NCT Discuss the procedural requirements associated with the Emergency Diesel Generator, including an explanation of major precaution and limitations in the Emergency Diesel Generator procedures

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Direct From Source

Question Source Comments: VISION Q63757

Given the following conditions:

- Units 1 and 2 are operating at 100% power.
- 4KV Vital buses 1A and 1C are powered from 13 SPT. 1B is powered from 14 SPT.
- 4KV Vital buses 2A and 2B are powered from 24 SPT. 2C is powered from 2C EDG running in parallel with the grid.
- All other electric system lineups are normal for full power operation.
- A fault occurs which sends a trip signal to the North 13KV ring bus breaker 1-6, but it does NOT open.

Which of the following describes the effect this will have on the plant with NO Operator action?

a. BOTH Reactors will trip.

b. Main Generator Mwe output will lower.

c. 2C 4KV Vital bus will deenergize and reload in MODE II*.

d. ALL Unit 1 Vital Bus Battery Chargers must be declared INOPERABLE since their power supplies are now ALL powered from the same off-site power supply.

Answer	b	Exam Level	R	Cognitive Level	Application	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Plant Systems			RO Group	1	SRO Group	1		062000K301
062	A.C. Electrical Distribution							Record Number	56

K3.	Knowledge of the effect that a loss or malfunction of the A.C. Electrical Distribution will have on the following:		
K3.01	Major system loads	3.5	3.9

Explanation of Answer

Distracter a is incorrect because the group busses are powered from their respective APTs, so RCPs will not be deenergized and cause a Rx trip. Distracter c is incorrect because with the EDG running, the 2C vital bus will never see a UV condition. B is correct because the loss of 3 circulators per unit will cause MWE output to lower. Distracter d is incorrect because TSAS 3.8.2.3 for DC busses in MODES 1-4 requires one OPERABLE battery charger, and if it is not, connect the backup battery charger.

Reference Title

AC Electrical Distribution Simplified One-Line

Technical Specifications

Learning Objectives

13KVACE016	Given a 13KV Electrical System (excludes Unit No. 3) failure, predict the effect of the 13KV Electrical System (excludes Unit No. 3) failure on the following: (License Operator and STA only) Off-site power sources

Material Required for Examination

Question Source:	New	Question Modification Method:	

Question Source Comments:

Given the following conditions:

- U2 is performing actions to isolate a 125VDC ground on 2A 125VDC bus.
- An Equipment Operator depresses the local pushbutton for 2A 125VDC bus to read bus resistance-to-ground.

How will this action be noticed in the Main Control Room?

a. 2A 125VDC bus ground indication will indicate infinity.

b. 2A 125VDC bus ground indication will indicate zero ohms.

c. OHA B-2, 2A 125VDC CNTRL BUS VOLT LO will annunciate.

d. Aux Annuciator Alarm 0179 2A 125VDC GROUND FAULT DETECTION will alarm.

Answer	a	Exam Level	R	Cognitive Level	Memory	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Plant Systems	RO Group	1	SRO Group	1				063000A301
063	D.C. Electrical Distribution							Record Number	57
A3.	Ability to monitor automatic operations of the D.C. Electrical Distribution including:								
A3.01	Meters, annunciators, dials, recorders, and indicating lights							2.7	3.1

Explanation of Answer

A is correct because the test PB depressed locally disconnects the output going to the Main Control Room when depressed. With what appears to be an "open" circuit, the CR gauge will read infinite resistance. B is incorrect because of A above. C is incorrect because operation of the ground test PB doesn't interrupt the voltage indication for the bus and won't actuate the low voltage alarm. D is incorrect because there is no such AAT alarm. However, if the student does not know the correct answer, it is a plausible distracter.

Reference Title

2A 125 Volt Storage Battery Instrument and Alarm Circuit

OHA B Alarm Response

Learning Objectives

DCELECE008	Identify and describe the Control Room controls, indications, and alarms associated with the DC Electrical System, including: The Control Room location of DC Electrical System control bezels and indications. (Licensed Operator & STA only) The function of each DC Electrical System Control Room control and indication. (Licensed Operator & STA only) The effect each DC Electrical System control has upon DC Electrical System components and operation. (Licensed Operator & STA only) The plant conditions or permissives required for DC Electrical System Control Room controls to perform their intended function. (Licensed Operator & STA only)
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:			

Given the following conditions:

- Unit 1 is in MODE 4.
- An electrical fault causes the infeed breaker to 1A EDG 230V MCC to open.

Per Tech Specs, which of the following choices describes the effect this has on 1A EDG?

a. The EDG remains OPERABLE until Lube Oil temperature drops below 100 deg.

b. The EDG remains OPERABLE until Jacket Water temperature drops below 135 deg.

c. The EDG is INOPERABLE due to the loss of its Prelube pump and Jacket Water Heater.

d. The EDG is INOPERABLE until 125VDC control power has been transferred to its alternate source.

Answer	c	**Exam Level**	R	**Cognitive Level**	Memory	**Facility:**	Salem 1 & 2	**Exam Date:**	12/11/2006
Tier:	Plant Systems			**RO Group**	1	**SRO Group**	1		064000G222
064	Emergency Diesel Generators							**Record Number**	58
2.2	Equipment Control								
2.2.22	Knowledge of limiting conditions for operations and safety limits.							3.4	4.1

Explanation of Answer

The EDG is designed to auto start under conditions where AC power has been lost. Therefore, the loss of the EDG auxiliary MCC will not in itself make the EDG INOPERABLE. Distracter d is incorrect because 125VDC power remains available through its normal source. c is correct because the EDG will be declared INOP when the prelube pump and jacket water heater are both inoperable (P&L 3.5). Distracter b is incorrect because of c above. Distracter a is incorrect because of C above.

Reference Title

2A Diesel Generator Operation

Learning Objectives

EDG000E010	LOR Given a situation dealing with Emergency Diesel Generator 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 Emergency Diesel Generator 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: Direct From Source

Question Source Comments: VISION Q82890

Given the following conditions:

- Unit 1 is operating in MODE 1.
- While performing rounds, an NEO isolated the Air Compressor supply valves to 11A and 11B Diesel Generator Starting Air Receivers in order to perform a blowdown of the tanks for moisture IAW S1.OP-DL.ZZ-0006, PRIMARY PLANT LOGS.
- BOTH Air Receivers were left isolated.

Which of the following describes the effect this will have on 1A EDG OPERABILITY IAW Technical Specifications?

The EDG...

a. remains OPERABLE, since either air start receiver is designed to provide 3 cold starts.

b. remains OPERABLE, since the Turbo-Boost air receivers can be cross-connected to supply starting air if required.

c. became INOPERABLE when the Air Compressor supply valve to the second Air Receiver was closed since NO starting air is available to the EDG.

d. will become INOPERABLE when the "AIR RECEIVER #1 PRESSURE LOW", and "AIR RECEIVER #2 PRESSURE LOW" alarms are actuated at the EDG control panel, since this indicates <190 psig start air available.

Answer	a	Exam Level	R	Cognitive Level	Memory	Facility:	Salem 1 & 2	Exam Date:	
Tier:	Plant Systems			RO Group	1	SRO Group	1		064000K607
064	Emergency Diesel Generators							Record Number	59

K6. Knowledge of the of the effect of a loss or malfunction on the following will have on the Emergency Diesel Generators:

K6.07 Air receivers — 2.7 | 2.9

Explanation of Answer: A is correct. As shown on dwg 211315, the isolation of air from the compressors to the tanks will not affect the air supply path to the starting air motors. Each Starting air tank IS designed for three cold starts when at 160 psig. Distracter b is incorrect because the EDG remains OPERABLE, but the turbo boost air receivers can NOT be cross connected with the starting air receivers. Distracter c is incorrect because the air "in" to the tank is a separate line from the air "out" of the tank to the EDG. Distracter d is incorrect because the lo pressure alarm is at 182 psig, not 160 psig.

Reference Title

Component Design Basis

Primary Plant Logs

No. 1 Unit-1A Diesel Generator Start and Turbo Boost air System

Learning Objectives

EDG000E010 — LOR Given a situation dealing with Emergency Diesel Generator 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 Emergency Diesel Generator 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:

- 1 WMHUT is in recirc, a sample has been drawn and is in the process of being analyzed
- The RWO mistakenly places 1 WMHUT in service
- One hour later, the RWO realizes his error, and returns 1 WMHUT to recirc

What effect, if any, will this have on the release preparations for 1 WMHUT IAW S1.OP-SO.WL-0003, RELEASE OF RADIOACTIVE LIQUID WASTE FROM #1 WASTE MONITOR HOLDUP TANK?

a. The current sample is invalidated. A new sample must be drawn with no minimum required recirculation.

b. The current sample is invalidated. The tank will require further recirculation and resampling prior to release.

c. The release preparations may continue as long as volume added to tank does not exceed 1% of total tank volume ONLY.

d. The release preparations may continue as long as volume added to tank does not exceed 1% of total tank volume AND double verification of sample analysis is performed.

Answer	b	Exam Level	R	Cognitive Level	Memory	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Plant Systems	RO Group	2	SRO Group	2				068000A202
068	Liquid Radwaste System							Record Number	60

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

A2.02 Lack of tank recirculation prior to release 2.7* 2.8*

Explanation of Answer

S1.OP-SO.WL-0003, Section 5.1, Release preparations, "NOTE Any additions made to 1WMHUT after it is isolated and placed on recirculation for sampling will invalidate the sample analysis and require further recirculation time and resampling". The correct answer b contains these requirements. Distracter a is incorrect because the recirculation time must be reset after additions to the tank have been made. Distracters c and d are incorrect because there is no provision for waiving the recirc and resampling due to low volume of added liquid following initial recirculation and sampling.

Reference Title

RELEASE OF RADIOACTIVE LIQUID WASTE FROM #1 WASTE MONITOR HOLDUP TANK?

Learning Objectives

WASLIQE012	LOR NCT Discuss the procedural requirements associated with the Radioactive Liquid Waste System, including an explanation of major precaution and limitations in the Radioactive Liquid Waste System procedures

Material Required for Examination			
Question Source:	Facility Exam Bank	Question Modification Method:	Direct From Source
Question Source Comments:	VISION Q69947		

Operators are performing check source testing of Radiation Monitors IAW S1.OP-ST.RM-0001, RADIATION MONITORS - CHECK SOURCES. While performing the check source test for 1R9 - Fuel Storage Area Radiation Monitor, the counts seen by the monitor rise above the Hi Radiation Alarm setpoint.

Which of the following describes the consequences of this action?

a. The Fuel Handling Building (FHB) supply and exhaust fans will receive an auto start signal.

b. The FHB Ventilation system will automatically realign the exhaust to flow through #12 Filter Unit.

c. The FHB Hi Radiation Evacuation Horn will sound, but no ventilation system realignment will occur because it is MANUALLY blocked prior to source check testing.

d. The FHB Hi Radiation Evacuation Horn will sound, but no ventilation system realignment will occur because it is AUTOMATICALLY blocked during source check testing.

Answer	b	Exam Level	R	Cognitive Level	Memory	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Plant Systems		RO Group	2	SRO Group	2			072000A401
072	Area Radiation Monitoring System							Record Number	61
A4.	Ability to manually operate and/or monitor in the control room:								
A4.01	Alarm and interlock setpoint checks and adjustments							3.0*	3.3

Explanation of Answer

A is incorrect because the fans do NOT receive an auto start signal. B is correct, and c and d are incorrect because the check source test does not automatically block an actuation nor does the procedure have the RP1 block switch placed in BLOCK for the test.

Reference Title

RADIATION MONITORS - CHECK SOURCES

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.

RMS000E003

NCT Describe the function of the following components and how their normal and abnormal operation affects the Radiation Monitoring System:
R1A, Control Room Area Monitor
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
R15, Condenser Air Ejector Process Monitor
R17A and B, Component Cooling Liquid Monitor
R18, Liquid Waste Disposal
R19A, B, C, & D, Steam Generator Blowdown Liquid Monitors
1R31A, Letdown Gross Activity Process Monitor
2R31, Letdown Heat Exchanger/Failed Fuel Process Monitor
R32A, Fuel Handling Crane Area Radiation Monitor
R36, Evaporator and Feed Preheaters Condensate Monitor
2R37, Non-Radwaste Basin Process Monitor
R40, Condensate Filter Process Filter Monitor
R41A, B, C, & D, Plant Vent Radiation Monitor
R44A & B, Containment High Range Area Monitor

	R45A, B, C, & D, Plant Vent High Range Radiation Monitor R46A-E, Main Steam Line Process Monitor R47, Electrical Penetration Area Monitor 2R52, Liquid PASS Room Area Radiation Monitor R53, N16 Main Steam Line Radiation Monitor
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:	New	**Question Modification Method:**	
Question Source Comments:			

Given the following conditions:

- Unit 2 is operating at 100% power.
- Unit 1 is in MODE 5.
- All Unit 1 circulators are secured.
- Unit 2 Waste Liquid release is in progress from 21 CVCS MT to UNIT 2 CW system via the cross connect to Unit 1.

2R18 2 RAD MONIT LIQUID WASTE DISPOSAL PRCS RAD MON is OPERABLE.

Which choice identifies the AUTOMATIC action(s) that will take place if 2R18 fails HIGH?

a. 1WL115 and 2WL115 WASTE DISCHARGE HDR X-CONN VALVES will shut.

b. 2WL51 LIQUID RELEASE STOP VALVE ONLY will shut.

c. 2WL51will shut but release will continue through 1WL51.

d. 2WL51 AND 1WL51 will shut.

Answer	b	**Exam Level**	R	**Cognitive Level**	Memory	**Facility:**	Salem 1 & 2	**Exam Date:**	12/11/2006

Tier:	Plant Systems	**RO Group**	1	**SRO Group**	1	073000K301
073	Process Radiation Monitoring System			**Record Number**	62	

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

When performing a Radioactive Liquid release through the cross connect to the opposite unit, all the valves listed above will be open. A is correct because Unit 2 R18 automatic action affects only Unit 2 WL51. Distracter b is incorrect because 2R18 will not auto close the opposite unit WL51. Distracter c is incorrect because the WL115s are manually operated valves. Distracter d is incorrect because the release path will be isolated when the 2WL51 shuts.

Reference Title

No. 1 & 2 Units Waste Disposal System No. 1WL51 & 2WL51 Liquid Waste Disch Valves

No. 2 Unit Waste Disposal Liquid

Release of Radioactive Liquid Waste from 22 CVCS Monitor Tank

Learning Objectives

WASLIQE005	State the power supply to the following Radioactive Liquid Waste System components: a) Not applicable to this lesson

Material Required for Examination			
Question Source:	Facility Exam Bank	**Question Modification Method:**	Direct From Source
Question Source Comments:	VISION Q50403		

Given the following conditions:

- Unit 2 is operating at 100% power.
- Unit 1 is in MODE 5.
- All Unit 1 circulators are secured.
- Unit 2 Waste Liquid release is in progress from 21 CVCS MT to UNIT 2 CW system via the cross connect to Unit 1.

2R18 2 RAD MONIT LIQUID WASTE DISPOSAL PRCS RAD MON is OPERABLE.

Which choice identifies the AUTOMATIC action(s) that will take place if 2R18 fails HIGH?

a. 1WL115 and 2WL115 WASTE DISCHARGE HDR X-CONN VALVES will shut.

b. 2WL51 LIQUID RELEASE STOP VALVE ONLY will shut.

c. 2WL51will shut but release will continue through 1WL51.

d. 2WL51 AND 1WL51 will shut.

Answer	b	Exam Level	R	Cognitive Level	Memory	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Plant Systems	RO Group	1	SRO Group	1				073000K301
073	Process Radiation Monitoring System							Record Number	63

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

When performing a Radioactive Liquid release through the cross connect to the opposite unit, all the valves listed above will be open. A is correct because Unit 2 R18 automatic action affects only Unit 2 WL51. Distracter b is incorrect because 2R18 will not auto close the opposite unit WL51. Distracter c is incorrect because the WL115s are manually operated valves. Distracter d is incorrect because the release path will be isolated when the 2WL51 shuts.

Reference Title

No. 1 & 2 Units Waste Disposal System No. 1WL51 & 2WL51 Liquid Waste Disch Valves

No. 2 Unit Waste Disposal Liquid

Release of Radioactive Liquid Waste from 22 CVCS Monitor Tank

Learning Objectives

WASLIQE005	NCT Outline the interlocks associated with the following Radioactive Liquid Waste System components: Auxiliary Sump Tank and Pumps Chemical Drain Tank, Laundry and Hot Shower Tanks, and their respective pumps Reactor Coolant Drain Tank Pumps, RCDT Discharge Isolation Valves WL12 and WL13, and PRT Drain Valve PR14 Containment Sump Pumps, Containment Sump level, and Containment Isolation Valves WL16 and WL17 Radioactive Liquid Waste Discharge Isolation Valve WL51 and Liquid Waste Discharge Radiation Monitor RA4335/R18 Containment Isolation Phase A and Radioactive Liquid Waste valves WL12 & WL13, WL16 & WL17, WL96 & WL 97, WL98 & WL99, and WL108

Material Required for Examination			
Question Source:	Facility Exam Bank	Question Modification Method:	Direct From Source
Question Source Comments:	VISION Q50403		

Given the following conditions:

- Unit 1 is operating at 100% power.
- A motor short occurs on 11A Circulator, causing an overload condition on 1CW bus section 13.
- 1CW bus section 13 becomes deenergized when its infeed breaker 13CW1AD - #13 STA PWR XFMR INFD BKR TRIPS OPEN as expected.
- OHA K-2, 4KV CW BUSS DIFF OVRLD annunciates, and CRT point 433, CW Swgr Bus Section 13 Overload Trip is received on 1CC1.

Which of the following describes the effect this will have on the plant, and what actions are required IAW S1.OP-AB.CW-0001, CIRCULATING WATER SYSTEM MALFUNCTION?

a. Condenser backpressure will start to rise on BOTH east and west sides. Open the 11/12/13MC62, Turbine Hood Spray Bypass Valves.

b. The CW bus cross-tie breaker 1CW8AD, will close after a 15 second time delay. Re-start the affected Circulators after the CW26/126 valves have fully stroked to the closed position.

c. Control rods will begin stepping in as RCS temperature rises due to the loss of Main Turbine load. Steam Generator Blowdown must be isolated due to the loss of 12A and 12B Circulators.

d. Main Turbine load will immediately drop by at least 120MWe. Initiate a power reduction to less than or equal to 83% to prevent flashing in the Condensate System as hotwell temperature rises.

Answer	a	**Exam Level**	R	**Cognitive Level**	Comprehension	**Facility:**	Salem 1 & 2	**Exam Date:**	12/11/2006
Tier:	Plant Systems	**RO Group**	2	**SRO Group**	2				075000A202
075	Circulating Water System							**Record Number**	64

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

A2.02 Loss of circulating water pumps — 2.5 | 2.7

Explanation of Answer

Distracter b is incorrect because on a bus overload condition, the CW bus cross-tie receives a lockout signal along with the bus infeed breaker. Distracter c is incorrect because SGBD isolation is not required unless both 12A and 12B condensers are affected. The loss of 13 CW bus will affect the 11A, 12A, and 13A circulators. Control rods will begin stepping is as RCS temp rises. Distracter d is incorrect because Main Turbine load will not drop immediately, it will take several minutes before load starts to show the effect from the backpressure change in the condensers, and load will lower slowly. The action for distracter c is correct. A is correct because as temperature rises in the affected condensers, with no cooling water supplied to those condensers, there will be less condensing, and efficiency will drop. Opening of the hood spray bypass valves is correct.

Reference Title

Circulating Water System Malfunction

1CW 4KV Bus Operation

Overhead Annunciators Window K

Learning Objectives

ABCW01E005
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: New **Question Modification Method:**

Question Source Comments:

Given the following conditions:

- Unit 2 is operating at 100% power.
- A large earthquake 5 miles from the site causes a loss of off-site power.
- The reactor trips, and a MANUAL Safety Injection is initiated.
- 2B EDG output breaker does NOT close.

With NO other operator action, which choice contains the system lineup for the Service Water System 5 minutes after the SI?

a. 2SW26 TURB AREA SW MOV STOP VLV SHUT, 21SW122 CC HX SW INLET VALVE SHUT, 24SW223 CV FANS SW OUTLET V SHUT.

b. 2SW26 SHUT, 22SW122 OPEN, 25SW223 OPEN.

c. 2SW26 OPEN, 22SW122 SHUT, 23SW223 OPEN.

d. 2SW26 OPEN, 21SW122 OPEN, 22SW223 SHUT.

Answer	c	Exam Level	R	Cognitive Level	Comprehension	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Plant Systems			RO Group	1	SRO Group	1		076000A302
076	Service Water System							Record Number	65
A3.	Ability to monitor automatic operations of the Service Water System including:								
A3.02	Emergency heat loads							3.7	3.7

Explanation of Answer

2SW26 will close on a signal from the 2B SEC. In the case presented in the stem, 2B EDG does not energized 2B 4KV vital bus following a loss of off-site power,(which supplies power to the B SW 230VAC MCC) and 2SW26 will remain in its normal position of open. The 21/22SW122s receive a SHUT signal from the "A" and "B" SECs respectively. The SEC, while not having vital loads to sequence, still performs its ancillary control functions of closing the SW122.

Reference Title

No. 2 Unit - Auxiliary Building #22 Component Cooling Heat Exchanger

No. 2 Unit - Auxiliary Building # 21 and 22 CCHX Inlet Control

Learning Objectives

SW0NUCE006	NCT Outline the interlocks associated with the following Service Water - Nuclear Header System components: Containment Fan Coil Unit High and low Speed Breakers CFCU Service Water Inlet Pressure Control Valve CFCU Motor Cooling Flow Control Valve SW Accumulator Building Ventilation fans Chiller Service Water Inlet Valve SI Pump Lube Oil Coolers Service Water Inlet Valve
SW0NUCE007	NCT Identify and describe the local controls, indications, and alarms associated with the Service Water - Nuclear Header System, including: The location of Service Water û Nuclear Header System local controls and indications. (Licensed Operator & Non-licensed Operator only) The function of Service Water û Nuclear Header System local controls and indications. (Licensed Operator & Non-licensed Operator only) The plant conditions or permissives required for Service Water û Nuclear Header System local controls to perform their intended function. (Licensed Operator & Non-licensed Operator only) The setpoints associated with the Service Water û Nuclear Header System local alarms. (Licensed Operator & STA only)

Material Required for Examination			
Question Source:	Facility Exam Bank	Question Modification Method:	Editorially Modified
Question Source Comments:	Modified a distracter to even out all the open/shuts.		

Given the following conditions:

- Unit 1 is operating at 100% power.
- 12 charging pump is in service.
- Normal letdown must be secured to troubleshoot a control problem with 1CV18, LETDOWN PRESSURE CONTROL VALVE.
- Excess letdown has been placed in service.

Prior to securing Normal Letdown, which of the following actions MUST be performed IAW S1.OP-SO.CVC-0001, CHARGING, LETDOWN, AND SEAL INJECTION, and why?

a. Fully open 1CV71, CHG HDR PCV, to ensure flashing in the Excess Letdown line does not occur.

b. Fully close the 1CV55, CENT CHG PMP FLOW CONT VALVE, to ensure RCP seal injection remains above 6 gpm per pump.

c. Place 1CA2015, CONTROL AIR SUPPLY TO CV55 BYPASS VALVE, in BYPASS, to allow the 1CV55 to control flow less than the normal minimum flow position.

d. Adjust the position of the speed control linkage for 13 charging pump to a lower pressure position, to prevent exceeding the Tech Spec limit of 40 gpm total flow to the RCP seals.

Answer	c	Exam Level	R	Cognitive Level	Memory	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Plant Systems		RO Group	1	SRO Group	1			078000K401
078	Instrument Air System							Record Number	66
K4.	Knowledge of Instrument Air System design feature(s) and or interlock(s) which provide for the following:								
K4.01	Manual/automatic transfers of control							2.7	2.9

Explanation of Answer

Distracter a is incorrect because flashing is only a concern on the LETDOWN line when charging flow is reduced below ~ 60 gpm with normal letdown flow established due to the cooling of letdown flow in the Regenerative Heat Exchanger. The Excess letdown HX is only cooled by CCW, there is no Regen function. Distracter b is incorrect because the CV55 will lower flow to the RCP seals, and the minimum flow stop will be in excess of the required CVCS flow to minimize PZR level rise AND is not required by the procedure. C is correct because at step 5.3.2 of the procedure, between the steps for placing Excess Letdown in service and securing Normal Letdown, is the step(s) to change the min flow stop for either the centrifugal charging pump FCV CV55 or the speed linkage for the PDP, whichever is in service. Distracter d is incorrect because the speed linkage is adjust to allow lower charging flow to minimize the gain in PZR level while maintaining >6 gpm per pump to the RCP seals. Also the pump is not in service, and the linkage adjustment is only required when it is.

Reference Title

Charging, Letdown, and Seal Injection

Excess Letdown Flow

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

CVCS00E013	LOR NCT Discuss the procedural requirements associated with the Chemical and Volume Control System, including an explanation of major precaution and limitations in the Chemical and Volume Control System procedures.

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

Which of the following events would require the transfer of spent fuel elements to the Spent Fuel Pool to be suspended during MODE 6 refueling operations IAW S2.OP-SO.SF-0009, REFUELING OPERATIONS?

a. Fuel Handling Area Rad monitor 2R5 fails low.

b. Only one FHB Supply Fan and 2 FHB Exhaust Fans are running.

c. An SRO over-seeing Spent Fuel Pool manipulations leaves the area under supervision of a qualified Reactor Engineer.

d. 21 Spent Fuel Pool Cooling pump is discovered to have no oil in its pump oil bubbler with 22 Spent Fuel Pool Cooling Pump in service.

Answer	d	**Exam Level**	R	**Cognitive Level**	Application	**Facility:**	Salem 1 & 2	**Exam Date:**	12/11/2006
Tier:	Plant Systems	**RO Group**	1	**SRO Group**	1				103000G228
103	Containment System							**Record Number**	67
2.2	Equipment Control								
2.2.28	Knowledge of new and spent fuel movement procedures.							2.6	3.5

Explanation of Answer

Distracter a is incorrect because only one of the two FHB area rad monitors are required to be OPERABLE IAW TSAS 3.3.1.1, Table 3.3-6. Distracter b is the complement of fans required to be running to have an OPERABLE FHB ventilation system. Distracter c is incorrect because the requirement for supervision of loads in the Spent Fuel Pool is a SRO OR a Qualified RE. D is correct because in S2.OP-SO.SF-0009, REFUELING OPERATIONS, P&L 3.12 specifically requires suspension of irradiated fuel into the SFP when either 21 or 22 SFP pump becomes INOPERABLE. The loss of all oil in the pump bubbler renders the pump INOPERABLE.

Reference Title
Refueling Operations
Technical Specifications

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)
REFUELE010	Given a situation dealing with Refueling System operability, examine the situation and apply the appropriate Technical Specification action. (License Operator and STA only) State the Technical Specification associated with the component, parameters and operation of the Refueling 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:

- You are a Licensed Reactor Operator, assigned to an off-shift administrative position.
- During the first calendar quarter, you have stood the following duties, all 12 hours plus turnover:
  - 1/6/06 U1 RO
  - 1/30/06 U2 PO
  - 2/2/06 U2 RO
  - 2/27/06 WCC RO
  - 3/10/06 U2 RO

Today is 4/1/06

With regards to watch standing hours, which of the following describes the status of your license in accordance with OP-AA-105-102, NRC ACTIVE LICENSE MAINTENANCE?

a. Active. You may stand watch with no restrictions.

b. Active. You must regain qualification as RO by standing one additional 12 hour shift in the RO or PO position.

c. Inactive. You must reactivate your license by standing at least 40 hours under instruction as either the RO or PO.

d. Inactive. You must reactivate your license by standing at least five 12 hour shifts under instruction in the RO/PO/WCC RO position.

Answer	c	Exam Level	R	Cognitive Level	Application	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Generic Knowledge and Abilities		RO Group	1	SRO Group	1			194001G101
GENERIC								Record Number	68

2.1	Conduct of Operations		
2.1.1	Knowledge of conduct of operations requirements.	3.7	3.8

Explanation of Answer

A is incorrect because the WCC NCO is not a Licensed Position. OP-AA-105-102, NRC Active License Maintenance, specifically states that the quarterly watch requirements are to be stood by "performing the duties of the Unit RO and/or the Unit Assist RO" (Plant Operator at Salem). As such, the individual has only stood 48 hours of watch, plus ~1 hour of turnover, and has not met the requirement of 5 12-hour shifts per calendar quarter. Distracter b is incorrect because the 1st calendar quarter is completed, and NO watchstanding can be performed until the license is re-activated. C is correct because 40 hours must be stood at the RO or PO position. (OP-AA-105-102, Rev. 7, Attachment 2, Reactivation of License Log). Distracter d is incorrect because the WCC RO position does not count towards Licensed Duties hours.

Reference Title

NRC Active License Maintenance

Learning Objectives

CONDOPE005 Describe requirements for the following Control Room Activities in accordance with SH.OP-AS.ZZ-0001(Z), Operations Standards Reactivity Management
Industrial Safety Practices
Radiation Worker Practices
Conservative Decision Making
Control Room ôAt the Controls Areaö
Communication
Shift Relief and Turnover
Alarm Response
Operator Appearance
Self Assessment/Corrective Action
Plant/Control Board Awareness and Maintenance of Critical Parameters

Housekeeping/Cleanliness/FME
Operator Rounds
Briefs
Human Error Reduction Techniques
Log Keeping
Training
Supervisor Involvement
Accessing Equipment
Attachment 1, Shift Briefing Format
Attachment 2, Pre-Job Briefing Guidelines
Attachment 3, Pre-Job Briefing Checklist
Attachment 4, Pre-Job Briefing
Attachment 5, Human Performance, Top Ten Human Error Traps
Attachment 6, Control Room Interaction Model

Material Required for Examination	
Question Source:	Other Facility
Question Modification Method:	Concept Used
Question Source Comments:	Beaver Valley 1/19/06 NRC Exam

Which of the following sets of conditions would require the most rapid action in order to comply with the applicable Unit 2 Technical Specifications Action Statement?

a. The Unit is in MODE 1 at 40% power. QPTR has just been reported as 1.08.

b. The Unit is in MODE 4. Operators have just removed control power from 22 SI pump. 21 SI pumps is C/T.

c. The Unit is in MODE 1 at 6% power. 23 SW pump is C/T. A SW leak has occurred in SW Bay #4, and operators have just removed control power from 24-26 SW pumps IAW S2.OP-AB.SW-0003, SERVICE WATER BAY LEAK.

d. The Unit is in MODE 5. Spent fuel assemblies have just been shuffled in the Spent Fuel Pool. A Fuel Storage Pool Verification has NOT been performed since the fuel assembly movement stopped. Chemistry reports Spent Fuel Pool Boron concentration is 1990 ppm.

Answer	c	Exam Level	R	Cognitive Level	Memory	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Generic Knowledge and Abilities	RO Group	1	SRO Group	1				194001G111
GENERIC								Record Number	69
2.1	Conduct of Operations								
2.1.11	Knowledge of less than one hour technical specification action statements for systems.							3.0	3.8

Explanation of Answer	Distracter a is incorrect because while the QPTR is high, it is only applicable greater than 50% power in MODE 1. Distracter b is incorrect because Tech Spec 3.5.3 requires 1 ECCS subsystem OPERABLE in MODE 4 consisting of ONE charging pump and flow path from RWST, and capable of taking suction from RHR discharge piping, and discharging into each RCS cold leg, AND ONE RHR pump and flowpath from RWST & able to transfer to Cont Sump and discharging into each RCS cold leg, and 2 RCS hot legs upon manual initiation. SI pumps are NOT required in MODE 4. (TSAS 3.5.3 has a one hour action to restore). c is correct because in MODEs 1-4, 2 independent SW loops are required to be OPERABLE IAW TSAS 3.7.4. An OPERABLE SW loop consists of at least 2 SW pumps powered from separate vital busses. 21 and 22 SW pumps are powered from A vital bus. This puts the unit in TS 3.0.3, since there is only an action for 1 SW loop INOPERABLE in the spec. Distracter d is incorrect because the boron requirement for Spent Fuel Pool OPERABILITY in MODE 5 (TSAS 3.7.11) is greater than or equal to 800 ppm when the Fuel Pool Storage Verification has NOT been performed since the last fuel movement in the SFP. The requirement stated in the COLR is ONLY applicable in MODE 6. (TSAS 3.7.11 has an IMMEDIATE action when not in compliance)

Reference Title
Salem Tech Specs
Service Water System Operation

Learning Objectives	
FLUNCYE002	State those items in the Licensed Operator Fluency List A. Permissives and Control Grade Interlocks B. Reactor Trips C. Safety Injection D. Containment Isolation E. AFW Pump Auto Starts F. SEC Mode Ops G. RMS Automatic Actuations H. Reactivity Coefficients I. Red and Purple Paths J. TRIP-1 CASs K. Steamline Isolations L. Feedwater Isolations M. Feedwater Interlock N. Key Relief Valves O. Tank Thumbrules P. 1 hour or less Technical Specifications

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

Which of the following describes a difference between the Unit 1 and Unit 2 control consoles?

a. Unit 1 has the Demineralized Water system indications, controls, and alarms. Unit 2 only has the Demin System alarms.

b. Unit 1 RWST has four channels of level indication for defense in depth considerations. Unit 2 was designed with two RWST level channels.

c. If armed, the Unit 2 RHR HX CCW Outlet valves 21/22CC16 will automatically open at 15.2' of level in the RWST following a SI. Unit 1 11/12CC16 must be manually opened.

d. If armed, the Unit 2 SI CROSS-OVER VALVES 21SJ113 and 22SJ113 will automatically open when a Safety Injection signal is received. Unit 1 11/12SJ113 valves must be manually opened.

Answer	c	**Exam Level**	R	**Cognitive Level**	Memory	**Facility:**	Salem 1 & 2	**Exam Date:**	12/11/2006
Tier:	Generic Knowledge and Abilities		**RO Group**	1	**SRO Group**	1			194001G204
GENERIC								**Record Number**	70

2.2	Equipment Control		
2.2.4	(multi-unit) Ability to explain the variations in control board layouts, systems, instrumentation and procedural actions between units at a facility.	2.8	3.0*

Explanation of Answer

A is incorrect because Unit has no demineralized water alarms, indications, or controls. B is incorrect because Unit 1 has 2 channels, and Unit 2 has four channels for the auto swapover feature. C is correct because the 2CC16 will auto open on S signal and 15.2' in RWST on 2/4 channels. D is incorrect because BOTH a SI signal and 15.2' signal must be received.

Reference Title

Swap to Cold Leg Recirculation

Learning Objectives

LCA3U1E004	Determine the indications that are monitored to ensure proper system/component operation for each step in 2-EOP-LOCA-3
LCA3U2E004	Determine the indications that are monitored to ensure proper system/component operation for each step in 2-EOP-LOCA-3

Material Required for Examination

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

Question Source Comments:

Given the following condition:

- Unit 2 Reactor Trip Breakers were opened at 2000 on November 10, 2005.

Which of the following identifies the EARLIEST time irradiated fuel may be moved in the Rx?

a. 0001 on November 15th.

b. 2001 on November 15th.

c. 0001 on November 17th.

d. 2001 on November 17th.

Answer	a	Exam Level	R	Cognitive Level	Application	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Generic Knowledge and Abilities	RO Group	1	SRO Group	1				194001G226
GENERIC								Record Number	71
2.2	Equipment Control								
2.2.26	Knowledge of refueling administrative requirements.							2.5	3.7

Explanation of Answer

TSAS 3.9.3.a requires at least 100 hours of subcriticality (Oct 15th-May 15th) during movement of irradiated fuel in the reactor pressure vessel. A is correct because it is 100 hours from when the reactor was made subcritical by opening the Rx trip breakers. Distracter c is incorrect because it is 168 hours from RTB opening, which is the minimum time required during the months of May 15th-Oct 15th. Distracters b and d are combinations of the other 2 choices.

Reference Title

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:	New	Question Modification Method:	
Question Source Comments:			

A Salem Radiation Worker has received a current year-to-date TEDE dose at Salem of 2,970 mrem. Additionally, he received 1,200 mrem at another plant in the current year.

The worker is 45 years old with a total lifetime TEDE dose of 17 REM, and has received any extension necessary for him to reach his CURRENT Salem dose.

Which of the following choices describes a situation that is allowed for this worker IAW NC.NA-AP.ZZ-0024 RADIATION PROTECTION PROGRAM, in regards to his future TEDE dose received in the same year?

a. The Radiation Protection Manager authorizes an administrative dose extension to 4,000 mrem.

b. The Radiation Protection Supervisor authorizes an administrative does extension above 3,000 mrem.

c. The Plant Manager authorizes an incremental increase above 4,000 mrem with no Emergency in progress.

d. The Senior Vice President-Site Operations authorizes a Planned Special Exposure which will result in the worker receiving 1,000 mrem

Answer	d	Exam Level	R	Cognitive Level	Application	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Generic Knowledge and Abilities	RO Group	1	SRO Group	1		194001G304		
GENERIC							Record Number	72	

2.3	Radiation Control		
2.3.4	Knowledge of radiation exposure limits and contamination control, including permissible levels in excess of those authorized.	2.5	3.1

Explanation of Answer

The Salem administrative dose limit for TEDE is 2,000 mrem per year. A limit of 3,000 mrem per year may be authorized by the Radiation Protection supervisor. A limit of 4,000 mrem per year may be authorized by the Radiation Protection Manager. An incremental limit of up to 4,750 mrem may be authorized by the Plant Manager under Emergency conditions. Salem admin limits apply ONLY to dose received at Salem. However, each nuclear plant is required by 10.CFR.20.1201(f) to reduce the allowable dose by that dose received by the worker anywhere else. In the instance described above, the worker has a total YTD exposure of 4,170 mrem. MOST authorizations provided which would allow an the worker to exceed 5 REM/ yr are illegal. HOWEVER, 10.CFR.20.1206, Planned special exposures, directs that this dose shall be maintained separate from the yearly occupational dose, as long as the special exposure dose plus the occupational dose does not EXCEED the occupational dose numbers found in 1201(a). This means that the 5 REM/yr TEDE dose cannot be exceeded by more than 5 REM TEDE. D is correct because even though the Planned special exposure dose will NOT be added to his occupational dose, and as a result, his occupational dose will not rise above 5 REM for the current year. Distracters a, b, and c are both wrong because it would raise the workers dose limit above 5 REM for the year, which is illegal.

Reference Title

Radiation Protection Program

Code of Federal Regulations

Learning Objectives

RADCONE002

List the following external radiation exposure limits, in accordance with Station Procedures, 10CFR20, and Reg. Guide 8.13:

A. 10CFR20 dose limits for external, internal, and total whole body, skin, extremities, and eyes, as well as extension limits and requirements
B. Administrative dose control levels for Category 1 and 2 Workers, as well as extension limits and requirements
C. Reg. Guide 8.13 limits and administrative dose control levels for Declared Pregnant Women
D. 10CFR20 and Administrative limits for members of the general public and minors
E. Category 1 Radiation Worker
F. Category 2 Radiation Worker

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

Given the following conditions:

- Unit 2 is operating at 75% power.
- Unit 1 is operating at 100% power.
- A MANUAL Rx trip and SI are initiated on UNIT 2 due to a LOCA.
- At step 15 of 2-EOP-TRIP-1, operators discover the Control Room Ventilation system is in the NORMAL Mode.

Which of the following identifies the actions required IAW EOP-TRIP-1, if any, and why?

a. No action is required, NORMAL is the correct post Rx trip alignment.

b. No action is required, since the R1B channels will automatically isolate the Control Room Envelope if outside air radiation levels rise.

c. Depress EITHER units Accident Pressurized PB. This will isolate ALL outside air supplied to the Control Room, and habitability requirements will be met.

d. Depress the Accident Pressurized PB on 2RP2 ONLY. This will allow only a small amount of outside air to mix with recirculated Control Room air, preventing a possible Control Room evacuation.

Answer	d	Exam Level	R	Cognitive Level	Application	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Generic Knowledge and Abilities	RO Group	1	SRO Group	1				194001G310
GENERIC								Record Number	73

2.3	Radiation Control		
2.3.10	Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure.	2.9	3.3

Explanation of Answer

A is incorrect because a Safety Injection signal is an automatic signal to realign the CAV system to Accident Pressurized. B is incorrect, because while the R1B channels (air intake radiation monitors) will automatically initiate Accident Pressurized on high radiation, but the TRIP-1 procedure explicitly states to Initiate Accident Pressurized. C is incorrect because a small amount (~1100 scfm out of 7700 scfm) will be drawn into the CAV system from the unaffected unit air intake. D is correct because the EOP step requires it, and the FSAR requires control room habitability to be maintained following the most credible accident.

Reference Title

Rx Trip or Safety Injection Bases Document

Tech Specs Bases

S2.OP-SO.CAV-0001

Learning Objectives

CAVENTE002	Describe the design bases of the Control Area Ventilation System. (Licensed Operator & STA only)

Material Required for Examination:

Question Source: New

Question Modification Method:

Question Source Comments:

Given the following conditions:

- Unit 2 is operating at 100% power.
- A release of 21 WGDT is in progress, 2WG41 is OPEN.
- Containment pressure is 0.21 psid.

Which choice states whether or not a Containment Pressure Relief may be performed, and why?

Containment pressure relief...

a. CAN be performed with a WGDT release in progress because both releases are simultaneously monitored by the 2R41 monitor.

b. CAN be performed with a WGDT release in progress because each release path has its own Rad Monitor to isolate its specific release path.

c. CANNOT be performed with a WGDT release in progress because the 2R41D does NOT isolate both releases on a high radiation signal from either release path.

d. CANNOT be performed with a WGDT release in progress because the postulated combined activity from a fuel element failure and the shortest decay time of the GDT prior to release exceeds 10CFR20 assumptions.

Answer	a	**Exam Level**	R	**Cognitive Level**	Memory	**Facility:**	Salem 1 & 2	**Exam Date:**	12/11/2006
Tier:	Generic Knowledge and Abilities	**RO Group**	1	**SRO Group**	1				194001G311
GENERIC								**Record Number**	74
2.3	Radiation Control								
2.3.11	Ability to control radiation releases.							2.7	3.2

Explanation of Answer: A is correct and distracters b and c are incorrect because the R41D will isolate both the WG41 AND the VC1-6 on a high radiation signal. The dose is monitored as per FSAR 9.4.1.1.6 to prevent exceeding 10CFR20. The postulated activity has no consequence as long as the release can be automatically isolated.

Reference Title
Discharge of 21 Gas Decay Tank to Plant Vent
Containment Pressure-Vacuum Relief System Operation

Learning Objectives	
WASGASE011	Identify the differences between Unit 1 and Unit 2 Radioactive Waste Gas System components, parameters, and operation. a) Not applicable to this lesson

Material Required for Examination:

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

Question Source Comments: Modified a distracter that said you couldn't perform the release because the procedure prohibits it, to the one about the postulated off site dose.

Given the following conditions:

- Unit 2 is operating at 100% power.
- A release of 21 WGDT is in progress, 2WG41 is OPEN.
- Containment pressure is 0.21 psid.

Which choice states whether or not a Containment Pressure Relief may be performed, and why?

Containment pressure relief...

a. CAN be performed with a WGDT release in progress because both releases are simultaneously monitored by the 2R41 monitor.

b. CAN be performed with a WGDT release in progress because each release path has its own Rad Monitor to isolate its specific release path.

c. CANNOT be performed with a WGDT release in progress because the 2R41D does NOT isolate both releases on a high radiation signal from either release path.

d. CANNOT be performed with a WGDT release in progress because the postulated combined activity from a fuel element failure and the shortest decay time of the GDT prior to release exceeds 10CFR20 assumptions.

Answer	a	Exam Level	R	Cognitive Level	Memory	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Generic Knowledge and Abilities	RO Group	1	SRO Group	1				194001G311
GENERIC								Record Number	75

2.3	Radiation Control		
2.3.11	Ability to control radiation releases.	2.7	3.2

Explanation of Answer	A is correct and distracters b and c are incorrect because the R41D will isolate both the WG41 AND the VC1-6 on a high radiation signal. The dose is monitored as per FSAR 9.4.1.1.6 to prevent exceeding 10CFR20. The postulated activity has no consequence as long as the release can be automatically isolated.

Reference Title
Discharge of 21 Gas Decay Tank to Plant Vent
Containment Pressure-Vacuum Relief System Operation

Learning Objectives	
WASGASE011	LOR NCT Discuss the procedural requirements associated with the Radioactive Waste Gas System, including an explanation of major precaution and limitations in the Radioactive Waste Gas System procedures

Material Required for Examination			
Question Source:	Facility Exam Bank	Question Modification Method:	Editorially Modified
Question Source Comments:	Modified a distracter that said you couldn't perform the release because the procedure prohibits it, to the one about the postulated off site dose.		

Given the following conditions:

- Unit 2 is in MODE 5.
- 21 RHR pump is in service for shutdown cooling.
- A large fire is reported in 21 RHR pump room.

Which of the following describes the required action which must be performed IAW S2.OP-AB.FIRE-001, CONTROL ROOM FIRE RESPONSE, and why?

a. At 2RP2, select FIRE INSIDE CONTROL AREA to maintain the control room habitable.

b. RHR cooling must be terminated in order to transfer shutdown cooling to 22 RHR pump. Initiate S2.OP-AB.RHR-001, Loss of RHR.

c. Isolate the PZR PORVs from the control room by manually closing PZR PORVs 2PR1 and 2PR2, and PZR PORV BLOCK VALVES 2PR6 and 2PR7, for RCS inventory and pressure control.

d. Isolate RHR and the RCS from the containment sump by stopping 21 RHR pump and closing 2RH1, 2RH2, 2SJ69, 21RH4 and 22RH4 to prevent spurious valve operation which could drain the RCS to containment.

Answer d **Exam Level** R **Cognitive Level** Memory **Facility:** Salem 1 & 2 **Exam Date:** 12/11/2006

Tier: Generic Knowledge and Abilities **RO Group** 1 **SRO Group** 1 194001G427

GENERIC **Record Number** 76

2.4 Emergency Procedures / Plan

2.4.27 Knowledge of fire in the plant procedure. 3.0 3.5

Explanation of Answer

Distracter c is incorrect because PORVs are isolated if the fire is in the relay room or control area. Distracter a is incorrect because Fire Inside Control Room is not required, fire OUTSIDE control room is. Distracter b is incorrect because RHR cooling cannot be transferred to the other pump for the same reason.
The correct answer is d because S2.OP-AB.FIRE-1 directs the isolation of the RCS-RHR from the containment sump because the cabling for the SJ44s and the RH4s runs in the room, and spurious hot short could cause these valves to open.

Reference Title

Control Room Fire Response

Learning Objectives

ABFP1E003 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:** Direct From Source

Question Source Comments: VISION Q63516

Given the following conditions:

- Unit 2 is operating at 100% power.
- An interior Control Bank rod drops fully into the core.
- A Rx trip signal is not generated, NOR is it required.

Which of the following alarms is NOT consistent with these conditions?

a. P-250 Computer Alarm.

b. OHA E-48, ROD BOTTOM.

c. OHA E-24, ROD DEV OR SEQ.

d. Rod Control NON-URGENT FAILURE 2CC2 Bezel Alarm.

Answer	d	Exam Level	R	Cognitive Level	Memory	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Generic Knowledge and Abilities		RO Group	1	SRO Group	1			194001G446
GENERIC								Record Number	77

2.4	Emergency Procedures / Plan		
2.4.46	Ability to verify that the alarms are consistent with the plant conditions.	3.5	3.6

Explanation of Answer

C is incorrect because the alarm is driven by ANY rod more than 12 steps deviation from its group counter. B is incorrect because it is driven by ANY rod being <20 step with Control Bank D > 35 steps. A is incorrect because the P-250 computer will alarm for both the deviation and rod bottom alarms. d is correct because the non urgent failure is driven from the loss of DC power supplies, none of which would cause any rods to drop.

Reference Title

Overhead Annunciators Window E

Control Console 2CC2

Learning Objectives

RODS00E011	Identify and describe the Control Room controls, indications, and alarms associated with the Rod Control and Position Indication Systems, including: The Control Room location of Rod Control and Position Indication Systems control bezels and indications (Licensed Operator & STA only) The function of each Rod Control and Position Indication Systems Control Room control and indication (Licensed Operator & STA only) The effect each Rod Control and Position Indication Systems control has upon Rod Control and Position Indication Systems components and operation (Licensed Operator & STA only) The plant conditions or permissives required for Rod Control and Position Indication Systems Control Room controls to perform their intended function (Licensed Operator & STA only) The setpoints associated with the Rod Control and Position Indication Systems 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 2 is operating at 100% power.
- A fire has broken out in the 104 panel, and has spread to the overhead.
- Due to the location of the fire, Fire Protection cannot control the fire.

Which of the following identifies an action that will be taken IAW S2.OP-AB.FIRE-0002 FIRE DAMAGE MITIGATION, and why?

a. Place the CVCS cross-connect in service from Unit 1 to supply RCP seal injection.

b. Stop ALL RCP's stopped to ensure only heat being added to RCS is from decay heat.

c. Close both PZR PORVs and block valves to prevent potential loss of RCS inventory and RCS pressure control.

d. Stop ALL CCW pumps stopped to ensure the CCW system is available to support achieving and maintaining HSB conditions within 24 hours of the fire event.

Answer	a	Exam Level	R	Cognitive Level	Comprehension	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Generic Knowledge and Abilities	RO Group	1	SRO Group	1				194001G448
GENERIC								Record Number	78

2.4	Emergency Procedures / Plan		
2.4.48	Ability to interpret control room indications to verify the status and operation of system, and understand how operator actions and directives affect plant and system conditions.	3.5	3.8

Explanation of Answer

B is incorrect because RCP's are stopped to prevent damage due to potential loss of CCW cooling. A is correct. C is incorrect because PORVs and block valves will only be shut in AB.FIRE-001 if fire is in the relay room or control room area. (Step 3.18 and 3.19) D is incorrect because CCW pumps will be stopped in AB.Fire-2 if fire is in Aux Bldg 64', but the reason is so CSD can be achieved in 72 hours..

Reference Title

Fire Damage Mitigation

Learning Objectives

FIRPROE012	Discuss the procedural requirements associated with the Fire Protection System, including an explanation of major precaution and limitations in the Fire Protection System procedures. (Licensed Operator only)
CVCS00E008	LOR Identify and describe the Control Room controls, indications, and alarms associated with the Chemical and Volume Control System, including: a) The Control Room location of Chemical and Volume Control System control bezels and indications (N/A NEO) b) The function of each Chemical and Volume Control System Control Room control and indication (N/A NEO) c) The effect each Chemical and Volume Control System control has upon Chemical and Volume Control System components and operation (N/A NEO) d) The plant conditions or permissives required for Chemical and Volume Control System Control Room controls to perform their intended function e) The setpoints associated with the Chemical and Volume Control System control room alarms.

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Direct From Source

Question Source Comments:

KEY

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

Applicant Information

Name:	Region: I
Date: 12/18/2006	Facility: Salem 1 & 2
License Level: SRO	Reactor Type: W
Start Time: KEY	Finish Time:

Instructions

Use the answer sheets provided to document your answers. Staple this cover sheet On top of the answer sheets. The passing grade requires a final grade of at least 70.00 percent on the SRO section of the examination, 80.00 percent on the RO section, and a combined grade of at least 80.00 percent. Examination papers will be collected EIGHT hours after the examination starts.

Applicant Certification

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

KEY

____
Applicant's Signature

Results

Examination Value	________ Points
Applicant's Score	________ Points
Applicant's Grade	________ Percent
SRO/RO Combined Score	________ Points
SRO/RO Combined Grade	________ Percent

Given the following conditions:

- Salem Unit 1 has experienced a LBLOCA coincident with numerous equipment failures and losses of power.
- A majority of CETs have exceeded 1200 deg. F.
- 1-EOP-LOCA-1 was in progress when a transition to 1-FRCC-1, RESPONSE TO INADEQUATE CORE COOLING was made
- 1-FRCC-1 has been ineffective at lowering CET temperatures.
- The TSC is activated.

Which of the following describes how this condition will be addressed?

a. Return to Step 1 of FRCC-1 and continue in a "do" loop until any action has reduced CET temperatures less than 1200 deg. F.

b. Return to LOCA-1 proedure in effect until transfer to HL recirc is required while continuing any available mitigation actions.

c. Transition to SAMG-CRG-1 CONTROL ROOM INTIAL RESPONSE FOR SEVERE ACCIDENT, since the normal EOP network has been ineffective at protecting the core.

d. Transition directly to SAMG-CRG-2 CONTROL ROOM INTIAL RESPONSE FOR SEVERE ACCIDENT-TSC ACTIVATED since additional protective actions are required IAW the Guide.

Answer	c	Exam Level	S	Cognitive Level	Comprehension	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Emergency and Abnormal Plant Evolutions			RO Group	1	SRO Group	1		000011A208
011	Large Break LOCA							Record Number	1
EA2.	Ability to determine and interpret the following as they apply to Large Break LOCA:								
EA2.08	Conditions necessary for recovery when accident reaches stable phase							3.4*	3.9*

Explanation of Answer

55.43(5) This question is designed to test the candidates ability to determine when the accident can NOT be recovered from by using the normal EOP network. In this sense it meets the K/A since knowing when the accident is essentially "non-recoverable" is logically tied to conditions which would let a normal recovery happen. With CET temps >1200 degrees in FRCC, the transition is made to SAMG-CRG-1 at step 27. The only way into SAMG-CRG-2 is to enter SAMG-CRG-1 first.

Reference Title

Severe Accident Mitigation Guidelines

Learning Objectives

FRCC00T001	Given a set of plant conditions, take corrective actions for an inadequate core cooling condition, in accordance with 2-EOP-FRCC-1

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 in service.
- At 0200, an automatic VCT makeup occurs.
- At 0400, another auto makeup occurs.
- PZR level and charging flow are stable and have remained constant.

Which of the following describes what is occurring, and what procedure should be implemented?

a. A > 1 gpm RCS leak, S2.OP-AB.RC-0001, REACTOR COOLANT SYSTEM LEAK.

b. 2LT-112 has failed to 14%, S2.OP-AR.ZZ-0012, 2CC2 CONTROL CONSOLE Alarm Response.

c. A < 2 gpm leak on the 21 charging pump discharge check valve flange, S2.OP-AB.CVC-0001 LOSS OF CHARGING.

d. PZR Master Flow Controller setpoint has drifted high, S2.OP-SO.CVC-0001, CHARGING, LETDOWN, AND SEAL INJECTION.

Answer	c	Exam Level	S	Cognitive Level	Comprehension	Facility:	Salem 1 & 2	Exam Date:	12/11/2006

Tier:	Emergency and Abnormal Plant Evolutions	RO Group	1	SRO Group	1	000022A202
022	Loss of Reactor Coolant Makeup				Record Number	2

AA2.	Ability to determine and interpret the following as they apply to Loss of Reactor Coolant Makeup:		
AA2.02	Charging pump problems	3.2	3.7

Explanation of Answer

55.43(5) A is incorrect because both PZR level and charging flow would have changed to compensate for the loss of fluid from the system. B is incorrect because the auto makeup would never stop since it would see 14% level continuously. C is correct because wih the leak on the discharge of the pump but upstream of the CV-55, charging flow would remain the same, and PZR level would remain constant. The loss would be seen as VCT level. Using 20 gallons per % in the VCT, and the makeup band of 14-24%, the system is losing 10% every 2 hours, or 200 gallons evey 2 hours, or 100 gal/hour, or 1.67 gpm. D is incorrect because charging flow would have risen and PZR level would have gone up.

Reference Title
Loss of Charging
Charging letdown and seal in

Learning Objectives	
CVCS00E012	Describe the procedures which govern the operation of the Chemical and Volume Control System, including significant prerequisites and precautions associated with each operating procedure which are required to be considered by either Licensed or Non-Licensed Operators.

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

Given the following conditions:

- Unit 1 is operating at 100% power when 12 SGFP trip.
- The Main Turbine runs back as expected.
- The RO is unable to initiate a normal boration.
- Operators receive OHA E-16, ROD INSERT LMT LO-LO
- Control Bank D position is 75 steps.
- Reactor power is stable at 64%.
- RCS boron concentration is 650 ppm.

Using the attached REM figures, determine the LEAST amount of time a rapid boration through 1CV175 is required IAW S1.OP-SO.CVC-0008, RAPID BORATION, in order to clear OHA E-16?

a. 5 minutes.

b. 8 minutes.

c. 23 minutes.

d. 30 minutes.

Answer	b	Exam Level	S	Cognitive Level	Application	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Emergency and Abnormal Plant Evolutions	RO Group	2	SRO Group	2				000024A205
024	Emergency Boration							Record Number	3

AA2. Ability to determine and interpret the following as they apply to Emergency Boration:

AA2.05 Amount of boron to add to achieve required SDM 3.3 3.9

Explanation of Answer

55.43(6) Using REM Figure 14, the RIL at 64% power is 85 steps. The ARP for RIL lo-lo states that rods must be withdrawn at least 2 steps past the limit to clear the alarm. With rods at 75 steps from stem, rods must be withdrawn 12 steps to the 87 step position. Using REM figure 4, the reactivity from this 12 step movement is ~100 pcm. Using REM Figure 13, the differential rod worth for 650 ppm is -6.9 pcm/ppm. 100 pcm/-6.9 pcm.ppm = 14.49, rounded up to 15 ppm. A is incorrect because it is the time on the chart for a 10 ppm boration. B is correct because IAW chart on page 3 of procedure 5 minutes are required for 10 ppm change. Therefore, a 15 ppm change would require 7.5 minutes. So 8 minutes would be the least amount of time of the choices presented to inject enough boron. C is incorrect because it is the amount of boron added if there were 15 steps of control bank misalignment. D is incorrect because it is the time required if injection was from the RWST.

Reference Title

Reactor Engineering Manual

Rapid Boration

Learning Objectives

CVCS00E012	Describe the procedures which govern the operation of the Chemical and Volume Control System, including significant prerequisites and precautions associated with each operating procedure which are required to be considered by either Licensed or Non-Licensed Operators.

Material Required for Examination: REM Figures 4,13,18. S1.OP-SO.CVC-0008

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

Question Source Comments:

Given the following conditions:

- Unit 2 has experienced a LBLOCA.
- While performing Faulted SG Evaluation at Step 1 of 2-EOP-LOCA-1, LOSS OF REACTOR COOLANT, ALL off-site power is lost.
- 2C EDG reenergizes 2C 4KV Vital bus.
- 2A and 2B EDGs start but BOTH their Vital busses are locked out on Bus Differential.

Which of the following identifies the correct procedure flowpath, starting from when off-site power was lost?

a. IMMEDIATELY GO TO LOCA-5, LOSS OF EMERGENCY RECIRCULATION, and refer to S2.OP-AB.LOOP-0001, LOSS OF OFFSITE POWER

b. Reset SI and SECs in LOCA-1, start ECCS pumps that stopped when SEC C reloaded in MODE II, GO TO LOCA-3 at 15.2' RWST level, return to LOCA-1.

c. Remain in LOCA-1 until transition is required. If LOCA-3 is entered go IMMEDIATELY to LOCA-5, otherwise go directly to LOCA-5, perform in its entirety, transition to IOP-6, HOT STANDBY TO COLD SHUTDOWN.

d. Initiate S2.OP-AB.LOOP-0001 while continuing in LOCA-1. At 15.2' RWST level, GO TO LOCA-3, and perform all action up until starting of RHR pumps is required. GO TO LOCA-5, complete applicable actions, and return to LOCA-1.

Answer	c	Exam Level	S	Cognitive Level	Application	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Emergency and Abnormal Plant Evolutions	RO Group	1	SRO Group	1				000025G416
025	Loss of Residual Heat Removal System					Record Number			4
2.4	Emergency Procedures / Plan								
2.4.16	Knowledge of EOP implementation hierarchy and coordination with other support procedures.							3.0	4.0

Explanation of Answer

55.43(5) A is incorrect because LOCA-5 is only entered from LOCA-1 at Step 16, not immediately when 2 RHR pumps are not available. B is incorrect because the Loss of Offsite power occurred before the SI and SEC's were reset, so the SECs never loaded in MODE II (Blackout). C is correct because once in LOCA-3, the CAS action to go to LOCA-5 should be performed immediately to conserve RWST level and initiate makeup. D is incorrect because there is no transition prior to RWST level of 15.2' or Step 16 of LOCA-1 to go anywhere else with a LBLOCA and no other even t in progress. Once in LOCA-3

Reference Title

Loss of Emergency Recirculation

Loss of Reactor Coolant

Transfer to Cold Leg Recirculation

Learning Objectives

LOCA00T005	perform actions for a Loss of Emergency Recirculation IAW EOP-LOCA-5
LOCA00T001	perform actions for a loss of reactor coolant IAW EOP-LOCA-1
LCA3U2T001	Given a set of plant conditions, perform actions for a Transfer to Cold Leg Recirculation in accordance with 2-EOP-LOCA-3

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

Given the following conditions:

- Unit 2 is operating at 100% power with a leaking fuel pin.
- The specific activity of the RCS has been ~ 0.3 uCi/gram DOSE EQUIVALENT IODINE for 1 week.
- A radiation protection technician reports the latest RCS sample indicates that specific activity has jumped to 70 uCi/gram.
- Prior to any action being taken, a 300 gpm tube rupture occurs on 22 SG
- A MANUAL Rx trip and a MANUAL Safety Injection were initiated successfully.
- IMMEDIATELY following the reactor trip, 22MS10, SG Atmospheric Relief Valve failed open.
- Operators cannot enter the affected penetration area to manually isolate the malfunctioning valve until TWO hours have passed.

Which of the following describes how radiological conditions will be affected by this failure?

a. A Qualified Radiological Worker inside the Protected Area exposed to the entire release would exceed the Salem Administrative Dose Control limit of 500 DAC-hours (1250 mrem/year CEDE).

b. A person located at any point on the outer boundary of the low population zone during the entire time of the release may be exposed to more than an acceptable portion of the 25 Rem whole body dose limit.

c. The malfunctioning MS10 will cause RCS temperature to drop below 500 degrees, which ensures the resulting 2 hour dose at the site boundary will not exceed an appropriately small fraction of 10CFR100 limits.

d. A person located at any point of the Exclusion Area boundary for the 2 hours immediately following the fission product release may receive more than the 10CFR100 limit of 50 Rem to the thyroid from iodine exposure.

Answer	b	Exam Level	S	Cognitive Level	Application	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Emergency and Abnormal Plant Evolutions		RO Group	1	SRO Group	1			000038G304
038	Steam Generator Tube Rupture							Record Number	5
2.3	Radiation Control								
2.3.4	Knowledge of radiation exposure limits and contamination control, including permissible levels in excess of those authorized.							2.5	3.1

Explanation of Answer

55.43(4) The bases for reducing RCS temp to <500 degrees when RCS specific activity exceeds its limits is that in case of a SGTR, the RCS will be below the saturation pressure for the SG Atmospheric Relief valves automatic lift setpoint. In the case presented above, the MS10 valve is failed open. C is incorrect because of the above statement, even though the site boundary part of the distracter is correct for the bases of specific activity. A is incorrect because the dose control limit is 200 DAC-hours (500 mrem/yr CEDE), and its only a threshold for monitoring, not a control limit. B is correct because the 10CFR100 limits assume a LOCA, and the SGTR scenario with specific activity in the RCS is to limit exposure to a small fraction of those limits. With the failed open atmospheric release, that assumption cannot be made. D is incorrect because thyroid limit is 300 Rem.

Reference Title
Tech Specs
Radiation Protection Program
Code of Federal Regulations

Learning Objectives	
RADCONE002	List the following external radiation exposure limits, in accordance with NC.NA-AP.ZZ-0024(Q), 10CFR20, and Reg. Guide 8.13: A. 10CFR20 dose limits for external, internal, and total whole body, skin, extremities, and eyes, as well as extension limits and requirements

B. Administrative dose control levels for Category 1 and 2 Workers, as well as extension limits and requirements
C. Reg. Guide 8.13 limits and administrative dose control levels for Declared Pregnant Women
D. 10CFR20 and Administrative limits for members of the general public and minors
E. Category 1 Radiation Worker
F. Category 2 Radiation Worker

Material Required for Examination	

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

Given the following conditions:

- Unit 1 is operating at 100% power.
- A steam leak is identified and the CRS orders the reactor tripped.

Which of the following conditions would require a Safety Injection to be MANUALLY initiated after the Rx trip is attempted, and what procedure would require the Safety injection?

a. The reactor trips and steam flow only drops to 5%; S1.OP-AB.STM-0001, Excessive Steam Flow.

b. The reactor trips and containment pressure is rising; S1.OP-AB.STM-0001, Excessive Steam Flow.

c. The reactor does NOT trip from the Control Room; 1-EOP-FRSM, Response to Nuclear Power Generation.

d. The reactor does NOT trip until the Rx Trip Breakers are opened from 1CC2; 1-EOP-TRIP-1, Reactor Trip Response.

Answer	b	Exam Level	S	Cognitive Level	Application	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Emergency and Abnormal Plant Evolutions	RO Group	1	SRO Group	1				000040A204
040	Steam Line Rupture							Record Number	6

AA2.	Ability to determine and interpret the following as they apply to Steam Line Rupture:		
AA2.04	Conditions requiring ESFAS initiation	4.5	4.7

Explanation of Answer

55.43(5), A is incorrect because the initial post trip steam flow is expected to be 5% on any normal Rx trip. B is correct because after the reactor trip, containment pressure rising indicates the steam leak/rupture is located in containment and is unisolable, and aB.STM requires a safety injection if the steam leak/rupture is NOT isolated. (CAS Step 1.1.C and D). C is incorrect because an ATWT will NOT require a Safety Injection, nor is it a wanted occurrence. D is incorrect because the Rx trip from the control console is only significant in that the initial Rx trip demand did not occur and an ECG call is required for failure to trip. If an AUTOMATIC SI were to occur while trying to get the Rx trip, a MANUAL SI would not be required in TRIP-1

Reference Title
Excessive Steam Flow
Response to Nuclear Power Generation
Reactor Trip Response

Learning Objectives	

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

Given the following conditions:

- Unit 1 is operating a 75% power.
- PZR pressure channel II is removed from service for calibration.
- An electrical fault causes the 500 KV switchyard to become deenergized.
- The PZR Master Pressure Controller (MPC) fails low, causing sprays to close and all heaters to energize.

If pressure rises above their lift setpoint, which of the following describes how this will affect PZR PORV operation?

a. ONLY 1PR1 will open. Since the PORVs are not designed to prevent exceeding RCS design pressure, one OPERABLE PORV is an acceptable plant configuration.

b. ONLY 1PR2 will open. Since the Rx has already tripped due to the Loss of Off-Site power, the PORVs are not necessary for plant control.

c. BOTH PORVs will open since the MPC does not control PORV operation. The plant will NOT exceed design parameters since two PORVs have enough relief capacity to prevent exceeding 2485 psig RCS pressure.

d. Neither PORV will open since the MPC has failed low. The plant will not exceed design parameters as long as the PZR Safety Valves function properly.

Answer	a	Exam Level	S	Cognitive Level	Application	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Emergency and Abnormal Plant Evolutions	RO Group	1	SRO Group	1				000056A201
056	Loss of Off-Site Power							Record Number	7
AA2.	Ability to determine and interpret the following as they apply to Loss of Off-Site Power:								
AA2.01	PORV controller indicator and setpoint							3.3*	3.4

Explanation of Answer: 55.43(2) A is correct because the PR2 will not open since it is a 2/2 coincidence and it has one of its channels out for calibration. The PORV's are designed to prevent PZR pressure from reaching the High Pressure Rx trip, not for exceeding design pressure. B is incorrect because of (a) above, even though it has the right reason. C is incorrect because of (a) above, and also that 2 PORV's are NOT designed to keep RCS pressure less than 2485 psig. D is incorrect because the MPC does not directly control PORV operation. Each PORV is 2/2 to operate on alternate PZR pressure channels. 1 and 3 , and 2 and 4.

Reference Title
Technical Specifications and Bases
Reactor Protection System PZR pressure and level control

Learning Objectives	
PZRP&LE002	Describe the design bases of the Pressurizer Pressure and Level Control system

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 a steam leak upstream of 22MS167 occurred.
- The Rx was tripped and a MSLI performed successfully.
- Operators have transitioned out of EOP-TRIP-1.
- The PO is attempting to open 21-24SS94s, SG B/D Sample Valves, but they will not open.
- SGBD sample isolation bypass has been RESET.

Which of the following conditions identifies the reason the valves won't open?

a. 22 SG NR level is <9%.

b. SI was not reset properly.

c. Phase A isolation failed to reset.

d. CA330s have not been reopened.

Answer	c	Exam Level	S	Cognitive Level	Memory	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Emergency and Abnormal Plant Evolutions	RO Group	1	SRO Group	1				00WE05A202
E05	Loss of Secondary Heat Sink							Record Number	8

EA2.	Ability to determine and interpret the following as they apply to Loss of Secondary Heat Sink:		
EA2.2	Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.	3.7	4.3

Explanation of Answer

55.43(4) SGBD Sample Isolation bypass requires deliberate action to open SG Sample valves to prevent the spread of contamination. For a trip and SI due to a single faulted SG (unisolable) the flow path will go from TRIP-1 to LOSC-1. B is incorrect because the SI will NOT have been reset in TRIP-1, nor will it be reset in LOSC-1. C is correct because the SGBD sample isolation reset will be performed in LOSC-1 (step 6.1) in order to open the SS94's. The step prior to that is RESET PHASE A. This is due to the fact that the blowdown isolation bypass only bypasses the lo-lo level input into the AFW auto start circuit, which closes the SS94's. If the Phase A hasn't been reset, the 94s can not be reopened. D is incorrect because the SS94s supplied air from outside cont

Reference Title

Loss of Secondary Coolant

Learning Objectives

LOSC01E005	Identify possible radioactivity release paths during a loss of secondary coolant event and describe how the actions of 2-EOP-LOSC-1 minimize the potential for a release

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

Which of the following choices identifies the mitigation strategy of EOP-FRCC-3 RESPONSE TO SATURATED CORE COOLING?

a. Repressurize the RCS to collapse voids, verify letdown isolation and establish charging.

b. Establish ECCS Injection flow to maintain minimum RCS subcooling and check for open RCS vent paths.

c. Verify/ establish PZR level > 17% to restore letdown and PZR heaters, energize heaters to raise RCS pressure.

d. Establish charging and letdown to stabile PZR level, ensure both PORVs are closed and PORV stop valves open.

Answer	b	Exam Level	S	Cognitive Level	Memory	Facility:	Salem 1 & 2	Exam Date:	12/11/2006

Tier:	Emergency and Abnormal Plant Evolutions	RO Group	2	SRO Group	2	00WE07A202
E07	Saturated Core Cooling				Record Number	9

EA2.	Ability to determine and interpret the following as they apply to Saturated Core Cooling:		
EA2.2	Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.	3.3	3.9

Explanation of Answer: 55.43(5) b comes out of the Att. 1 of 2-EOP-FRCC-3 basis document. All distracters are convolutions of other procedures mitigating actions.

Reference Title

Response to Saturated Core Cooling

Learning Objectives

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:	Facility Exam Bank	Question Modification Method:	Direct From Source
Question Source Comments:	VISION Q48662		

Which of the following choices identifies the condition which would result in the highest priority CFST?

a. Rx tripped 15 minutes ago, 21-23 SG levels are 4% NR, 24SG level is 80% NR.

b. Rx is tripped from 100% power, IR SUR is +0.1 dpm, Rx power is 5x10-9 Amps, PZR level is 22%.

c. Rx trip and SI from 100% power due to LOCA, ALL RCPs stopped, RVLIS Full Range 35%, highest CET in each quadrant reading 600 deg.

d. Rx was tripped from 80% power due to steam rupture, ALL RCPs are stopped, RCS Tc's are 230 deg. RCS pressure is 1200 psig.

Answer	d	Exam Level	S	Cognitive Level	Memory	Facility:	Salem 1 & 2	Exam Date:	12/11/2006

Tier:	Emergency and Abnormal Plant Evolutions	RO Group	1	SRO Group	1	00WE08A201
E08	Pressurized Thermal Shock				Record Number	10

EA2.	Ability to determine and interpret the following as they apply to Pressurized Thermal Shock:		
EA2.1	Facility conditions and selection of appropriate procedures during abnormal and emergency operations.	3.4	4.2

Explanation of Answer

55.43(5) A is incorrect because it is a YELLOW priority CFST. B is incorrect because it is a PURPLE path. C is incorrect because it is a PURPLE Path CFST. D is correct because it is the only RED path CFST. The Figure 4A Thermal Shock Limit Curve shows that ant cold leg temperature of 230 deg with pressure > 0 psig is to the left of Limit A. The combination of > 100 deg/hr cooldown rate, and pressure temp to the left of Limit A results in RED path.

Reference Title

Critical Safety Function Status Trees

Learning Objectives

FRTS00E001	State the RED paths for the Thermal Shock Status Tree

Material Required for Examination

Question Source:	Facility Exam Bank	Question Modification Method:	Editorially Modified
Question Source Comments:	Changed 2 distracters to make them PURPLE path. Correct answer remains correct.		

Given the following conditions:

- Unit 1 is at 100% power.
- With SSPS testing and troubleshooting in progress a Phase B Containment Isolation signal was generated and all related valves closed.
- Before operators could re-open any of the Phase B valves, the operating Charging Pump breaker tripped on an electrical fault.

Which of the following describes the required operator actions?

a. Immediately start the other Charging Pump and monitor RCP bearing and seal inlet temps.

b. Initiate a MANUAL reactor trip and stop all RCP's if Phase B can NOT be reset. RCP's can be re-started anytime after seal injection has been restored.

c. Start the other Charging Pump OR restore CCW to the thermal barrier within five minutes or initiate a MANUAL reactor trip.

d. Initiate a MANUAL reactor trip and stop all RCP's. Cooldown to desired temperature IAW EOP-TRIP-4, NATURAL CIRCULATION COOLDOWN.

Answer	d	Exam Level	S	Cognitive Level	Memory	Facility:	Salem 1 & 2	Exam Date:	12/11/2006

Tier:	Emergency and Abnormal Plant Evolutions	RO Group	1	SRO Group	1	00WE09A201

E09	Natural Circulation Operations	Record Number	11

EA2. Ability to determine and interpret the following as they apply to Natural Circulation Operations:

EA2.1	Facility conditions and selection of appropriate procedures during abnormal and emergency operations.	3.1	3.8

Explanation of Answer: 55.43(5) A is incorrect because there is no procedural direction to start another charging pump after a loss of seal injection AND loss of Thermal Barrier Cooling flow. B is incorrect because RCP's cannot be restarted anytime, seal inlet temps must be below the threshold for hot seal. C is incorrect because there is no 5 minute timer for actions. IMMEDIATELY go to Att 2 of AB.RCP to trip Rx and RCPs. D is correct because the actions are correct per AB.RCP, and the cooldown will have to be performed naturally circulated.

Reference Title

Reactor Coolant Pump Abnormality

Natural Circulation Cooldown

Learning Objectives

TRP004T001	Given a set of plant conditions, perform actions for a Natural Circulation Cooldown in accordance with EOP-TRIP-4
ABRCP1E004	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:	Reworded distracters for psychometric attributes.		

Given the following conditions:

- Unit 2 has experienced a Large Break Loss of Coolant Accident.
- The Reactor trip and Safety Injection occurred successfully.
- 2-EOP-LOCA-1 LOSS OF REACTOR COOLANT is in effect.
- PZR pressure is 35 psig.
- 1 CET is reading 900°F, ALL other CET's are reading ~550°F.
- RVLIS Full Range is reading 74%.
- Containment pressure is 13 psig.
- Containment sump level is 62%.
- R44A radiation monitor is indicating 50 R/hr.

Which choice identifies a procedural transition that is allowed under these conditions?

a. FRCI-3, RESPONSE TO VOID IN REACTOR VESSEL.

b. FRCC-2, RESPONSE TO DEGRADED CORE COOLING.

c. FRCE-1, RESPONSE TO EXCESSIVE CONTAINMENT PRESSURE.

d. FRCE-3, RESPONSE TO HIGH CONTAINMENT RADIATION.

Answer	d	**Exam Level**	S	**Cognitive Level**	Memory	**Facility:**	Salem 1 & 2	**Exam Date:**	12/11/2006
Tier:	Emergency and Abnormal Plant Evolutions	**RO Group**	2	**SRO Group**	2				00WE16A202
E16	High Containment Radiation							**Record Number**	12

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	55.43.(5) D is correct because the YELLOW path entry requirements are met of 2R44Rad level > 2R/hr and no other FRCE conditions present. Distracter A is incorrect because PZR level will be offscale low with LBLOCA. Distracter B is incorrect because 5 or more CET's are NOT > 700 and RVLIS level is NOT < 39%. Distracter C is incorrect because containment pressure is < 15 psig.

Reference Title
Response to High Containment Radiation
Critical Safety Function Status Trees

Learning Objectives	
FRCE00T003	Given a set of plant conditions, take corrective actions for an high containment radiation in accordance with 2-EOP-FRCE-3

Material Required for Examination			
Question Source:	Previous 2 NRC Exams	**Question Modification Method:**	
Question Source Comments:	"H" NRC SRO Exam June 2004		

Given the following conditions:

- Unit 1 is in Mode 5 with 11 Residual Heat Removal (RHR) pump in service for cooling.
- The RO reports that Pressurizer (PZR) level is slowly lowering unexpectedly.
- NO Overhead Annunciator OR Auxiliary Typewriter alarms have been received.
- Refueling Water Storage Tank (RWST) level is stable.
- 11 Waste Hold Up Tank level is rising slowly.

Which of the following describes:

1. The effect this would have if NO operator action were to be taken.
2. What procedure and action would terminate this problem.

a. Eventual caviation and gas binding of 11 RHR pump. Shut 1CV8 IAW S1.OP-AB.RHR-0001, LOSS OF RHR.

b. Eventual caviation of 11 RHR pump and gas binding of BOTH RHR pumps. Close 1CV132, Excess Letdown IAW S1.OP-SO.CVC-0003, EXCESS LETDOWN FLOW.

c. Loss of pressure control when the PZR heaters deenergize. Remove 11 RHR Loop from service and put 12 RHR loop in service IAW S1.OP-SO.RHR-0001, INITIATING RHR.

d. Dilution of the RCS due to CVCS Make-up System repeated auto make-ups. Restore PZR level by performing MANUAL borated make-ups IAW S1.OP-SO.CVC-0006, BORON CONCENTRATION CONTROL.

Answer	a	Exam Level	S	Cognitive Level	Comprehension	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Plant Systems			RO Group	1	SRO Group	1		005000A204
005	Residual Heat Removal System							Record Number	13

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:		
A2.04	RHR valve malfunction	2.9	2.9

Explanation of Answer

55.43(5) A is correct because if the leak were to continue, cavitation and gas binding would occur. AB.RHR-1 addresses the loss of inventory in MODE 5 while NOT at reduced inventory, and has the correct action to close the CV8. B is incorrect because the CV132 is not physically located at an elevation which could provide flow into the WHUT. C is incorrect because pressure control is provided by the RHR pump discharge, and putting the redundant loop in service would just cause it to become gas bound too. D is incorrect because dilution would not occur with blended makeups.

Reference Title

Loss of RHR

RHR system drawing

Learning Objectives

ABRHR1E005	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:	New	Question Modification Method:	
Question Source Comments:			

Given the following conditions:

- Unit 2 is operating at 100% power.
- ALL station Air Compressors trip.
- BOTH Units Emergency Control Air Compressors start.
- 2CC71 LTDWN HX CC CONT VALVE, sensed a low header pressure on its primary air supply, and transferred to its backup supply. When it transferred, the valve diaphragm failed, and the valve moved to its failed position.
- NO other air operated valves have been adversely affected by the air system perturbation.

Which of the following describes the effect this will have on the CVCS system, and what actions are required?

a. Letdown must be manually isolated due to the inability to control letdown temperature, and Excess Letdown must be placed in service IAW S2.OP-SO.CVC-0003, EXCESS LETDOWN FLOW.

b. VCT temperature will lower, causing less effective aeration of letdown flow into the VCT through the spray nozzle. Additional RCS lithium control adjustments will be required IAW SC.CH-AP.RC-0106, IMPLEMENTATION OF SALEM LITHIUM CONTROL PROGRAM.

c. Letdown temperature will rise until OHA E-41, LTDWN HX OUT TEMP HI alarm is received. CVCS Mixed Bed Demineralizers must be removed from service by manually repositioning the 2CV21, LTDWN DM BYP V from Control Console 2CC2 IAW S2.OP-AR.ZZ-0005, OVERHEAD ANNUNCIATORS WINDOW E.

d. Failure of the 2CC71 will cause the 2CV7, LTDWN HX INLET VALVE, to automatically close due to the OPEN interlock between the two valves. Letdown must be further isolated by closing 2CV2 and 2CV277, LETDOWN LINE ISOL VALVES, IAW S2.OP-SO.CVC-0001, CHARGING LETDOWN, AND SEAL INJECTION.

Answer	a	**Exam Level**	S	**Cognitive Level**	Application	**Facility:**	Salem 1 & 2	**Exam Date:**	12/11/2006
Tier:	Plant Systems			**RO Group**	1	**SRO Group**	1		008000A205
008	Component Cooling Water System							**Record Number**	14

A2.	Ability to (a) predict the impacts of the following on the Component Cooling Water System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:		
A2.05	Effect of loss of instrument and control air on the position of the CCW valves that are air operated	3.3*	3.5

Explanation of Answer

55.43(5) Lithium production is based on RCS boron concentration. The effect of the 2CC71 failing closed will cause letdown temp through the demineralizers to rise, and also cause VCT temperature to rise. The effect of rising temperature in the demineralizers from the normal inlet temperature up to 136 degrees, which is when the demineralizers are automatically bypassed with CV21, will cause the beds to change their boron affinity, but the effect is negligible. Also, the spray nozzle in the VCT is designed for better H2 absorption to scavenge O2 from Radiolysis. B is incorrect because VCT temperature will rise, not lower, and also because of the discussion above. C is incorrect because the 2CV21 will AUTOMATICALLY reposition when the OHA E-41 is received, and the stem stated no other AOVs were adversely affected. D is incorrect because the interlock works the opposite way, that is, when the CV7 is closed, it closes the CC71. A is correct because the ARP directs the manual letdown isolation and directs excess letdown be placed in service.

Reference Title

Overhead Annunciators Window E

Loss of Control Air

Learning Objectives	
CCW000E004	Describe the function and operating characteristics for the following Component Cooling Water System Components: a) CCW Surge Tank b) CCW Pumps c) CCW Heat Exchangers d) Isolation/Control Valves i) CC-190, RCP Thermal Bar Disch Valve ii) CC-117 & 118, RCP Cooling Water Inlet Valves iii) CC-136 & 187, RCP Bearing Cooing Outlet Valves iv) CC-215 & 113, Excess Letdown Hx CCW Inlet & Outlet Valves v) CC-16s, RHR Hx Outlet Isolation Valves vi) CC-17 & 18, CCW Pump Suction X-connect Valves vii) CC-3s, CCP Outlet Valves viii) CC-30 & 31, CCHX Outlet to Aux. Header (Non-safety Related Header Isolation Valves) ix) CC-71, Letdown Temperature Control Valve x) CC-149, Surge Tank Vent Valve xi) CC-131, RCP Thermal Barrier Discharge Flow Control Valve e) Radiation Monitors
CVCS00E004	Describe the function and operating characteristics for the following Chemical and Volume Control System components: a) Letdown/Charging i) Letdown Isolaiton Valves, CV2, CV277 ii) Regenerative Heat Exchanger iii) Letdown Orifices iv) Letdown Orifice Isolation Valves, CV3, CV4, CV5 v) Letdown Releif Valve, CV6 vi) Letdown Line Containment Isolation Valve, CV7 vii) RHR Flow Control Valve, CV8 viii) Letdown Heat Exchanger ix) Low Pressure Letdown Control Valve, CV18 x) Temperature Control Valve, CV21 xi) Demineralizers (Mixed Bed, Cation, and Deborating xii) Inlet Valve to Deborating Demin, CV27 xiii) Reactor Coolant Filter xiv) Diversion Valve, CV35 xv) CVCS Holdup Tanks xvi) Volume Control Tank xvii) VCT Isolation Valves, CV40, CV41 xviii) Chemical Mixing Tank xix) Charging Pumps (Centrifugal and PD) xx) Miniflow Recirc. Valves, CV139, CV140 xxi) Seal pressure Control Valve, CV71 xxii) Chg. Line Containment Isol. Valves, CV68, CV69 xxiii) Charging to Loop 3 Valve, CV77, Loop 4 Valve, CV79 xxiv) PZR Auxiliary Spray Valve, CV75 xxv) CCP Flow Control Valve, CV55 b) RCP Seal Water i) Seal Water Injection Filters ii) Seal Bypass Flow Valve, CV114 iii) Seal Water Return Isolation Valve, CV104 iv) Seal Water Return Relief Valve, CV115 v) Seal Return Cont. Isol. Valves, CV116, CV284 vi) Seal Return Filter vii) Seal Water Heat Exchanger c) Excess letdown i) Excess Letdown Isolation Valves, CV278, CV131 ii) Excess Letdown Heat Exchanger iii) Excess letdown Flow Cotrol Valve, CV132 iv) Excess Letdown Diversion Valve, CV134 d) Makeup i) Primary Water Storage Tank ii) Primary Water Makeup Pumps iii) Boric Acid Batch Tank iv) Boric Acid Tanks v) Boric Acid Transfer Pumps vi) Boric Acid Filter vii) Boric Acid Blender viii) Primary Water Flow Control Valve, CV179 ix) Boric Acid Flow Control Valve, CV172 x) Charging Pump Suction Valve, CV185 xi) VCT Makeup Isolation Valve, CV181 xii) Rapid Borate Stop Valve, CV175
CVCS00E006	Describe the interlocks associated with the following Chemical and Volume Control System components: a) VCT Isolation Valves, CV40 and CV41 b) Letdown Isolation Valves, CV2 and CV277 c) Letdown Orifice Isolation Valves, CV3, CV4 and CV5 d) Centrifugal Charging Pumps

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

Given the following conditions:

- Unit 1 is in Mode 6.
- Rx power is 100 cps on both SR channels.
- The Rx vessel upper internals are being put in place following core reload.
- Audible count rate indication is lost in the Control Room.
- Containment audible count rate is NOT lost.
- BOTH SR channels continue to indicate 100 cps.

PRIOR to taking any action IAW S2.OP-AB.NIS-0001, NUCLEAR INSTRUMENTATION SYSTEM MALFUNCTION, which of the following identifies the required action, if any, to be taken IAW Tech Specs?

a. No action is required since redundant audible indication remains available.

b. No action is required since containment audible indication was never interrupted.

c. Suspend CORE ALTERATIONS ONLY.

d. Suspend CORE ALTERATIONS and any positive reactivity additions in progress.

Answer	d	Exam Level	S	Cognitive Level	Memory	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Plant Systems	RO Group	1	SRO Group	1				012000G226
012	Reactor Protection System							Record Number	15
2.2	Equipment Control								
2.2.26	Knowledge of refueling administrative requirements.							2.5	3.7

Explanation of Answer

55.43(7) TSAS 3.9.2 states that BOTH SR channels shall be operating EACH with continuous visual indication in the control room, and ONE with audible indication in the containment and control room. The ACTION for one of the monitors INOPERABLE is to suspend core alts and positive reactivity additions. AB-NIS has operators select the other channel, but the stem states that no action have been performed in the AB yet. The TSAS ACTION must be taken since there is no audible indication in the control room. A is incorrect because the channel selector has not been transferred to the other channel yet, and there is no way of knowing if the Audio Count Rate Monitor itself is broken for control room audible indication. B is incorrect because the audible indication is required in both the containment AND the CR. C is incorrect because it encompasses only half the action required by TS. D contains the correct action for TSAS 3.9.2.

Reference Title

Technical Specifications

Nuclear Instrumentation System Malfunction

Learning Objectives

REFUELE010 — State the Technical Specifications associated with the components, parameters, and operation of the Refueling System, including:
a) The Limiting Condition(s) for Operation
b) The Bases for the LCO(s)
c) The applicability of the LCO(s)
d) The LCO Action Statement(s) (N/A NEO)

Material Required for Examination	
Question Source:	Facility Exam Bank
Question Modification Method:	Significantly Modified
Question Source Comments:	Modified VISION Q69728 to make a distracter the correct answer and the correct answer into a distracter.

Given the following conditions:

- Unit 2 is in MODE 5.
- OHA C-35 SFP LO Alarms.
- The NEO dispatched to investigate reports SFP level just below the alarm setpoint, and appears to be stable.
- No leak identification action has been initiated.

Which of the following describes the actions required for this condition?

a. Dispatch an operator to determine source of leak, and refill the SFP using demineralized water IAW S2.OP-SO.SF-0001, FILL AND TRANSFER OF THE SPENT FUEL POOL.

b. IMMEDIATELY GO TO S2.OP-AB.FUEL-0002,LOSS OF REFUELING CAVITY OR SPENT FUEL LEVEL , to isolate the SFP cooling pumps individually to isolate the most likely source of leakage.

c. Since occasional SFP low level alarms are to be expected due to the leak on the SFP liner, refill the SFP using CVCS HUT water if available to maintain boron concentration as high as possible IAW S2.OP-SO.SF-0001.

d. Monitor 2R5 and 2R32 SFP Area Radiation Monitors, which will initiate 22 HEPA PLUS CHAR mode of FHV IAW S2.OP-AB.FUEL-0002.

Answer	a	Exam Level	S	Cognitive Level	Memory	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Plant Systems		RO Group	2	SRO Group	2			033000A203
033	Spent Fuel Pool Cooling System							Record Number	16

A2.	Ability to (a) predict the impacts of the following on the Spent Fuel Pool Cooling System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:		
A2.03	Abnormal spent fuel pool water level or loss of water level	3.1	3.5

Explanation of Answer	55.43(7)(5) A is correct because there is no indication of a leak as per stable SFP level. B is incorrect because with the level just below the setpoint and no leakage indicated, going to AB.FUEL-2 is inappropriate and could cause problems if trying to isolate a phantom leak. C is incorrect since the CVCS HUT is the 3rd preferred source of makeup water to SFP behind demin water and PWST. D is incorrect because the 2R32 does not perform any automatic ventilation function, it stops outward crane movement.

Reference Title
Fill and Transfer of the Spent Fuel Pool
Overhead Annunciator Window C

Learning Objectives	
SFP000E012	Describe the procedures which govern the operation of the Spent Fuel Pool Cooling System, including significant prerequisites and precautions associated with each operating procedure which are required to be considered by either Licensed or Non-Licensed Operators

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

Unit 1 is operating at 80% power with all equipment operable and all plant parameters within their normal operating bands when 11MS10, fails open.

Which of the following describes the INITIAL system parameter response assuming NO operator action were taken initially and all effected control systems are in MANUAL, and which procedure would be most effective in responding to this event?

a. Tave lowers, PZR level rises, 11 SG NR level lowers; S1.OP-AB.STM-0001 EXCESSIVE STEAM FLOW.

b. Tave lowers, PZR level lowers, 11 SG NR level rises; S1.OP-AB.STM-0001 EXCESSIVE STEAM FLOW.

c. Rx power rises, PZR level rises, 211 SG NR level rises; S1.OP-AB.CN-0001, MAIN FEEDWATER/CONDENSATE SYSTEM ABNORMALITY.

d. Rx power rises, PZR level lowers, 11 SG NR level lowers; S1.OP-AB.CN-0001, MAIN FEEDWATER/CONDENSATE SYSTEM ABNORMALITY.

Answer	b	Exam Level	S	Cognitive Level	Comprehension	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Plant Systems	RO Group	2	SRO Group:	2				035000A201
035	Steam Generator System							Record Number	17

A2.	Ability to (a) predict the impacts of the following on the Steam Generator System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:		
A2.01	Faulted or ruptured S/Gs	4.5	4.6

Explanation of Answer

55.43(5) With all control systems in manual, the effect of a Main steam Atmospheric Relief valve failing open would be to raise steam flow from that SG, lower SG pressure, lower Tc of that loop. Lowering Tc would cause Tave to lower. Tave (auct hi) is the input to control PZR level, and while 21 loop may not be the auctioneered hi loop, ALL loops will be affected by any loop Tave dropping, and cause a corresponding drop in PZR level. The higher steam flow will cause 21 SG NR level to rise due to swell. AB Steam has specific steps to address malfunctioning MS10.

Reference Title

EXCESSIVE STEAM FLOW

Learning Objectives

ABSTM1E004	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:	Added procedure to choices to make 55.43		

Given the following conditions:

- Unit 1 is at 90% power steady state
- 14BF19 fails full closed over a period of 1 minute
- All other controls respond as expected

With NO operator action, which of the following responses will be apparent FIRST to the operators, and what procedure will be used to respond to this event?

a. BF19 DEMAND rises on unaffected SG's. S2.OP-AB.CN-0001, MAIN FEEDWATER / CONDENSATE SYSTEM ABNORMALITY

b. PZR B/U heaters turn ON in AUTOMATIC. S1.OP-AB.PZR-0001, PRESSURIZER PRESSURE MALFUNCTION.

c. SGFP Master Speed Controller demand signal rises. S1.OP-AR.ZZ-0012, CONTROL CONSOLE 1CC2 Alarm Response.

d. Reactor trip at 14% NR level on 2/3 channel on 14 SG, 1-EOP-TRIP-1 REACTOR TRIP OR SAFETY INJECTION

Answer	d	**Exam Level**	S	**Cognitive Level**	Comprehension	**Facility:** Salem 1 & 2	**Exam Date:** 12/11/2006
Tier:	Plant Systems	**RO Group**	1	**SRO Group**	1		059000A212
059	Main Feedwater System					**Record Number**	18

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

A2.12 Failure of feedwater regulating valves | 3.1* | 3.4*

Explanation of Answer

55.43(5) B is incorrect because PZR pressure will not be affected when a BF19 fails closed as long as heat transfer in SG is not affected. In actuality, as LESS cold feedwater enters 11 SG, the heat transfer rate will lower, resulting in less heat being transferred from 11 RCS loop. Temp will rise in that loop, which would cause PZR pressure to rise. Heaters will turn off, not on. C is incorrect because SGFP speed control is based on average steam flow, and is lag compensated. Steam flow should not change much, if any. The feed pressure to steam pressure D/P is compared to the D/P reference signal developed from avg. steam flow. With a BF19 failing closed, feed header pressure will rise, and if the size of the feed header pressure rise is large enough, it would act to LOWER SGFP speed to lower feed header pressure, to maintain the proper D/P. A is incorrect because the unaffected SG BF19's will not change position, they are loop controlled, and their parameters will not have changed.

Reference Title

Reactor Trip or Safety Injection

MAIN FEEDWATER / CONDENSATE SYSTEM ABNORMALITY

Learning Objectives

CN&FDWE008 Identify and describe the Control Room controls, indications, and alarms associated with the Condensate and Feedwater System, including:
a) The Control Room location of Condensate and Feedwater System control bezels and indications
b) The function of each Condensate and Feedwater System Control Room control and indication
c) The effect each Condensate and Feedwater System control has upon Condensate and Feedwater System components and operation
d) The plant conditions or permissives required for Condensate and Feedwater System Control Room controls to perform their intended function
e) The setpoints associated with the Condensate and Feedwater System control room alarms

CN&FDWE009 State the setpoints for the following automatic actuations associated with the Condensate and Feedwater System:
a) Condenser Hotwell Makeup and Rejection
b) Condensate Pump Low Flow Recirculation
c) Steam Generator Feedwater Pump Recirculation (N/A NEO)
d) Feedwater Isolation and Feedwater Interlock (N/A NEO)

Material Required for Examination			
Question Source:	Facility Exam Bank	Question Modification Method:	Editorially Modified
Question Source Comments:	Added procedure flow path to ensure 55.43(5) was applicable		

Which ONE of the following is correct concerning S2.OP-AB.CR-0001, CONTROL ROOM EVACUATION?

a. It only supports actions to shutdown the Rx and maintain the plant in HSB, and does NOT support any required cooldown/depressurization for ANY type of accident.

b. Supports shutting down the Reactor AND any required cooldown/depressurization for ANY type of accident.

c. Supports shutting down the Reactor AND any required cooldown/depressurization for ANY type of accident EXCEPT loss of coolant accidents.

d. Supports shutting down the Reactor AND any required cooldown/depressurization for ALL accidents EXCEPT those requiring entry into SAMGs.

Answer	a	Exam Level	S	Cognitive Level	Memory	Facility:	Salem 1 & 2	Exam Date:	12/11/2006

Tier:	Generic Knowledge and Abilities	RO Group	1	SRO Group	1	194001G106
GENERIC					Record Number	19

2.1	Conduct of Operations		
2.1.6	Ability to supervise and assume a management role during plant transients and upset conditions.	2.1	4.3

Explanation of Answer: 55.43(5) Section 2.0 Immediate Action NOTE: The EOPs are not applicable during Control Room Evacuation. EOPs should be used for information only or as directed by the TSC while performing this procedure.

Reference Title
Control Room Evacuation

Learning Objectives	
ABCR01E002	Describe, in general terms, the actions taken in S2.OP-AB.CR-0001(Q) and the bases for the actions.

Material Required for Examination			
Question Source:	Other Facility	Question Modification Method:	Editorially Modified
Question Source Comments:	Browns Ferry Unit 2, 9/17/2001 nrc Exam, modified to Salem terminology.		

Given the following conditions:

- Unit 2 is operating normally at 100% power when 21 SGFP trips.
- The Main Turbine runs back to 60% as expected.
- All systems respond as expected to the runback.
- 2 minutes after the runback, the PO announces that condenser backpressure is 2.6"Hg and rising at 1.0" every 1 minute.
- The CRS directs entry into AB-LOAD, and commences a 1%/minute load reduction, then directs the unloading rate raised to 3%/min when vacuum continues to degrade.
- With the reactor at 52% power, the Secondary NEO reports that there is a 2" diameter hole in the SGFP exhaust line to 21 condenser, he can hear a loud whistling noise around the hole.

Which actions and procedures which should be performed?

a. TRIP the reactor, GO TO TRIP-1.

b. PLACE rods in manual, TRIP the Main Turbine, GO TO AB-TRB.

c. Continue the load reduction in AB-LOAD until <49% power then TRIP the Main Turbine.

d. STOP heater drain pumps IAW IOP-4 to reduce rate of vacuum degradation, then TRIP the Main Turbine IAW AB-TRB-1.

Answer	a	**Exam Level**	S	**Cognitive Level**	Application	**Facility:**	Salem 1 & 2	**Exam Date:**	12/11/2006
Tier:	Generic Knowledge and Abilities		**RO Group**	1	**SRO Group**	1			194001G107
GENERIC								**Record Number**	20

2.1 Conduct of Operations

2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation. 3.7 4.4

Explanation of Answer: 55.43(5) >P-9 That makes distracter B incorrect. Continuing the load reduction would be non-conservative since the loss of vacuum is external to the turbine load reduction. Stopping HDPs IAW IOP-4 is using wrong procedure.

Reference Title

Rapid Load Reduction

Loss of Condenser Vacuum

Learning Objectives

ABLOADE003	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.

Material Required for Examination

Question Source: Facility Exam Bank | **Question Modification Method:** Direct From Source

Question Source Comments:

Which of the following condition(s) would REQUIRE Field Engineering to review a Troubleshooting Plan developed in accordance with SH.OP-AP.ZZ-0008, OPERATIONS TROUBLESHOOTING AND EVOLUTIONS PLAN DEVELOPMENT:

I. Equipment is NOT removed from service or tagged and presents a risk of tripping the plant either directly or as a result of causing a major plant transient. (Very High Risk)

II. Equipment is NOT removed from service or tagged. Could result in an unexpected load reduction, a plant transient, or a reportable event. Should NOT result in a reactor, turbine, or generator trip. (High Risk)

III. Equipment is NOT removed from service or tagged. Could have an effect on plant equipment but shall NOT present a risk of causing an unexpected load reduction, plant transient or reportable event. (Medium Risk)

IV. Equipment is removed from service or tagged such that troubleshooting or testing activities shall NOT adversely affect the operation or safety of the plant. (Low Risk)

a. I only.

b. I and II only.

c. I, II, and III only.

d. I, II, III and IV.

Answer	b	Exam Level	S	Cognitive Level	Memory	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Generic Knowledge and Abilities	RO Group	1	SRO Group	1				194001G220
GENERIC								Record Number	21
2.2	Equipment Control								
2.2.20	Knowledge of the process for managing troubleshooting activities.							2.2	3.3

Explanation of Answer: 55.43(5) A is incorrect because both High Risk and VERY High Risk must be evaluated. C is incorrect because Medium Risk does NOT need to be evaluated. D is incorrect because Medium Risk and Low Risk do NOT need to be evaluated.

Reference Title

OPERATIONS TROUBLESHOOTING AND EVOLUTIONS PLAN DEVELOPMENT

Learning Objectives

PROCEDE002 — Given a list of purposes, select the purpose of the following types of procedures in accordance with NC.NA-AP.ZZ-0001(Q), Nuclear Procedure System: SELECT the purpose of the following types of procedures in accordance with NC.NA-AP.ZZ-0001(Q), Nuclear Procedure System:

a. Abnormal Operating Procedures
b. Administrative Procedures
c. Alarm Response Procedures
e. Emergency Operating Procedures
f. Integrated Operating Procedures
g. In-service Test Procedures
h. Operating Procedures
i. MMIS Work Standards

Material Required for Examination

Question Source: Other Facility | **Question Modification Method:** Direct From Source

Question Source Comments:	Hope Creek NRC Exam 11/1/2005

During hydrostatic testing of the RCS in Mode 5, RCS pressure is rose to 2770 psig.

Which ONE of the following describes the MAXIMUM time allowed in accordance with Technical Specifications to reduce pressure below the Safety Limit?

a. 5 minutes.

b. 15 minutes.

c. 30 minutes.

d. 1 hour.

Answer	a	Exam Level	S	Cognitive Level	Memory	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Generic Knowledge and Abilities	RO Group	1	SRO Group	1				194001G222
GENERIC								Record Number	22

2.2	Equipment Control		
2.2.22	Knowledge of limiting conditions for operations and safety limits.	3.4	4.1

Explanation of Answer	55.43(2) A is correct per TS when in MODE 3-5. B is incorrect and is action time for several < 1 hour TS's. C is incorrect and only provides symmetry of choices. D is the action time if in MODES 1-2.

Reference Title
Salem Tech Specs

Learning Objectives	
TECHSPE006	Explain the term Safety Limit as it applies to the Technical Specifications, and describe the Safety Limits for Salem Nuclear Generating Station

Material Required for Examination			
Question Source:	Other Facility	Question Modification Method:	Direct From Source
Question Source Comments:	5/12/2005 Beaver Valley NRC Exam		

While reviewing a release permit for a 21 Waste Monitor Tank, it is determined that 2R18 Waste Disposal Liquid Rad Monitor has failed its source check. IAW S2.OP-SO.WL-0001, Release of Radioactive Liquid Waste, what are the required actions, if any, of the Control Room Supervisor ?

a. Do not approve the liquid waste discharge until the Unit 1 R18 can be source checked and aligned to monitor the release.

b. Do not approve the liquid waste discharge, secure the lineup, the liquid waste discharge is not permitted until 2R18 is repaired.

c. Approve the liquid waste discharge and ensure that continuous effluent sampling is conducted throughout the liquid waste discharge.

d. Approve the liquid waste discharge as long as a second sample was drawn, analyzed, and calculations were second verified prior to the release.

Answer	d	Exam Level	S	Cognitive Level	Memory	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Generic Knowledge and Abilities		RO Group	1	SRO Group	1			194001G306
GENERIC								Record Number	23
2.3	Radiation Control								
2.3.6	Knowledge of the requirements for reviewing and approving release permits.							2.1	3.1

Explanation of Answer: 55.43(4) A is incorrect because Unit 1 monitor cannot be aligned to unit 2 discharges. C is correct because the procedure allows for the release after double sample and double calc have been performed. B is incorrect because of C. D is incorrect because there is no provision to do manual effluent sampling.

Reference Title

Release of Radioactive Liquid Waste

Learning Objectives

WASLIQE012	Describe the procedures which govern the operation of the Radioactive Liquid Waste System, including significant prerequisites and precautions associated with each operating procedure which are required to be considered by either Licensed or Non-Licensed Operators

Material Required for Examination:

Question Source: Other Facility

Question Modification Method: Editorially Modified

Question Source Comments: Modified one distracter since the original question had 3 "approve the release" and only 1 "do not approve"

Given the following conditions:

- Unit 2 is operating at 100% power.
- Operators receive OHA G-7, ADFCS SWITCH TO MANUAL.
- The board operator notes both SGFPs Speed Controllers have switched to MANUAL.
- All BF19 and BF40 valves remain in AUTO.
- 21 SGFP speed is lowering slowly, and remains latched.
- All SG NR levels are 40% and dropping slowly

Which of the following describes the procedure which would be most effective in responding to these indications?

a. S2.OP-AR.ZZ-0007, OVERHEAD ANNUCIATORS WINDOW G to address the SGFP switch to MANUAL.

b. 2-EOP-TRIP-1, REACTOR TRIP OR SAFETY INJECTION, to respond to the Rx trip caused by SG lo-lo level.

c. S2.OP-AB.CN-0001, MAIN FEEDWATER / CONDENSATE SYSTEM ABNORMALITY, to address the imminent loss of 21 SGFP.

d. S2.OP-AR.ZZ-0012, CONTROL CONSOLE 2CC2, to respond to the PROGRAM DEVIATION SETPOINT ACTUAL alarms on all SGs.

Answer	c	Exam Level	S	Cognitive Level	Application	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Generic Knowledge and Abilities	RO Group	1	SRO Group	1				194001G410
GENERIC								Record Number	24

2.4	Emergency Procedures / Plan		
2.4.10	Knowledge of annunciator response procedures.	3.0	3.1

Explanation of Answer

55.43(5) A incorrect because the ARP would not address the lowering level, and there are no automatic actions besides the swapping to manual of controllers. B is incorrect because the CAS step 4.0 to take manual control of affected controller in AB.CN should preclude having a Rx trip. C is correct because the immediate action contained in AB.CN would trip the malfunctioning SGFP and cause an automatic turbine runback which would allow SG levels to recover, and give operators time to assess the ADFCS failure. D is incorrect because the control console alarm response would only take time away from entering the CN AB.

Reference Title

OVERHEAD ANNUCIATORS WINDOW G

MAIN FEEDWATER / CONDENSATE SYSTEM ABNORMALITY

USE OF PROCEDURES

Learning Objectives

PROCEDE002

Given a list of purposes, select the purpose of the following types of procedures in accordance with NC.NA-AP.ZZ-0001(Q), Nuclear Procedure System: SELECT the purpose of the following types of procedures in accordance with NC.NA-AP.ZZ-0001(Q), Nuclear Procedure System:
a. Abnormal Operating Procedures
b. Administrative Procedures
c. Alarm Response Procedures
e. Emergency Operating Procedures
f. Integrated Operating Procedures
g. In-service Test Procedures
h. Operating Procedures
i. MMIS Work Standards

ABCN01E004

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

Given the following conditions:

- Salem 1 and 2 are operating at 100% power.
- Hope Creek is operating at 100% power.
- Fire Brigade manning consists of 6 qualified personnel, which includes one Fire Brigade Leader.
- A Fire Brigade member falls ill, and is transported off-site by Medical Department personnel.

Which of the following describes the status of the Fire Brigade, and action(s), if any, which are required to be performed IAW NC.FP-AP.ZZ-0001, Fire Protection Organization, Duties, and Staffing?

a. The Fire Brigade remains adequately staffed. Only five members are required IAW Salem FSAR. No compensatory measures are required.

b. The Fire Brigade remains adequately staffed. The assumption is made that concurrent fires at Salem and Hope Creek are not plausible events. NRC notification of reduced staffing level is required within 2 hours.

c. The Fire Brigade staffing is inadequate. Initiate call-out of qualified personnel to ensure manning is restore to six members within 2 hours, otherwise submit a 24 hour report to the NRC.

d. The Fire Brigade staffing is inadequate. Initiate call-out of qualified personnel to ensure manning is restore to six members within 2 hours, otherwise initiate an Action Request and review for licensing commitment violation.

Answer	a	Exam Level	S	Cognitive Level	Application	Facility:	Salem 1 & 2	Exam Date:	12/11/2006
Tier:	Generic Knowledge and Abilities		RO Group	1	SRO Group	1			194001G426
GENERIC							Record Number		25

2.4	Emergency Procedures / Plan		
2.4.26	Knowledge of facility protection requirements including fire brigade and portable fire fighting equipment usage.	2.9	3.3

Explanation of Answer: 55.43(1) A is correct. Per the procedure and the FSAR, 5 fire brigade members are required. No compensatory actions are required by procedure. B is incorrect because there is no assumption made regarding 2 fires at once. C is incorrect because only 5 members are required, and there is not an E-plan NRC notification required. D is incorrect because staffing is adequate. Action is correct.

Reference Title

Fire Protection Organization, Duties, and Staffing?

Salem FSAR

Learning Objectives

CONDOPE006	Describe the minimum manning Salem Shift Complement for Modes 1-6 in accordance with NC.NA-AP.ZZ-0005(Q), Station Operating Practices

Material Required for Examination:

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

Question Source Comments: