US Nuclear Regulatory Commission
Diablo Canyon
L081 SRO Written Examination

Applicant Information	
Name: **ANSWER KEY**	
Date: 01/08/2010	Facility/Unit: Diablo Canyon Power Plant
Region: I ☐ II ☐ III ☐ IV ■	Reactor Type: W ■
Start Time:	Finish Time:

Instructions

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. To pass the examination you must achieve a final grade of at least 80.00 percent overall, with 70.00 percent or better on the SRO-only items. You have 8 hours to complete the combined examination.

Applicant Certification

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

____
Applicant's Signature

Results		
RO/SRO-Only/Total Examination Values	75 / 25 / 100	Points
Applicant's Scores	____ / ____ / ____	Points
Applicant's Grade	____ / ____ / ____	Percent

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Results

Examination Value	75 Points
Applicant's Score	________ Points
Applicant's Grade	________ Percent

Multiple Choice (Circle or X your choice) NAME: **ANSWER KEY**

If you change your answer, write your selection in the blank and initial.

001	A	●	C	D	______	026	A	B	C	●	______
002	A	B	●	D	______	027	A	B	●	D	______
003	●	B	C	D	______	028	●	B	C	D	______
004	A	●	C	D	______	029	●	B	C	D	______
005	A	●	C	D	______	030	A	B	●	D	______
006	●	B	C	D	______	031	A	B	●	D	______
007	●	B	C	D	______	032	A	B	C	●	______
008	A	B	●	D	______	033	A	B	C	●	______
009	A	B	●	D	______	034	A	●	C	D	______
010	●	B	C	D	______	035	●	B	C	D	______
011	A	●	C	D	______	036	A	●	C	D	______
012	A	B	C	●	______	037	A	●	C	D	______
013	●	B	C	D	______	038	A	B	C	●	______
014	A	●	C	D	______	039	A	B	C	●	______
015	A	B	C	●	______	040	A	●	C	D	______
016	A	B	●	D	______	041	●	B	C	D	______
017	●	B	C	D	______	042	A	B	●	D	______
018	A	B	●	D	______	043	●	B	C	D	______
019	A	●	C	D	______	044	A	B	●	D	______
020	A	●	C	D	______	045	●	B	C	D	______
021	A	B	●	D	______	046	A	B	C	●	______
022	A	B	●	D	______	047	A	B	C	●	______
023	●	B	C	D	______	048	A	B	C	●	______
024	A	B	●	D	______	049	A	●	C	D	______
025	●	B	C	D	______	050	A	B	●	D	______

L081 NRC Exam – 1/08/2010

Multiple Choice (Circle or X your choice) NAME: **ANSWER KEY**

If you change your answer, write your selection in the blank and initial.

051	●	B	C	D	_____	076	A	B	C	●	_____
052	A	B	●	D	_____	077	A	●	C	D	_____
053	A	B	C	●	_____	078	A	●	C	D	_____
054	●	B	C	D	_____	079	A	B	●	D	_____
055	A	B	●	D	_____	080	A	●	C	D	_____
056	A	●	C	D	_____	081	A	B	●	D	_____
057	A	●	C	D	_____	082	●	B	C	D	_____
058	A	●	C	D	_____	083	A	B	●	D	_____
059	A	●	C	D	_____	084	A	B	C	●	_____
060	A	B	C	●	_____	085	A	●	C	D	_____
061	A	B	C	●	_____	086	●	B	C	D	_____
062	A	B	●	D	_____	087	●	B	C	D	_____
063	A	B	●	D	_____	088	A	B	●	D	_____
064	A	●	C	D	_____	089	A	B	●	D	_____
065	A	B	C	●	_____	090	●	B	C	D	_____
066	A	B	●	D	_____	091	A	B	C	●	_____
067	A	B	C	●	_____	092	A	●	C	D	_____
068	●	B	C	D	_____	093	●	B	C	D	_____
069	A	B	C	●	_____	094	A	B	●	D	_____
070	●	B	C	D	_____	095	A	B	●	D	_____
071	A	B	C	●	_____	096	●	B	C	D	_____
072	●	B	C	D	_____	097	●	B	C	D	_____
073	A	●	C	D	_____	098	A	B	●	D	_____
074	●	B	C	D	_____	099	A	●	C	D	_____
075	A	B	C	●	_____	100	●	B	C	D	_____

Examination Outline Cross-Reference	**Level**	RO
Knowledge of the physical connections and/or cause-effect relationships between the RCPS (reactor Coolant Pump System) and the following: CVCS	**Tier #**	2
	Group #	1
	K/A #	003 K1.04
	Rating	2.6

Question 1

Which of the following are symptoms that the #1 seal for RCP 1-4 has failed?

A Lower #1 seal leakoff; higher VCT pressure

B Higher #1 seal leakoff; higher VCT pressure

C Lower #1 seal leakoff; lower VCT pressure

D Higher #1 seal leakoff; lower VCT pressure

Proposed Answer: B

Explanation:

A. Incorrect – Seal leakoff flow would be high

B. Correct – No 1 seal failure cause high flow to VCT increasing pressure. PK 05-01 sends the operator to section B for number 1 seal failure if leakoff flow is high. Per AP-28, the following are listed as plant indications of a number 1 seal failure: High VCT Pressure

C. Incorrect – Both seal leakoff flow and VCT pressure would increase

D. Incorrect – VCT pressure Low is inconsistent with excessive No 1 seal leak off going into the VCT

Technical References: A-6, Chemical and Volume Control System, Section 2.2
OIM 6-1, Rev. 28.
AR PK05-02, RCP No. 11.
OP AP-28, Section B, RCP No.1 Seal Failure, section 2, Plant Indications
References to be provided to applicants during exam: None
Learning Objective: 35754 - Describe system interrelationships between the CVCS and other plant systems

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X

Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41 (2 to 9)	

Examination Outline Cross-Reference	**Level**	RO
Knowledge of bus power supplies to the following: Charging Pumps.	**Tier #**	2
	Group #	1
	K/A #	004 K 2.03
	Rating	3.3

Question 2

GIVEN:

- Unit 1 is at 100% power
- DG 1-2 is cleared for maintenance
- ECCS CCP 1-1 is off
- ECCS CCP 1-2 is running
- Normal CCP 1-3 is off

If 25 kV to Auxiliary Transformer 1-2 is lost, ECCS CCP 1-2 will ___________.

A. trip. ECCS CCP 1-1 will auto start, powered from its emergency diesel generator

B. trip. ECCS CCP 1-1 will auto start, powered from Startup Transformer 1-2.

C. remain running, powered from Startup Transformer 1-2. ECCS CCP 1-1 remains off.

D. remain running, powered from Startup Transformer 1-2. ECCS CCP 1-1 will auto start being powered from Startup Transformer 1-2.

Answer: C

Explanation:

A. Incorrect – Automatic transfer to Startup Transformer keeps running pump powered.

B. Incorrect – Automatic transfer to Startup Transformer running pump powered

C. Correct – Automatic transfer to Startup Transformer running pump powered and no auto start signal for standby pump is generated

D. Incorrect –No auto start signal for standby pump is generated.

Technical References: B-1A, Chemical and Volume Control System, Page 2.2-6
OIM Page J-1-1, J-5-1c Rev. 28

References to be provided to applicants during exam: None

Learning Objective: 4295 - Analyze automatic features and interlocks associated with the Electric Power Transfer System

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41 (7)	

Examination Outline Cross-Reference	**Level**	RO
Knowledge of the effect that a loss or malfunction of the RHRS will have on the following: RCS	**Tier #**	2
	Group #	1
	K/A #	005 K 3.01
	Rating	3.9

Question 3

GIVEN:

- The Residual Heat Removal (RHR) System has been placed in service
- 40% Steam dumps are open in MANUAL and STEAM PRESSURE mode
- CCW is aligned to both RHR heat exchangers
- RHR heat exchanger bypass valve, HCV-670 is 30% open to establish a 20°F/hour RCS cooldown rate

Which of the following events would <u>reduce</u> the RCS cooldown rate?

A. Instrument air is lost to RHR heat exchanger bypass valve, HCV-670.

B. Instrument air is lost to RHR flow control valve, HCV-637.

C. Flow input to RHR pump recirculation valve FCV-641A, fails high.

D. Main Steam header pressure transmitter, PT-507, fails low.

Answer: A

Explanation:

A. Correct – HCV-670 fails open. Opening the bypass valve will divert coolant away from the HX's resulting is less cooling

B. Incorrect – HCV-637 fails open. This increases RHR flow and forces more flow through the HX's and increase the cooldown rate

C. Incorrect – FCV-641A closes if flow is high(>1398 gpm)

D. Incorrect – in Manual, steam dumps will not respond to a failure of PT-507

Technical References: OP B-2:V Rev. 25, section 6; LB2-RHRS, Rev. 13, page 6 of 40; OP AP-9, Loss of IA, Rev. 23, page 17 (for failure condition of HCV-637 and 670)

References to be provided to applicants during exam: None

Learning Objective: 7050 - Discuss abnormal conditions associated with the RHR system

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	X
	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41 (7)	

Examination Outline Cross-Reference	Level	RO
Knowledge of the ECCS design feature(s) and/or interlocks which provide for the following: Reset of containment isolation.	**Tier #**	2
	Group #	1
	K/A #	006 K4.13
	Rating	3.8

Question 4

Which of the following describes the minimum actions required to reset both trains of Phase A?

A. Either Phase A reset pushbutton taken to RESET.

B. Both Phase A reset pushbuttons taken to RESET.

C. Both Trains of SI reset and either Phase A reset switch taken to RESET.

D. Both Trains of P-4 actuated, both Trains of SI reset and both Phase A reset switches taken to RESET.

Answer: B
Explanation:

A. Incorrect – Requires both trains

B. Correct – Both trains required, however, no other action is required.

C. Incorrect – SI reset is not required.

D. Incorrect –. P-4 and SI not part of the reset circuitry.

Technical References: B-6A, Reactor Protection System Page 2.2-18
OIM Pages B-6-5 and B-6-5-7
References to be provided to applicants during exam: None
Learning Objective: 37048 - Analyze automatic features and interlocks associated with the RPS

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41 (7)	

Examination Outline Cross-Reference	**Level**	RO
Knowledge of PRTS design feature(s) and/or interlock(s) which provide for the following: Quench Tank Cooling	**Tier #**	2
	Group #	1
	K/A #	007 K4.01
	Rating	2.6

Question 5

GIVEN:

- A leaking Pressurizer PORV has been isolated by the crew
- PK05-25, PRT PRESS LVL/TEMP is in alarm due to high PRT temperature (140°F)
- PRT level (62%) and pressure (7 psig) are slowly rising

In accordance with AR PK05-25, PRT Press/Lvl/Temp, PRT temperature is lowered by _________.

A. Venting the PRT to the Vent Header.

B. Filling the PRT using primary water, then draining, if required, to the RCDT.

C. Draining the PRT to the Containment Structure Sump, refilling with primary water to the high level alarm, and draining again.

D. Verifying RCS-1-PCV-472, PRT Vent to Vent Header, closes.

Answer: B

Explanation:

A. Incorrect – Plausible. The temperature is lowered by adding primary water to the PRT. Isolating the leaking PORV would prevent a further increase in the temperature.
B. Correct – In accordance with the AR PK step 2.1.6, primary water introduction is used to cool the quench volume, once quenched, the PRT is drained if level is high.
C. Incorrect – The volume is cooled first with primary water and then drained, not vice versa.
D. This is an automatic action that occurs at 10 psig and has no effect on cooling the PRT.

Technical References: AR PK05-25
References to be provided to applicants during exam: None
Learning Objective: 40576 - Discuss abnormal conditions associated with the PRT system

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X

Question History:	Last NRC Exam	NO
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41 (7)	

Examination Outline Cross-Reference	Level	RO
Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CCWS controls including: CCW flow rate.	**Tier #**	2
	Group #	1
	K/A #	008 A1.01
	Rating	2.8

Question 6

GIVEN:

- Unit 1 is at 100% steady-state power
- TCV-130, Letdown Heat Exchanger CCW Temperature Control valve is in Manual

Which of the following is a potential consequence of opening TCV-130 and causing a significant change in letdown temperature?

A. A positive reactivity addition

B. Flashing of letdown

C. Automatic isolation of letdown to the demineralizers

D. A negative reactivity addition

Answer: A

Explanation:

A. Correct – Cooler water tends to deposit boron atoms in the demineralizers especially at BOL (high Cb). This could cause power to go above the license limit

B. Incorrect – Letdown flashing occurs at the outlet of the Regen heat exchanger as a result of letdown flow much higher than charging flow.

C. Incorrect – demin bypass occurs at high temperature, this would not be the case if TCV-130 was opened (temperature would decrease).

D. Incorrect – Lower temperature causes positive, not negative reactivity addition, higher temperature could cause negative reactivity.

Technical References: OP B-1ALXII, precautions and limitations, step 6.1. OP1.ID3 step 5.2.4 rev 5
References to be provided to applicants during exam: None
Learning Objective: 5346 - Discuss significant precautions and limitations associated with the ASW system

Question Source:	Bank #	X
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41 (7)	

Examination Outline Cross-Reference	**Level**	RO
Knowledge of the effect of a loss or malfunction of the following will have on the PZR PCS: PZR Sprays and Heaters.	**Tier #**	2
	Group #	1
	K/A #	010 K6.03
	Rating	3.2

Question 7

GIVEN:

- Unit 1 is at 100% power
- Two of the backup heater groups were manually placed in the "ON" position one hour ago for RCS mixing

Spray valve PCV-455A fails full open.

Which of the following sequences of events will occur?

A. PCV-455B spray valve closes, pressurizer heaters energize, pressure lowers below PORV low pressure interlock setpoint, Reactor trips.

B. PCV-455B spray valve closes, pressure lowers below PORV low pressure interlock setpoint, pressurizer heaters energize, Reactor trips.

C. No change in position of the spray valve PCV-455B, pressurizer heaters energize, Reactor trips, pressure lowers below PORV low pressure interlock setpoint.

D. No change in position of the spray valve PCV-455B, pressurizer heaters energize, pressure lowers below PORV low pressure interlock setpoint, Reactor trips.

Answer: A

Explanation:

A. Correct – Both spray valves should initially be open. Due to the integral nature of the PZR pressure master controller, a short time after the backup heater group was energized, the spray valves should have opened and will be the first pressure control component to respond (valve close) as PZR pressure decreases.

B. Incorrect – heaters energize before the PORV interlock is reached

C. Incorrect – the spray valve functioning spray valve is not addressed

D. Correct – the spray valve functioning spray valve is not addressed

Technical References: STG A4A, Rev. 14, page 22-8, OIM A-4-6, Rev. 26

References to be provided to applicants during exam: None
Learning Objective: 36926 - Discuss abnormal conditions associated with the Pzr, Pzr Pressure and Level Control System

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	X
	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41 (7)	

Examination Outline Cross-Reference	**Level**	RO
Knowledge of the operational impact of the following concepts as they apply to the PZR PCS: Determination of condition of fluid in PZR, using steam tables.	**Tier #**	2
	Group #	1
	K/A #	010 K5.01
	Rating	3.5

Question 8

GIVEN:

- Unit 1 is in MODE 5
- RHR is in service with RCS loop temperatures at 95°F
- Pressurizer level is 70% cold cal
- Pressurizer pressure is 3.5 psia
- Vacuum refill skid pump is shut down
- Pressurizer heaters are energized

At what pressurizer liquid temperature will bubble formation start in the pressurizer?

A. 140°F

B. 144°F

C. 148°F

D. 152°F

Answer: C

Explanation:

A. Incorrect –

B. Incorrect –

C. Correct – Tsat for 3.5 psia is 148°F

D. Incorrect.

Technical References: steam tables
References to be provided to applicants during exam: Steam Tables
Learning Objective: 40738 - Apply fundamentals topics associated with the Pzr, Pzr Pressure and Level Control System

Question Source:	Bank #	X
(note changes; attach parent)	Modified Bank #	

	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41 (5)	

Examination Outline Cross-Reference	**Level**	RO
Ability to (a) predict the impacts of the following malfunctions or operations on the RPS; and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations: Failure of RPS to trip the reactor.	**Tier #**	1
	Group #	1
	K/A #	012 A2.06
	Rating	4.4

Question 9

GIVEN:

- Unit 2 was at 100% power
- A Spurious Safety Injection occurs
- The reactor fails to trip

Which of the following describes the response of the Reactor Protection System (RPS) and the Function Restoration Procedure that should be used to mitigate the consequences?

A. The main turbine will not be automatically tripped. FR-S.1 is used to insert negative reactivity.

B. The main turbine will not be automatically tripped. FR-C.1 is used to establish effective core cooling.

C. SI will automatically trip the main turbine. FR-S.1 is used to insert negative reactivity.

D. SI will automatically trip the main turbine. FR-C.1 is used to establish effective core cooling.

Answer: C

Explanation:

A. Incorrect – The turbine is tripped by SI, but P-4 is not generated. FR-S.1 is used.

B. Incorrect – the turbine is tripped, and FR-C.1 is not used.

C. Correct – ECCS systems are designed to remove decay heat. FR-S.1 is the highest priority RED Path. The turbine is tripped by the SI signal, but P-4 is not generated.

D. Incorrect –The turbine is tripped, and P-4 is not generated. FR-C.1 is not used.

Technical References: OPS-Lesson Plan LMCDFRS Obj 4
OIM C-3-5
References to be provided to applicants during exam: None
Learning Objective: 37048 - Analyze automatic features and interlocks associated with the

RPS
9704 - Identify entry conditions for the FRPs

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41 (5)	

Examination Outline Cross-Reference	**Level**	RO
Knowledge of the effect that a loss or malfunction of the ESFAS will have on the following: Fuel.	**Tier #**	2
	Group #	1
	K/A #	013 K3.01
	Rating	4.4

Question 10

GIVEN:

- Unit 1 was at 100% power
- A reactor trip occurs due to a loss of all feedwater
- Auxiliary Feedwater (AFW) fails to actuate

Which of the following is the likely consequence of the AFW failure if no action is taken?

A. Core uncovery and overheating.

B. RCS pressure exceeding design pressure.

C. Return to criticality.

D. Multiple Steam Generator U-tube failures.

Answer: A

Explanation:

A. Correct – From FR-H.1 background, without operator action, once the secondary is lost as a heat sink, the RCS will heat up to saturation, boil off and result in core uncovery and damage.

B. Incorrect – Plausible if assuming the temperature increase would cause a large enough pressure increase to threaten RCS integrity

C. Incorrect – Core boil off is correct.

D. Incorrect – Steam generator dryout is possible

Technical References: FR-H.1, background
References to be provided to applicants during exam: None
Learning Objective: 11319 - Describe the loss of feedwater event leading to core damage.

Question Source:	Bank #	X
(note changes; attach parent)	Modified Bank #	

	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41 (7)	

Examination Outline Cross-Reference	Level	RO
Ability to monitor automatic operation of the ESFAS including: Inputs and logic.	**Tier #**	2
	Group #	1
	K/A #	013 A3.01
	Rating	3.7

Question 11

GIVEN:

- A plant shutdown in accordance with OP L-5, Plant Shutdown From Minimum Load to Cold Shutdown, is in progress
- Unit 1 is at 7% reactor power
- The Main Turbine has been MANUALLY tripped as part of the normal shutdown

What will be the status of the following permissive alarms?

- PK08-02, Low Power Permissive P-7 (white)
- PK08-05, Power Range at Power Permissive P-10 (yellow)
- PK08-08, Turbine Low Power Permissive P-13 (white)

A. PK08-02 is clear, PK08-05 and PK08-08 are lit.

B. PK08-05 is clear, PK08-02 and PK08-08 are lit.

C. PK08-08 is clear, PK08-02 and PK08-05 are lit.

D. Both PK08-08 and PK08-02 are clear, PK08-05 is lit.

Answer: B
B is the correct answer
Explanation:
A. Incorrect – Below 10%, PK08-02 and PK08-08 are on (off at power).
B. Correct – 05 will be out.
C. Incorrect – 08 will be lit
D. Incorrect – 08 and 02 will be lit

Technical References OIM B-6-2
References to be provided to applicants during exam: None
Learning Objective: 37049 - Describe controls, indications, and alarms associated with the RPS

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No

Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41 (7)	

Examination Outline Cross-Reference	**Level**	RO
Ability to manually operate and/or monitor in the control room: Containment reading of temperature, pressure, and humidity system.	**Tier #**	2
	Group #	1
	K/A #	022 A4.05
	Rating	3.1

Question 12

Containment average temperature is reading abnormally high due to one of the selected containment area temperature elements failing high..

Containment average temperature can be read in the Control Room at ______; and the faulty temperature element can be removed from the average calculation ______.

A. TI-26 (Containment Area TM Panel); on the Plant Process Computer

B. YR-26 (Containment Average Temperature); on the Plant Process Computer

C. TI-26 (Containment Area TM Panel); at TI-26

D. YR-26 (Containment Average Temperature); at TI-26

Answer: D

Explanation:

A. Incorrect. Both parts incorrect.
B. Incorrect. First part is correct.
C. Incorrect. Second part is correct
D. Correct. YR-26 is located in the control room. The elements can be deselected locally at TI-26 which is outside the control room.

Technical References: LH-11
References to be provided to applicants during exam: None
Learning Objective: 37589 - Describe controls, indications, and alarms associated with the Containment Structure

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41 (7)	

Examination Outline Cross-Reference	**Level**	RO
Knowledge of CSS design features and/or interlocks which provide for the following: Iodine scavenging via the CSS	**Tier #**	2
	Group #	1
	K/A #	026 K4.06
	Rating	2.8

Question 13

Which of the following describes how the sufficient amount of Sodium Hydroxide (NaOH) is assumed to be delivered to the Containment Sump?

NOTE: CS-8994A and CS-8994B are Spray Additive Tank Outlet to Eductor valves

A. Either CS-8994A OR CS-8994B must open to provide sufficient flow of NAOH to the suction of the Containment Spray pumps. Proper NaOH volume is assured by running the Containment Spray pumps until Low-Low RWST level (4%).

B. Both CS-8994A AND CS-8994B must open to provide sufficient flow of NAOH to the suction of the Containment Spray pumps. Proper NaOH volume is assured by running the Containment Spray pumps until Low-Low RWST level (4%).

C. Either CS-8994A OR CS-8994B must open to provide sufficient flow of NAOH to the suction of the Containment Spray pumps. Proper NaOH volume is assured by running the Containment Spray pumps until Low RWST level (33%).

D. Both CS-8994A AND CS-8994B must open to provide sufficient flow of NAOH to the suction of the Containment Spray pumps. Proper NaOH volume is assured by running the Containment Spray pumps until Low RWST level (33%).

Proposed Answer: A

Explanation:

A Correct: Either train is 100% capacity. Pumps run until the Low-Low Level (4%) is reached.

B Incorrect: Either train is 100% capacity.

C Incorrect: Pumps continue to at RWST low level (33%) until the Low-Low Level (4%) is reached.

D Incorrect: Pumps are secured at RWST Low-Low (4%), either train is 100% capacity.

Technical References STG I-2 page 1-4 and 3-10.

References to be provided to applicants during exam: None

Learning Objective: 40802 - Explain significant CSS design features and the importance to nuclear safety

Question Source: (note changes; attach parent)	Bank #	
	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41 (7)	

Note, modified distractor D for balance, modified second part of answers to inidicate pumps run until an RWST level is reached (not tripped…) rnf5, 9/8/09

Examination Outline Cross-Reference	**Level**	RO
Ability to determine operability and/or availability of safety related equipment	**Tier #**	2
	Group #	1
	K/A #	026 G2.2.37
	Rating	3.6

Question 14

Which of the following is the required line up for a properly filled and OPERABLE Containment Spray System?

A. SI-246 (RWST Supply to Containment Spray) closed; CS-9001A and B (Spray Pump Discharge Valves) open with power removed.

B. SI-246 RWST Supply to Containment Spray) sealed open; CS-9001A and B (Spray Pump Discharge Valves) closed and ready to auto open.

C. SI-246 (RWST Supply to Containment Spray) closed; CS-9001A and B (Spray Pump Discharge Valves) closed and ready to auto open.

D. SI-246 (RWST Supply to Containment Spray) sealed open; CS-9001A and B (Spray Pump Discharge Valves) open with power removed.

Proposed Answer: B
Explanation:

A Incorrect: This prevents filling the discharge header downstream of 9001A/B, but system will not automatically perform it function.

B Correct: Discharge valve are closed to prevent filling the discharge header while valve SI-246 is sealed open.

C Incorrect: System will not automatically perform it function.

D Incorrect: The discharge will be filled.

Technical References: STG I-2 page 3-1
Procedure OP I-2:I "Containment Spray System -Make Available" .
References to be provided to applicants during exam: None
Learning Objective: 40805 - Describe the operation of the CSS

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No

Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41 (7)	

Examination Outline Cross-Reference	**Level**	RO
Knowledge of the purpose and function of major system components and controls	**Tier #**	2
	Group #	1
	K/A #	039 G2.1.28
	Rating	4.1

Question 15

What is the function of the Turbine Overspeed Protection Circuit (OPC)?

A. Closes the reheat stop valves and extraction non-return valves permanently to prevent turbine overspeed following a turbine trip.

B. Closes the reheat stop valves and extraction non-return valves momentarily to prevent turbine overspeed following a turbine trip.

C. Closes the governor valves and intercept valves permanently to prevent turbine overspeed following a complete loss of electrical load.

D. Closes the governor valves and intercept valves momentarily to prevent turbine overspeed following a complete loss of electrical load.

Proposed Answer: D

Explanation:

A Incorrect: These valves do close on a turbine trip but due to either low PCV-23 or SV-40.

B Incorrect: the OPC does not control these valves.

C Incorrect: Valve close momentarily.

D Correct: The OPC circuit momentarily closes the governor and intercept valve at 103% but allows reopening when speed is less than 101% following a complete loss of electrical load.

Technical References: STG C-3b page 2.1-28, C3C, page 2-44. OIM C-3-4
References to be provided to applicants during exam: None
Learning Objective: 41077 - Discuss abnormal conditions associated with the Turbine Control Oil System

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	

10CFR Part 55 Content: 55.41 (7)
Note: added overspeed to B and D, and OPC to question. Rnf5, 9/8/09

Examination Outline Cross-Reference	Level	RO
Knowledge of the physical connections and/or cause effect relationships between the MRSS and the following systems: S/G	**Tier #**	2
	Group #	1
	K/A #	039 K1.01
	Rating	3.1

Question 16

GIVEN:

- Unit 1 is at 100% power
- Reheat Steam to one Moisture Separator Reheater (MSR) is isolated
- All systems operating normally in automatic control

The SFM has directed the CO to lower Unit Load to 98% in preparation for restoring HP reheat steam.

The reason for this power reduction is to avoid which of the following as HP Reheat is restored?

A. Lifting the MSR relief valves.

B. Water hammer.

C. An overpower condition.

D. Excessive thermal stress.

Proposed Answer: C

Explanation:

A Incorrect: Pressure changes are minimized by the RAMP open feature (modulate open at 75º/hour rate)

B Incorrect: Procedure has been written to limit or prevent water hammer (discussion item 2.2

C Correct: Increasing Main Steam flow will drop S/G pressure and cause a RCS cooldown. With a negative MTC, power will increase.

D Incorrect: Thermal stresses are minimized by the RAMP open feature (modulate open at 75º/hour rate)

Technical References: STG C5 OP-C-5:II step 6.5.2 and precaution & limitation 5.8.1
References to be provided to applicants during exam: None
Learning Objective: 3409 - Discuss significant precautions and limitations associated with the

MSRs

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41 (2 to 9)	

Examination Outline Cross-Reference	**Level**	RO
Ability to manually operate and monitor in the control room: Recovery from automatic feedwater isolation	**Tier #**	2
	Group #	1
	K/A #	059 A4.11
	Rating	3.1

Question 17

The plant experienced a reactor trip and Safety Injection (SI) from 100% power.

What actions must be taken before the Main Feedwater Regulating and Bypass valves can be opened?

A. Reset SI signal, cycle the Reactor trip breakers, reset Feedwater Isolation.

B. Reset SI signal, cycle the Reactor trip breakers only.

C. Cycle the Reactor trip breakers, reset Feedwater Isolation only.

D. Reset Feedwater Isolation only.

Proposed Answer: A
Explanation:

A. Correct. The sealin for P-4 and SI must be removed, then FWI reset.

B. In correct. Feedwater Isolation must be reset.

C. Incorrect. SI must be reset.

D. Incorrect. Both the P-4 and SI seal-in must be removed.

Technical References OIM, Feedwater Isolation Signals, Page B-6-12, Rev. 28
References to be provided to applicants during exam: None
Learning Objective: 37048 - Analyze automatic features and interlocks associated with the RPS

Question Source:	Bank #	X
(note changes; attach parent)	Modified Bank #	A-0730
	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41 (7)	

Examination Outline Cross-Reference	Level	RO
Knowledge of the effect of a loss or malfunction of the following will have on AFW components: Pumps	**Tier #**	2
	Group #	1
	K/A #	061 K6.02
	Rating	2.6

Question 18

Unit 2 was at 100% power when a Reactor Trip occurred.

Motor Driven AFW Pump 2-3 immediately trips.

As a result of the pump trip, Steam Generator AFW Supply Valves, LCV-115 and LCV-113 will:

A. Close but then reopen to provide runout protection.

B. Open but then close as Turbine Driven AFW pump restores level.

C. Open and remain open due to the pump trip.

D. Throttle and remain throttled due to loss of flow input to the circuitry.

Proposed Answer: C
Explanation:

A. Incorrect. Upon the pump trip, the EH pump loses power, the solenoid relief valves fail open, and the loss of EH pressure causes the valves to fail open.

B. Incorrect. Valve will close as level rises if MDAFWP not tripped.

C. Correct. Removal of the pump breaker signal causes the valves to fail open.

D. Incorrect. The valves fail open.

Technical References Ops lesson LD-1, Page 29 of 39, Rev. 12
References to be provided to applicants during exam: None
Learning Objective: 37635- Describe controls, indications, and alarms associated with the Auxiliary Feedwater System.

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	X
	New	
Question History:	Last NRC Exam	No

Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41 (7)	

Examination Outline Cross-Reference	**Level**	RO
Ability to monitor automatic operation of the AFW including: AFW startup and flow.	**Tier #**	2
	Group #	1
	K/A #	061 A3.01
	Rating	

Question 19

GIVEN:

- A Unit 1 plant startup is in progress per OP L-3, Secondary Plant Startup
- The reactor is at 9% power with a Main Feedwater Pump in service

While rolling the Main Turbine, S/G level control malfunctions which results in S/G 1-2 level exceeding 90%. All other steam generator levels are on program.

Which of the following automatic AFW pump starts, if any, will occur?

A. None.

B. Both motor driven AFW pumps will immediately start.

C. All AFW pumps will immediately start.

D. All AFW pumps will start following a time delay.

Answer: B

Explanation:

A. Incorrect. Although steam generator levels are not low, the AFW pumps (MDAFW) will start due to the trip of the MFW pumps.
B. Correct. Steam Generator Level > 90% is P-14. This Trips both Main feed Pumps resulting in an immediate start signal to Motor Driven pumps only.
C. Incorrect. Only the Motor Driven AFW pumps start on Trip of Main Feed Pumps.
D. Incorrect. Only the Motor Driven AFW pumps start on Trip of Main Feed Pumps. A time delay is only associated with the low-low steam generator start signal below 50% power

Technical References OIM B-6-2, STG-D1 page 2-9 & 27
References to be provided to applicants during exam: None
Learning Objective: 37637 - Analyze automatic features and interlocks associated with the AFW system

Question Source: Bank # Requal A-0687
(note changes; attach parent) Modified Bank #

	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41 (7)	

Examination Outline Cross-Reference	Level	RO
Ability to (a) predict the impacts of the following malfunctions or operations on the ac distribution system; and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations: Consequences of exceeding current limitations.	**Tier #**	2
	Group #	1
	K/A #	062 A2.09
	Rating	2.7

Question 20

GIVEN:
- Unit 1 is at 100% power
- Unit 2 is in MODE 5 on Startup power
- D/G 2-2 is out of service for maintenance

A line fault occurs on 230kV Switchyard causing a loss of power to Startup Transformers 1-1 and 2-1.

What is the impact on each Unit's 4kV vital system and what procedure would be used to mitigate the consequences?

A. No effect on Unit 1.
Unit 2 Vital 4kV Bus G is de-energized causing entry into OP-SD-0, "Loss of, or Inadequate Decay heat Removal".

B. No effect on Unit 1.
Unit 2 Vital 4kV Bus H is de-energized causing entry into OP OP-SD-0, "Loss of, or Inadequate Decay heat Removal".

C. Unit 1 Vital 4kV buses are energized from their respective Diesel Generators.
Unit 2 Vital 4kV Bus H is de-energized causing entry into OP-SD-1, "Loss of AC Power".

D. Unit 1 Vital 4kV buses are energized from their respective Diesel Generators.
Unit 2 Vital 4kV Bus G is de-energized causing entry into OP-SD-1, "Loss of AC Power".

Proposed Answer: B

Explanation:

A. Incorrect. Normal power to Unit 1 4kV buses is the Auxiliary Transformer, but OP-SD-0 will direct operator to OP-SD-1 Loss of Power. Restoring power will restore decay heat removal.

B. Correct. Normal power to Unit 1 4kV buses is the Auxiliary Transformer. OP-SD-1 Loss of Power is highest priority. Restoring power will restore decay heat removal.

C. Incorrect. . Normal power to Unit 1 4kV buses is the Auxiliary Transformer. OP-SD-0 will direct operator to OP-SD-1 Loss of Power. Restoring power will restore decay heat removal.

D. Incorrect. . Normal power to Unit 1 4kV buses is the Auxiliary Transformer, but OP-SD-1 is correct.

Technical References: OIM, Electrical Distribution Overview, Page J-1-1, Rev. 28,
LER 2007-001-00
STG J-6A & 6B
References to be provided to applicants during exam: None
Learning Objective: 41081 - Discuss abnormal conditions associated with the 4KV System

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41 (5)	

Examination Outline Cross-Reference	**Level**	RO
Ability to predict and/or monitor change in parameters associated with operating the DC electrical system controls including: Battery capacity as it is affected by discharge rate	**Tier #**	2
	Group #	1
	K/A #	063 A1.01
	Rating	2.5

Question 21

A battery is supplying its Vital 125 VDC bus.

Which of the following would be an indication in the Control Room that a heavy load has been placed on a Vital 125 VDC battery?

A. No initial change in voltage but a decrease over time.

B. An initial drop in voltage with a slow return to normal.

C. An initial drop in voltage with a gradual decrease over time.

D. An initial increase in voltage with a gradual decrease over time.

Answer: C

Explanation:

A. Incorrect. the large load will initially cause voltage to drop.
B. Incorrect. Voltage will drop, if load is removed, voltage may recover, but not if it continues to be applied.
C. Correct. As a cell discharges, lead sulfate precipitates out of solution onto both positive and negative plates. This causes the plates to clog. As the plates clog, the effective surface area of the plates is rapidly reduced. Since the effective surface area of the plates determines the maximum reaction rate, it will be limited and the total battery voltage will begin to drop. On a loss of all AC power, when the battery has a high load, the voltage drops at a rapid rate immediately, then at a slower rate as time progresses. Voltage, then, is a fair estimate of a battery's condition at any one time under load, and the Control Room Operators may use voltage as a good approximation of battery capacity.
D. Incorrect. Voltage initially decreases

Technical References: STG-J9 and T/S 3.8.6 basis
References to be provided to applicants during exam: None
Learning Objective: 10975 - STATE what happens to the lead terminals and the sulfuric acid during battery use under load.
10976 - STATE how each of the following may be used as an indicator of battery condition during battery loading:
a. Specific gravity of the electrolyte
b. Battery voltage

c. Battery capacity

Question Source:	Bank #	X (F-2879)
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41 (5)	

Examination Outline Cross-Reference	**Level**	RO
Knowledge of the effect of a loss or malfunction of the following will have on the EDG system: Air Receivers	**Tier #**	2
	Group #	1
	K/A #	K6.07
	Rating	2.7

Question 22

GIVEN:

- Unit 1 is at 100%
- A relief valve has failed open on Diesel Generator (DG) 1-1 Starting Air Receiver 1-1A
- The leakage exceeds the capacity of the starting air compressor and the 1-1A Starting Air Receiver is depressurized

Which of the following describes the affect of this failure on DG 1-1, if a start signal occurs prior to any operator action?

A. Start in the normal allowed time with air from the 1-1B Starting Air Receiver via all four starting air solenoids.

B. Will start, however, starting time will exceed the surveillance required response time.

C. Start in the normal allowed time with air from the 1-1B Starting Air Receiver via the two starting air solenoids associated with the 1-1B Starting Air Receiver.

D. Will not start because the starting air system will be depressurized.

Proposed Answer: C

Explanation:

A. Incorrect. Diesel will start in normal time, but the start will be with only 2 of the 4 starting motors.

B. Incorrect. Diesel will start, but will not exceed the surveillance time as the starting air systems are redundant.

C. Correct. Start will occur in normal time via 2 starting air solenoids associated with the 1-1B starting air system

D. Incorrect. The EDG will start.

Technical References STG 6B, Section 5 and T/S 3.8.8 Bases
References to be provided to applicants during exam: None
Learning Objective: 37728 - Decribe Diesel Generator System components

Question Source: Bank # X

(note changes; attach parent)	Modified Bank # New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	 X
10CFR Part 55 Content:	55.41 (7)	

Examination Outline Cross-Reference	Level	RO
Knowledge of the operational implications as they apply to concepts as to the PM system: Radiation theory, including sources, types, units, and effects.	**Tier #**	2
	Group #	1
	K/A #	073K5.01
	Rating	2.5

Question 23

Unit 1 is at full power.

A small steam generator tube leak is causing steam line radiation monitor RM-73 to read 1000 cpm above background.

If the monitor is functioning properly, what should happen to the indication if power is reduced to 50%?

A. Indication should decrease due to the decrease in N-16 production.

B. Indication should decrease due to the decrease in iodine production.

C. Indication should remain the same due to the continued tube leakage.

D. Indication should increase because there is less steam flow but the same amount of radioactivity.

Answer A

Explanation:

A Correct.

B Incorrect.

C Incorrect

D Incorrect

Technical References: OIM G-3-1, STG G-4a page 2.3-59.
References to be provided to applicants during exam: None
Learning Objective: 37877 - Describe the operation of the Radiation Monitoring system.

Question Source:	Bank #	X
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	No

Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41 (7)	

Examination Outline Cross-Reference	**Level**	RO
Knowledge of SWS design feature(s) and/or interlock(s) which provide for the following: Service water train separation. ***Note: SWS is analogous to Auxiliary Salt Water***	**Tier #**	2
	Group #	1
	K/A #	076 K4.06
	Rating	2.8

Question 24

Which of the following is the normal alignment of Auxiliary Salt Water Pump Crosstie Valves, FCV-495 and FCV-496, and what does this alignment ensure?

A. Both valves are open. This ensures that a water hammer event resulting from an ASW pump trip and restart will not affect both trains.

B. Both valves are closed. This ensures that a water hammer event resulting from an ASW pump trip and restart will not affect both trains.

C. Both valves are open. This ensures that a single active failure will not result in a significant reduction in heat removal capability.

D. Both valves are closed. This ensures that train separation will not be compromised.

Proposed Answer: C

Explanation:

A. Incorrect. With FCV-495 and 496 both open, the headers are tied together. This will allow a water hammer event to affect both trains. Vacuum reliefs perform this function.

B. Incorrect. With FCVV-495 and 496 both closed, a water hammer event will not affect both trains. This alignment is required by ECG 17.4 but only if both vacuum relief valves in one train are inoperable. This is NOT the normal arrangement

C. Correct. FCV-495 and 496 are both open. This ensure that a single active failure that will not result in a significant reduction in heat removal capability

D. Incorrect. If both FCV-495 and 496 are both closed this would ensure that train separation will not be compromised but this is NOT the normal arrangement.

Technical References: STG-E5. T/S 3.7.8 basis ECG 17.4
References to be provided to applicants during exam: None
Learning Objective: 37011 - State the purpose of the ASW system components

Question Source: Bank #
(note changes; attach parent) Modified Bank #

	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41 (7)	

Changed SWS to ASW for A and B. rnf5 09/08/09

Examination Outline Cross-Reference	**Level**	RO
Knowledge of bus power supplies to the following: service water	**Tier #**	2
	Group #	1
	K/A #	076 K2.01
	Rating	3.1

Question 25

What are the power supplies for the Unit 1 Auxiliary Saltwater Pumps, 1-1 and 1-2?

A. Pump 1-1 is powered from Bus F; Pump 1-2 is powered from Bus G

B. Pump 1-1 is powered from Bus G; Pump 1-2 is powered from Bus H

C. Pump 1-1 is powered from Bus F; Pump 1-2 is powered from Bus H

D. Pump 1-1 is powered from Bus H; Pump 1-2 is powered from Bus G

Proposed Answer: A

Explanation:

A Correct: Per OIM J-1, ASW pumps 21 and 22 are powered from buses F and G..

B Incorrect. G and H provide power to Unit 1 ASW pumps.

C Incorrect: Correct supply for 2-1 but 22 is G.

D Incorrect: Correct supply for 2-2 but 21 is F.

Technical References: STG E-5, OIM J-1-1.

References to be provided to applicants during exam: None

Learning Objective: 5339 - State the power supplies to ASW system components

Question Source:	Bank #	X
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41 (7)	

Examination Outline Cross-Reference	Level	RO
Knowledge of bus power supplies to the following: Emergency air compressor	**Tier #**	2
	Group #	1
	K/A #	078K2.02
	Rating	3.3

Question 26

A loss of power to which of the following buses / load centers would have the most significant impact on the plant's ability to maintain normal instrument air header pressure (100 psig – 108 psig)?

A. Unit 2, vital 4KV bus H
B. Unit 1, vital 480V bus/load center G
C. Unit 2, non-vital 480V bus/load center 22J
D. Unit 1, non-vital 480V bus/load center 15E

Proposed Answer: D
Explanation:

A. Incorrect. Compressors at both units are powered from non-vital sources.

B. Incorrect. Compressors at both units are powered from non-vital sources

C. Incorrect: 22J is non-vital 480VAC but does not power any air compressors.

D. Correct: Bus/load center 15E supplies power to one reciprocating and one rotary compressor.

Technical References: STG K-1
References to be provided to applicants during exam: None
Learning Objective: 7225 - State the power supplies to Compressed Air System components

Question Source:	Bank #	X
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41 (7)	

Examination Outline Cross-Reference	Level	RO
Knowledge of the effect that a loss or malfunction of the IAS will have on the following: Containment air system.	**Tier #**	2
	Group #	1
	K/A #	078 K3.01
	Rating	3.1

Question 27

GIVEN:

- Unit 1 was at 100% power with all systems in automatic control
- Instrument Air pressure is 80 psig and lowering
- The SFM has directed entry to OP-AP-9, Loss of Instrument Air
- Immediately after entering OP-AP-9, the reactor trips

Which of the following describes the effect on air-operated valves inside containment and how the plant will be stabilized (post-trip)?

A. No effect. All air operated valves were designed to fail to the safe position.

B. No effect. Letdown and charging valves modulate properly on N2 backup; no additional operator action is required.

C. Pressurizer spray valves fail closed. Operator must manually control pressure by cycling proportional heaters on/off.

D. PORVs 474, 455C and 456 fail closed and letdown isolates. Operator must restore letdown to avoid lifting a Pressurizer safety valve.

Proposed Answer: C

Explanation:

A. Incorrect. All air operated valves are fail safe, but with no operate action the pressurizer will over fill as letdown is isolated and charging flow continues. Since Pzr sprays fail shut, a PORV lift will occur.

B. Incorrect. Letdown must be restored by <u>manually</u> opening MS 1-902(sealed valve) and N2-1-34. (Appendix A page 17)

C. Correct. See Op AP-9 step 5b

D. Incorrect. PORV 455C and 456 will operate on backup N2.

Technical References STG K1 page 3-13, STG B1A pages 2.1-3&10, STG 4A page 2.1-46 and Procedure OP AOP-9

References to be provided to applicants during exam: None

Learning Objective: 7209 - Discuss abnormal conditions associated with the Compressed Air

System

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41 (7)	

Examination Outline Cross-Reference	Level	RO
Knowledge of the physical connections and/or cause effect relationships between the containment system and the following: SIS, including action of safety injection reset	**Tier #**	2
	Group #	1
	K/A #	103 K1.08
	Rating	3.6

Question 28

Some time after an accident, containment pressure rises and indicates the following on VB1:

- PI-934 = 20 psig
- PI-935 = 23 psig
- PI-936 = 20 psig
- PI-937 = 24 psig

Containment spray does not actuate.

Which of the following would explain why spray did not actuate?

A. Safety Injection signal has been reset.

B. RWST is below the low level alarm setpoint.

C. Containment Isolation Phase A has failed to actuate.

D. Not enough channels of containment pressure are at/above the nominal setpoint.

Proposed Answer:

A. Safety Injection signal has been reset.

Explanation:
A correct, an SI signal must be present
B incorrect, RWST level would not interfere with actuation and pumps trip at lo-lo level.
C incorrect, no interface.
D incorrect, setpoint is 2/4 at 22 psig

Technical Reference(s):
Drawings 4014233, SSPS Functional Diagram and 498006 Containment Pressure
STG I2, Containment Spray System, page 2-9

Proposed references to be provided to applicants during examination: None

Learning Objective: 5422 Explain the consequences of SSPS failures on plant operation

Question Source: Bank #
Modified Bank # P-49415
New _______

Question History: Last NRC Exam DCPP RO Exam 1995

Question Cognitive Level:
Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 41.7
55.43 _____

Comments: K/A: 006 K1.02 - Knowledge of the physical connections and/or cause effect relationships between the ECCS and the following: ESFAS

Examination Outline Cross-Reference	Level	RO
Knowledge of bus power supplies to the following: One-line diagram of power supply to trip breakers.	**Tier #**	2
	Group #	2
	K/A #	001 K2.02
	Rating	3.6

Question 29

Which of the following sequence of power supplies describes the source of power for a Reactor Trip Breaker Under Voltage Coil?

A. 480 VAC Bus F → PY-11 → SSPS 48VDC

B. 480 VAC Bus G → PY-12 → Eagle 21 48VDC

C. 125 VDC Bus 11 → SSPS 15 VDC

D. 125 VDC Bus 12 → Reactor Trip Breaker Control Power

Answer: A

Explanation:

A. Correct – Bus F supplies PY-11 which then supplies Train A 48VDC in SSPS
B. Incorrect – Bus G and PY-12 are correct, but Eagle 21 does not supply power; however, it does supply inputs to SSPS to trip the reactor.
C. Incorrect – Plausible if the applicant thinks 15 VDC power supply for reactor trip breaker UV coils comes from 125VDC. SSPS is correct, however, 15 VDC supplies circuit cards and test slave relays.
D. Incorrect – This is the sequence for the shunt trip coils

Technical References: B-6B (Eagle 21 and Solid State Protection System), J-10 (Instrument AC system)
References to be provided to applicants during exam: None
Learning Objective: 3291 - State the power supplies to the Eagle-21/SSPS components

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41 (7)	

Examination Outline Cross-Reference	**Level**	RO
Knowledge of annunciator alarms, indications, or response procedures.	**Tier #**	2
	Group #	2
	K/A #	002 G2.4.31
	Rating	4.2

Question 30

GIVEN:

- Unit 1 is in MODE 5
- LTOP is in service

RCS pressure rises to 445 psig and PK02-16, RHR System, alarms.

Based on this, PORV PCV-455C should be _______; and, RHR-8702, RHR suction valve, is _______.

A. Closed; Open

B. Closed; Closed and blocked from opening

C. Open; Open

D. Open; Closed and blocked from opening

Answer: C

Explanation:

A. Incorrect – PORV should be open

B. Incorrect – The PORV should be open

C. Correct – the PORV should be open and the alarm means that RCS pressure is high and the suction valve is open.

D. Incorrect – PK02-16 alarming means suction valve is open

Technical References: System Training Guide A-1, Reactor Coolant System
References to be provided to applicants during exam: None
Learning Objective: 36923 - Analyze automatic features and interlocks associated with the Pzr, Pzr Pressure and Level Control System
35317 - Analyze automatic features and interlocks associated with the RHR system

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41 (3, 5, 7)	

Examination Outline Cross-Reference	Level	RO
Ability to manually operate and/or monitor in the control room: NIS indicators	**Tier #**	2
	Group #	2
	K/A #	015 A4.02
	Rating	3.9

Question 31

GIVEN:

- A reactor startup is in progress on Unit 1
- SR Channel N-31 indicates 2 x 10^3 cps
- SR Channel N-32 indicates 2 x 10^3 cps
- IR Channel N-35 indicates 2 x 10^{-9} amps
- IR Channel N-36 indicates 2 x 10^{-11} amps

Which of the following describes the response of the Intermediate Range Excore Nuclear Instruments?

A. IR Channel N-35 is reading abnormally high; P-6 permissive is NOT enabled.

B. IR Channel N-36 is reading abnormally low; P-6 permissive is enabled.

C. IR Channel N-35 is reading abnormally high; P-6 permissive is enabled.

D. IR Channel N-36 is reading abnormally low; P-6 permissive is NOT enabled.

Answer: C

Explanation:

A. Incorrect. IR N-35 is high, but the logic for P-6 is satisified w/1 of 2 IR above the setpoint of 1 x 10^{-10} ICA.

B. Incorrect. N-36 is reading approximately what it should be for the current source range counts.

C. Correct – Based SR/IR overlap criteria, IR Channel N-35 is reading abnormally high for existing conditions; P-6 permissive is enabled when ONE IR channel > 10^{-10} amps. Manual action is required to de-energize SRHV.

D. Incorrect. N-36 reading is correct, and P-6 permissive is enabled.

Technical References: System Lesson Guide B-4, Excore Nuclear Instrumentation
OIM page B-4-1
References to be provided to applicants during exam: None
Learning Objective: 5992 - Discuss abnormal conditions associated with the NIS

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41 (2, 6)	

Examination Outline Cross-Reference	**Level**	RO
Ability to monitor automatic operation of the NNIS, including: Automatic selection of NNIS inputs to control systems	**Tier #**	2
	Group #	2
	K/A #	016 A3.01
	Rating	2.9

Question 32

GIVEN:

- Unit 1 is operating at 100%
- PK05-21 PZR LEVEL HI/LO is alarming
- PK05-22 PZR LEVEL HI/LO CONTROL is alarming
- PZR level is rising
- PZR heaters are de-energized

The Controlling Pressurizer level has failed ________________________________.

A. HIGH; actual PZR level will stabilize at a new higher level.

B. LOW; actual PZR level will stabilize at a new higher level.

C. HIGH; actual PZR level will increase until a high level reactor trip occurs.

D. LOW; actual PZR level will increase until a high level reactor trip occurs.

Answer: D

Explanation:

A. Incorrect – Controlling PZR level has failed LOW; Letdown will isolate, and without operator action to switch to a non-faulted controlling channel, pressurizer level will increase until a high level reactor trip occurs.

B. Incorrect – PZR level will not stabilize at a new higher level.

C. Incorrect – Controlling PZR level has failed LOW.

D. Correct – Initial indications of a failed PZR level controller failing LOW are heaters DE-ENERGIZING and pressurizer level INCREASING. Although pressurizer level will ultimately increase on a failed level controller failing HIGH, due to letdown isolation, the initial indications of a failed PZR channel HIGH are Backup heaters ENERGIZING and pressurizer level DECREASING. Letdown will isolate in both conditions and ultimately lead to a high level reactor trip without operator action.

Technical References: Operator Information Manual , A-4-2a (Pressurizer Level Control), A-4-2b (Pressurizer Level Channel Failures)
References to be provided to applicants during exam: None
Learning Objective: 36926 - Discuss abnormal conditions associated with the Pzr, Pzr Pressure and Level Control System

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41 (3, 7)	

Examination Outline Cross-Reference	**Level**	RO
Ability to (a) predict the impacts of the following malfunctions or operations on the ITM system; and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations: Core damage	**Tier #**	2
	Group #	2
	K/A #	017 A2.02
	Rating	3.6

Question 33

GIVEN:
- A small break LOCA has occurred
- In-core thermocouples indicate 1210°F

Which of the following describes the status of the Core Cooling Critical Safety Function (CSF) and the preferred mitigation strategy to restore core cooling?

A. The CSF is MAGENTA. RCPs should be restarted.

B. The CSF is RED. RCPs should be restarted.

C. The CSF is MAGENTA. Restore ECCS flow.

D. The CSF is RED. Restore ECCS flow.

Answer: D

Explanation:

A incorrect. Incores above 1200°F indicate inadequate core cooling. Restart of RCPs is the least preferred method.
B incorrect. Incores above 1200°F indicate inadequate core cooling. However, restoring ECCS flow is the preferred method, RCP restart is the least preferred.
C incorrect. Core cooling is inadequate.
D correct. Core cooling is inadequate and the preferred method of addressing the situation is to restore ECCS flow.

Technical References: FR-C.1
References to be provided to applicants during exam: None
Learning Objective: 6811 - Explain the most effective method to restore adequate core cooling
11307 - Define inadequate core cooling.

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X

Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41 (2, 7)	

Examination Outline Cross-Reference	**Level**	RO
Ability to predict and/or monitor changes in parameter (to prevent exceeding design limits) associated with operating the HRPS controls including: Hydrogen concentration	**Tier #**	2
	Group #	2
	K/A #	028 A1.01
	Rating	3.4

Question 34

GIVEN:

- Large Break LOCA is in progress
- Core damage is occurring
- Containment Hydrogen concentration is 3.2%

Which of the following describes the <u>minimum</u> equipment that is needed in operation in order to maintain hydrogen at or below the current concentration?

A. ONE Recombiner and the Containment Purge System in service.

B. ONE Recombiner ONLY in service.

C. ONE Recombiner and Containment Spray System in service.

D. BOTH Recombiners in service.

Answer: B

Explanation:

A. Incorrect – Each Containment Hydrogen Recombiner has 100% capacity. The Containment Purge System is to be used as a back up to the CHRs once hydrogen concentration reaches 3.5% by volume.

B. Correct – Only one Recombiner is needed to maintain hydrogen concentration in Containment below 4% by volume.

C. Incorrect – Containment Spray is only placed in service once Containment reaches 22 psig, well above the pressure when the Recombiners would be placed in service.

D. Incorrect – Only one Recombiner is needed to maintain hydrogen concentration in Containment below 4% by volume.

Technical References: System Lesson Guide H-8 (Containment Hydrogen Purge System), SLG H-9 (Containment Hydrogen Recombiners), SLG I-2 (Containment Spray System)
References to be provided to applicants during exam: None

Learning Objective: 40834 - Explain significant CHPS design features and the importance to nuclear safety

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41 (7, 9)	

Note: deleted reference to recombiners in service and about to be inservice. Confuses what the question is trying to test. Rnf5, 9/10/09

Examination Outline Cross-Reference	Level	RO
Knowledge of the effect of a loss or malfunction on the following will have on the SDS: Controller and positioners, including ICS, S/G, CRDS	**Tier #**	2
	Group #	2
	K/A #	041 K6.03
	Rating	2.7

Question 35

GIVEN:

- Unit 1 has tripped from 100% power
- Reactor Trip Breaker ‘A’ is open
- Reactor Trip Breaker ‘B’ is closed
- Steam Dump Mode Select Switch is selected to Tavg

What is the status of the Steam Dump System control logic?

A. Groups 1 and 2 steam dump valves will actuate on the Load Rejection Controller; Groups 3 and 4 are blocked.

B. Groups 1, 2, 3, and 4 steam dump valves will actuate on the Reactor Trip Controller.

C. Groups 1, 2, 3, and 4 steam dump valves will actuate on the Load Rejection Controller.

D. Groups 1, 2, and 3 steam dump valves will actuate on the Reactor Trip Controller; Group 4 is blocked.

Answer: A

Explanation:

A. Correct – Groups 1 and 2 will actuate on the Load Rejection Controller, which modulates due to Reactor Trip Breaker ‘B’ remaining closed. Groups 3 and 4 are blocked due to a lack of arming signal with Reactor Trip Breaker ‘A’ opening. Group 4 may open but due to the individual pressure controller, not due to the steam dump logic.

B. Incorrect – Groups 1 and 2 will not actuate on the Reactor Trip Controller due to Reactor Trip Breaker ‘B’ remaining closed on the trip but will open due to the load rejection controller. Groups 3 and 4 blocked by P-4 train A.

C. Incorrect – Groups 1 and 2 will actuate on the Load Rejection Controller, but Groups 3 and 4 will not.

D. Incorrect – Groups 1 and 2 will actuate on the Load Rejection Controller, not the Reactor Trip Controller. Group 3 is not controlled by the Reactor Trip Controller.

Technical References: OIM C-2-3
References to be provided to applicants during exam: None
Learning Objective: To be determined 8004 - Analyze automatic features and interlocks associated with the Steam Dump System

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	'A' Bank #58
	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41 (7)	

Examination Outline Cross-Reference	Level	RO
Knowledge of the operational implications of the following concepts as they apply to the MT/B System: Purpose of low-power reactor trips (limited to 25% power)	**Tier #**	2
	Group #	2
	K/A #	045 K5.18
	Rating	2.7

Question 36

GIVEN:

- Unit 1 is at 7% power
- The crew is shutting down the plant in accordance with OP L-5, Plant Cooldown From Minimum Load to Cold Shutdown

The Main Unit Turbine EHC malfunctions, and turbine load begins increasing uncontrollably at approximately 50 MW/minute.

If no operator action is taken, the reactor will trip ________.

A. at 25% reactor power, but it will not cause a turbine trip. The reactor trip protects the core from the positive reactivity excursion.

B. at 25% reactor power and cause a turbine trip. The reactor trip protects the core from the positive reactivity excursion.

C. on Power Range Rate trip and cause a turbine trip. The reactor trip protects the core against violating the DNBR limit.

D. on Power Range Rate trip, but it will not cause a turbine trip. The reactor trip protects the core against violating the DNBR limit.

Answer: B

Explanation:
on a plant shutdown, the turbine is taken off line with power below 10%. The at power trips are blocked and low power trips are enabled. Any reactor trip causes a turbine trip regardless of power level.

A. Incorrect – Both the reactor and turbine will trip.

B. Correct – both reactor and turbine will trip on the unblocked high flux (low) to protect against positive reactivity excursion from low power.

C. Incorrect – PR rate is +5%/2 sec. 50 MW/min is approximately 5%/minute, reason for trip is correct.

D. **Incorrect – PR rate is +5%/2 sec. 50 MW/min is approximately 5%/minute. Reason for trip is correct.**

Technical References: System Lesson Guide B-6A (Reactor Protection System). OIM B-6-4a and B-6-4b
References to be provided to applicants during exam: None
Learning Objective: 41316 - Explain significant RPS design features and the importance to nuclear safety
37048 - Analyze automatic features and interlocks associated with the RPS

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41 (5, 6, 7)	

Examination Outline Cross-Reference	Level	RO
Knowledge of ARM system design feature(s) and/or interlock(s) which provide for the following: Containment Ventilation Isolation	**Tier #**	2
	Group #	2
	K/A #	072 K4.01
	Rating	3.3

Question 37

Which of the following radiation monitors, if in high alarm, will cause Containment Purge Exhaust valves (RCV-11 and RCV-12) to automatically close?

A. RM-13, RHR Exhaust Duct Air Particulate monitor

B. RM-44A, Containment Radiation Monitor

C. RM-29, Plant Vent Gross Gamma monitor

D. RM-34, Plant Vent ALARA monitor

Answer: B

Explanation:

A. Incorrect

B. Correct – RM-44A/B in high alarm initiates a containment isolation signal. The containment isolation signal will close RCV-11 and -12.

C. Incorrect

D. Incorrect

Technical References: LH-4 Containment Purge System
References to be provided to applicants during exam: None
Learning Objective: 5119 - Analyze automatic features and interlocks associated with the Containment Purge System

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No

Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41 (7, 11)	

Examination Outline Cross-Reference	**Level**	RO
Knowledge of the physical connections and/or cause-effect relationships between the circulating water system and the following systems: Emergency/essential SWS	**Tier #**	2
	Group #	2
	K/A #	075 K1.08
	Rating	3.2

Question 38

In accordance with OP AP-10, Loss of Auxiliary Saltwater, when is it appropriate to cross-tie the Auxiliary Saltwater (ASW) and the Circulation Water bays?

A. If one unit loses its ASW pumps and the other unit's ASW pumps are not available.

B. If the Circulation Water screens are severely clogged and the ASW screens are not available.

C. If Chlorine injection into the ASW system is necessary.

D. If bay level is low and the ASW pumps are losing suction.

Answer: D

Explanation:

A. Incorrect – The correct action would be to enter OP AP-11, Loss of CCW.

B. Incorrect – This would require isolating the two trains for the affected unit.

C. Incorrect – This would require operating with the opposite unit supplying.

D. Correct – If the bay level is low and the pumps are losing suction (or cavitating), cross-tying would be appropriate if the Circulation Water screens are less affected.

Technical References: AR PK01-03 (Auxiliary Saltwater Pumps), AP-10 (Loss of Auxiliary Salt Water)
References to be provided to applicants during exam: None
Learning Objective: 5354 - State the purpose of actions when restoring ASW after malfunctions

Question Source:	Bank #	X #65 February 2005 Exam

(note changes; attach parent)	Modified Bank # New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	 X
10CFR Part 55 Content:	55.41 (10)	

Examination Outline Cross-Reference	**Level**	RO
Knowledge of the operational implications of the following concepts as they apply to a Pressurizer Vapor Space Accident: Change in leak rate with change in pressure.	**Tier #**	1
	Group #	1
	K/A #	008 AK 1.02
	Rating	3.1

Question 39

GIVEN:

- A Pressurizer PORV fails open and cannot be isolated
- The plant trips and SI actuates
- 30 minutes after the reactor trip the crew enters E-1.2, Post LOCA Cooldown and Depressurization

Which of the following describes the expected plant conditions as the crew enters E-1.2?

A. Break flow is unchanged from its original value; Pressurizer level off-scale high.

B. Break flow has decreased from its original value; Pressurizer level on scale and decreasing.

C. Break flow is unchanged from its original value; Pressurizer level on scale and decreasing.

D. Break flow has decreased from its original value; Pressurizer level off-scale high.

Proposed Answer: D. Break flow has decreased from its original value; Pressurizer level is off-scale high.

Explanation:

Answer A incorrect – RCS pressure will be less than NOP as a result break flow will be reduced.

Answer B incorrect – Pressurizer level will be off scale high (even if still on scale, it would be increasing due to increased SI flow and the open PORV).

Answer C incorrect – Break flow will be reduced.

Answer D correct – lower RCS pressure will result in lower break flow and due to vapor space break, level will be off scale high.

Technical Reference(s): LMCD-FRC page 17 and page 37

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 41697 - Describe the plant response to a loss of reactor coolant including:
- Vapor Space LOCAs

Question Source: Bank # X
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam DCPP 2008

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5
55.43 ______

10 CFR Part 55 Content: Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments:

Examination Outline Cross-Reference	**Level**	RO
Knowledge of the interrelations between the small break LOCA and the following: S/Gs	**Tier #**	1
	Group #	1
	K/A #	009 EK 2.03
	Rating	3.0

Question 40

GIVEN:
- A small break LOCA has just occurred in containment
- RCS pressure is 1600 psig and slowly lowering
- One train of ECCS has actuated
- MSIVs are shut
- All steam generators are available
- All RCPs are running

Is a secondary heat sink necessary in order to maintain adequate core cooling?

A. No, because RCS decay heat removal is through the break.

B. Yes, because this will allow ECCS flow to equal break flow.

C. Yes, because secondary pressure must be maintained below primary pressure during all LOCA events.

D. No, because the RCPs provide core cooling.

Answer: B
Lowered pressure, 1900 is above SI pump shutoff head. Changed D.
Explanation:

A. Incorrect – Secondary heat sink is required in order for the RCS to reach equilibrium pressure, allowing high pressure SI flow to equal break flow, and thus guarantee that core cooling and decay heat removal are adequate.

B. Correct

C. Incorrect – Secondary heat sink is required for a small break LOCA and not for a large break LOCA, when the RCS will completely depressurize.

D. Incorrect – RCPs transport heat to the steam generators for removal.

Technical References: WOG, Background Information E-1, pages 5 and 52
References to be provided to applicants during exam: None
Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41 (7)	

Examination Outline Cross-Reference	**Level**	RO
Knowledge of the reasons for the following responses as they apply to the Large Break LOCA: RCP tripping requirement	**Tier #**	1
	Group #	1
	K/A #	011 EK 3.14
	Rating	4.1

Question 41

GIVEN:

- Unit 1 was at 100% power
- A large break LOCA occurred
- Containment pressure is 25 psig and rising
- The crew has transitioned to E-1, Loss of Reactor or Secondary Coolant

Why will the crew trip the RCPs?

A. Loss of cooling to the motor bearing oil coolers.

B. To preserve them in case they are needed for core cooling later in the event.

C. To prevent severe core uncovery if the RCPs trip later in the event.

D. To maximize the time the steam generators are available as a heat sink.

Answer: A
Lowered question to memory, modified question to directly address KA
Explanation:

A. Correct – loss of cooling occurs when Phase B occurs (22 psig). RCPs are stopped to prevent damage to the motors due to overheating.

B. Incorrect – this is the reason in C.1.

C. Incorrect – This is the design basis for a small break LOCA, not a large break LOCA

D. Incorrect – This is why RCPs are tripped for loss of secondary heat sink.

Technical References: Background Information WOG, Generic Issue, RCP Trip/Restart, pages 7-10
References to be provided to applicants during exam: None
Learning Objective: 5442 - Explain the conditions affecting RCP trip criteria

Question Source: Bank #
(note changes; attach parent) Modified Bank #

	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41 (10)	

Examination Outline Cross-Reference	**Level**	RO
Ability to operate and/or monitor the following as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow): RCP vibrations.	**Tier #**	1
	Group #	1
	K/A #	015/17 AA1.23
	Rating	3.1

Question 42

GIVEN:

- Unit 1 is at 20% Power
- PK05-03, RCP 3 UP BRG L/O RESVR LVL HI/LO, is ON
- RCP 1-3 motor bearing temperature is 222°F
- PK05-05, RCP HIGH VIBRATION, is ON
- Vibration Monitoring System is indicating 25 mils on the Pump Shaft of RCP 1-3

In accordance with OP AP-28, RCP Malfunctions, what action should the operators take?

A. Stop RCP 1-3 and then perform the immediate actions of E-0, Reactor Trip or Safety Injection.

B. Shutdown the reactor per OP AP-25, Rapid Load Reduction, and then stop RCP 1-3.

C. Trip the reactor, enter E-0, Reactor Trip or Safety Injection, and then stop RCP 1-3.

D. Continue to monitor RCP 1-3 conditions, no stopping of the RCP is required at this time.

Answer: C
Lowered power level to make a shutdown more plausible (below one loop loss of flow reactor trip setpoint)
Explanation:

A. Incorrect – This has to be done in conjunction with tripping the reactor.

B. Incorrect – This would be the appropriate action if the #1 seal on RCP 3 had a leak rate of .8 GPM and bearing/motor temperatures were stable. The #1 seal on RCP 3 is intact.

C. Correct – Tripping the reactor and the RCP are the immediate actions per OP AP-28 for any RCP that has any bearing temperature alarm concurrent with a vibration alarm.

D. Incorrect – Foldout page 4.0, RCP trip requirements met.

Technical References: OP AP-28, Reactor Coolant Pump Malfunction, foldout page

References to be provided to applicants during exam: None
Learning Objective: **3477** Given an abnormal condition, summarize the major actions of the abnormal operating procedure to mitigate an event in progress.

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41 (3 and 5)	

Examination Outline Cross-Reference	**Level**	RO
Ability to determine and interpret the following as they apply to the Loss of Reactor Coolant Makeup: Whether charging line leak exists.	**Tier #**	1
	Group #	1
	K/A #	022 AA2.01
	Rating	3.2

Question 43

GIVEN:

- Unit 1 is at 100% power
- Pressurizer level is slowly lowering
- RCS pressure is slowly lowering
- FI-128 (charging flow) is lowering
- PI-142 (charging header pressure) is lowering
- PK11-21, HIGH RADIATION, is ON due to an Aux Bldg area radiation monitor
- The crew has entered OP AP-17, Loss of Charging

Based on this, the most likely location of the leak is _______.

A. between the CCP discharge valves and FCV-128, CCP flow control valve

B. on the charging line, downstream of the regenerative heat exchanger

C. downstream of HCV-142, RCP Seal flow control valve

D. on one of the ECCS cold leg injection lines

Answer: A

Explanation:

A. Correct – since the leak is between the CCP discharge valve and FCV-128 isolation valve, FT-128 will see decreased flow, and thus pressure on the charging header will drop (PT-142 correctly indicating lower pressure).

B. Incorrect – If the leak were downstream of the RHX, the pressure on the header would increase (due to increase charging demand from the Pressurizer which is seeing reduced flow) and FT-128 would see increased flow. Also there is no high radiation alarm coming from containment, where the RHX is located, but from the Auxiliary Building, where the CCPs and FCV-128 are located.

C. Incorrect – HCV-142 is downstream of FT-128. FT-128 would indicate an increase flow due to the demand from a leak downstream of HCV-142 and FT-128.

D. Incorrect – Leak location is incorrect.

Technical References: OP AP-17, Loss of Charging
References to be provided to applicants during exam: None
Learning Objective: 3465 - Identify and discuss RCS leakage paths, including means used to detect and identify leaks.

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41 (5 and 10)	

Examination Outline Cross-Reference	**Level**	RO
Loss of RHR: Knowledge of how abnormal operating procedures are used in conjunction with EOPs.	**Tier #**	1
	Group #	1
	K/A #	025 G 2.4.8
	Rating	3.8

Question 44

Unit 1 is in MODE 5.

A break on the RHR suction line occurs. The RHR pumps begin to cavitate and are secured by the crew.

As a result of the loss of RCS inventory and RHR cooling the crew will…

A. Go to OP AP SD-2, Loss of RCS Inventory and implement E-1, Loss of Reactor or Secondary Coolant.

B. Go to OP AP SD-2, Loss of RCS Inventory and refer to E-1, Loss of Reactor or Secondary Coolant.

C. Go to OP AP SD-2, Loss of RCS Inventory only.

D. Go to E-1, Loss of Reactor or Secondary Coolant only.

Answer: C

Explanation:

A. Incorrect – EOPs do not apply in MODE 5. E-1 is not used at this time.

B. Incorrect – EOPs do not apply in MODE 5. E-1 is not used at this time.

C. Correct. Only AP SD-2 would be used.

D. Incorrect – E-1 does not apply in MODE 5.

Technical References: OP AP SD-0 (Loss of or inadequate decay heat removal), OP AP SD-1 (Loss of AC Power)
References to be provided to applicants during exam: None
Learning Objective: 3478 - Given initial conditions, assumptions, and symptoms, determine the correct abnormal operating procedure to be used to mitigate an operational event

Question Source: Bank #

(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41 (5 and 10)	

Examination Outline Cross-Reference	**Level**	RO
Ability to determine and interpret the following as they apply to the Loss of Component Cooling Water: The cause of possible CCW loss.	**Tier #**	1
	Group #	1
	K/A #	026 AA2.02
	Rating	2.9

Question 45

GIVEN:
- Unit 1 is at full power
- CCW A header flow has increased
- CCW flow on headers B and C has decreased
- Unit 1 CCW Surge Tank level is decreasing
- Containment Structure sump level is rising

Which of the following is a possible location of the Component Cooling Water leak?

A. CFCU 1-4

B. RCP 1-3 thermal barrier return line

C. Seal Water Heat exchanger inlet line

D. RHR Heat Exchanger 1-2 inlet line

Answer: A

Explanation:

A. Correct – Vital header alarm due to low flow. Because containment sump level is increasing, and only one side of the surge tank is decreasing, the leak must be on a vital CCW header load in containment (ie CFCU).

B. Incorrect – Inside Containment but not a vital header load. No Header C alarm.

C. Incorrect – Not vital or located inside Containment, but could be at a lower pressure than CCW pressure and a source of outleakage.

D. Incorrect – Vital load and could cause the vital header alarm, but not inside Containment.

Technical References: AR PK01-06, AR PK01-07
References to be provided to applicants during exam: None
Learning Objective: 3466 - Discuss the effects and actions associated with a loss of CCW.

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41 (5 and 10)	

Examination Outline Cross-Reference	**Level**	RO
Ability to operate and monitor the following as they apply to an ATWS: Reactor trip switch pushbutton.	**Tier #**	1
	Group #	1
	K/A #	029 EA1.08
	Rating	4.5

Question 46

RCS pressure begins to rapidly decrease. The operator takes the Reactor Trip Switch to "TRIP" and PK04-14, Reactor Trip Actuated, alarms.

Which of the following has occurred?

A. There is a demand for a reactor trip but the trip breakers are still closed.

B. A manual reactor trip demand signal has been generated by the Reactor Protection System.

C. RCS pressure has decreased below the Low Pressure Trip setpoint generating an automatic reactor trip demand.

D. The manual trip was successful in opening at least one or both reactor trip breakers.

Answer: D
Removed window dressing and changed question.
Explanation:

A. Incorrect –

B. Incorrect –

C. Incorrect –

D. Correct. Alarm actuates when both trains actuate. Does not differentiate between manual or auto trip.

Technical References: AR PK04-14
References to be provided to applicants during exam: None
Learning Objective: 37049 - Describe controls, indications, and alarms associated with the RPS

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X

Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41 (5 and 10)	

Examination Outline Cross-Reference	**Level**	RO
Knowledge of the reasons for the following responses as they apply to the SGTR: Equalizing pressure on primary and secondary sides of ruptured S/G.	**Tier #**	1
	Group #	1
	K/A #	038 EK3.01
	Rating	4.1

Question 47

A Steam Generator Tube Rupture has occurred and EOP E-3, "Steam Generator Tube Rupture", has been entered. Depressurization of the RCS is in progress.

The PRIMARY goal of depressurizing the RCS to match the pressure of the ruptured Steam Generator is for which of the following reasons?

A. To refill the Pressurizer and allow the restoration of Letdown and normal pressure control.

B. To refill the RCS and collapse any voids in the reactor vessel head.

C. To maximize sub-cooling to help prevent the transfer of primary coolant to the secondary system.

D. To minimize sub-cooling to help prevent the transfer of primary coolant to the secondary system.

Answer: D
Replaced with Ruptured, not faulted
Explanation:

A. Incorrect – Per the Caution of EOP E-3, step 35, "An UNMONITORED atmospheric release MAY occur IF RCS <u>or</u> Ruptured S/Gs Pressures EXCEED 1040 PSIG".

B. Incorrect – The PRIMARY concern of equalizing RCS pressure and faulted Steam Generator pressure, with the addition of MINIMIZING sub-cooling, is to stop the leak and prevent further primary coolant from entering the secondary system.

C. Incorrect

D. Correct

Technical References: EOP E-3 (Steam Generator Tube Rupture), AOP AP-3 (Steam Generator Tube Failure), EOP E-3 Bases,
References to be provided to applicants during exam: None

Learning Objective: 7920 - Explain basis of emergency procedure steps

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41 (5 and 10)	

Examination Outline Cross-Reference	**Level**	RO
Knowledge of the interrelations between the Steam Line Rupture and the following: Sensors and detectors.	**Tier #**	1
	Group #	1
	K/A #	040 AK2.02
	Rating	2.6

Question 48

The plant is at normal operating temperature and pressure.

Which of the following would satisfy the MINIMUM coincidence necessary to cause Main Steam Isolation actuation?

A. 1 of 4 Containment pressure channels sensing greater than or equal to 3 psig.

B. 2 of 4 Containment pressure channels sensing greater than or equal to 3 psig.

C. 1 of 4 Containment pressure channels sensing greater than or equal to 22 psig.

D. 2 of 4 Containment pressure channels sensing greater than or equal to 22 psig.

Answer: D

Explanation:

A. Incorrect

B. Incorrect

C. Incorrect

D. Correct – Main Steam Line Isolation Actuation signal is generated by two of four Containment pressure channels reading greater than or equal to 22 psig. Containment Phase 'B' CI signal is generated by this same logic.

Technical References: LB-6A, Reactor Protection System
References to be provided to applicants during exam: None
Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No

Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41 (7)	

Examination Outline Cross-Reference	**Level**	RO
Knowledge of the operational implications of the following concepts as they apply to the Station Blackout: <u>Effect of battery discharge rates on capacity.</u>	**Tier #**	1
	Group #	1
	K/A #	055 EK1.01
	Rating	3.3

Question 49

The site is in a station blackout condition and the operators have entered ECA-0.0 (Loss of All Vital AC Power). Step 15 directs the operators to begin shedding non-essential DC loads.

What is the design capacity of the vital DC batteries and the primary reason for the load shed?

A. Two hours battery capacity – To prevent a potentially explosive hydrogen build-up in the battery rooms due to high discharge.

B. Two hours battery capacity – To conserve the battery for monitoring and control of the plant until power can be restored.

C. Three hours battery capacity – To prevent a potentially explosive hydrogen build-up in the battery rooms due to high discharge.

D. Three hours battery capacity – To conserve the battery for monitoring and control of the plant until power can be restored.

Answer: B

Explanation:

A. Incorrect – design is 2 hours, load shed is to conserve the battery.

B. Correct – 2 hour design and conserve the battery (step 15 basis).

C. Incorrect – Battery capacity is two hours, not three hours.

D. Incorrect - Battery capacity is two hours, not three hours.

Technical References: ECA-0.0 (Loss of All Vital AC Power), J-9 (DC Power)
References to be provided to applicants during exam: None
Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41 (7 and 8)	

Examination Outline Cross-Reference	**Level**	RO
Knowledge of the operational implications of the following concepts as they apply to Loss of Offsite Power: Definition of saturation conditions, implication for the systems.	**Tier #**	1
	Group #	1
	K/A #	056 AK1.04
	Rating	3.1

Question 50

The reactor has tripped from 100% power and Safety Injection is in progress when the following events occur:

- Seismic event in the Buttonwillow area causing loss of offsite power
- PORV-474 has lifted and stuck open
- Isolation Valve 8000A has failed open on the loss of power
- Pressurizer level at 66% and increasing
- RCS temperature at 550° F
- RCS pressure at 1015 psig

What is the condition of the RCS and the primary concern associated with this condition?

A. 25° F sub-cooled, which is insufficient as a prerequisite to re-starting a RCP.

B. Saturated, boiling is now occurring in the RCS which may bind RCPs.

C. Saturated, boiling is now occurring in the core which could uncover the fuel.

D. 25° F sub-cooled, boiling is now occurring in the core with the potential to uncover fuel.

Answer: C

Explanation:

A. Incorrect – The core is at saturation conditions, therefore sub-cooling do not exist. The primary concern is uncovering fuel as the coolant boils away, not re-starting a RCP.

B. Incorrect – Core is at saturation, but uncovering the fuel is the primary concern, not binding the RCPs.

C. Correct

D. **Incorrect – Sub-cooling does not exist when the RCS is at saturation.**

Technical References: LA-4A (Pressurizer, Pressure and Level Control)

References to be provided to applicants during exam: None
Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41 (5)	

Examination Outline Cross-Reference	**Level**	RO
Knowledge of the reasons for the following responses as they apply to the Loss of Vital AC Instrument Bus: Actions contained in EOP for loss of vital AC electrical instrument bus.	**Tier #**	1
	Group #	1
	K/A #	057 AK3.01
	Rating	4.1

Question 51

GIVEN:

- Unit 1 operating at 100% power
- Channel I bistable status lights – lit
- Rods are moving in prior to the operator placing them in MANUAL
- PK19-19 (VITAL UPS FAILURE) Alarming

Which of the following identifies the failure that occurred and what caused rods to begin to insert?

A. Loss of Vital Bus PY-11 causing PT-505 (Tref) to fail low.

B. Loss of Vital Bus PY-12 causing PT-505 (Tref) to fail low.

C. Loss of Vital Bus PY-11 causing Power Range N-41 to fail low.

D. Loss of Vital Bus PY-12 causing Power Range N-42 to fail low.

Answer: A
Removed redundant action in all answers and moved to question.
Explanation:

A. Correct – PT-505 (Tref) powered from PY-11 has failed low on a loss of power to PY-11, causing rods to move in to bring Tave back to Tref. Operator should take manual control of rods per AP-4 once it is recognized that rods are moving in due to the loss of PY-11.

B. Incorrect – If power were lost to PY-12, the Channel II bistable lights would be lit, not Channel I. PT-505 modules would fail low, also causing rods to insert.

C. Incorrect – Power Range N-41 does fail on the loss of PY-11, but the failure of one PR NI does not affect rod control. Rods are moving in due to the failure of PT-505.

D. Incorrect – PY-11 has failed, PR failure would not cause rods to insert.

Technical References: AP-4 (Loss of Vital or Non-Vital Instrument AC), LPA-4 (Loss of

Vital or Non-Vital Instrument AC), J10 (Instrument AC System)
References to be provided to applicants during exam: None
Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41 (5 and 7)	

Examination Outline Cross-Reference	Level	RO
Ability to operate and/or monitor the following as they apply to the Loss of Nuclear Service Water (SWS): Loads on the SWS in the control room.	**Tier #**	1
	Group #	1
	K/A #	062 AA1.02
	Rating	3.2

Question 52

Which of the following two ASW parameters would indicate a fouling of an in service CCW heat exchanger?

A. Low header pressure and high pump amps

B. Low header pressure and low pump amps

C. High CCW heat exchanger differential pressure and low pump amps

D. High CCW heat exchanger differential pressure and high pump amps

Answer: C

Explanation:

A. Incorrect – this is indication of system rupture

B. Incorrect – variation of system rupture and there is low amps with fouled heat exchanger

C. Correct. Fouling causes dp to increase, and amps will decrease.

D. Incorrect. Pump amps will be low.

Technical References: OP AP-10, Loss of Aux Saltwater
References to be provided to applicants during exam: None
Learning Objective: 3477 - Given an abnormal condition, summarize the major actions of the abnormal operating procedure to mitigate an event in progress.

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No

Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41 (5 and 7)	

Examination Outline Cross-Reference	**Level**	RO
Ability to determine and interpret the following as they apply to the Loss of Instrument Air: When to trip reactor if instrument air pressure is decreasing.	**Tier #**	1
	Group #	1
	K/A #	065 AA2.06
	Rating	3.6

Question 53

GIVEN:

- Narrow range Steam Generator levels 58% and decreasing
- Instrument Air pressure is 69 psig and decreasing
- Letdown has automatically isolated
- RCS pressure is 2295 psig and increasing
- PK13-16, PLANT INSTRUMENT AIR is alarming

After entering OP AP-9, Loss of Instrument Air, which of the following describes the operator action?

A. Take manual control to maintain Pressurizer level at program.

B. Bypass the in-service Instrument Air Dryer.

C. Take manual control of Digital Feedwater Control System.

D. Trip the reactor and enter EOP E-0, Reactor Trip or Safety Injection.

Answer: D

Explanation:

A. Incorrect – OP AP-9 (Loss of Instrument Air), step 1, directs the operator to trip the reactor and enter EOP E-0 if S/G levels, PZR levels, PZR pressure can not be maintained within their normal band.

B. Incorrect

C. Incorrect

D. Correct

Technical References: OP AP-9 (Loss of Instrument Air), LPA-9 (OP AP-9, Loss of Instrument Air)
References to be provided to applicants during exam: None
Learning Objective: 7927 - Given initial conditions and assumptions, determine if a reactor trip or safety injection actuation is required

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41 (5 and 7)	

Examination Outline Cross-Reference	Level	RO
Loss of ECR: Ability to locate and operate components, including local controls.	**Tier #**	1
	Group #	1
	K/A #	W/E11 G 2.1.30
	Rating	4.4

Question 54

GIVEN:

- Operators have entered ECA-1.1, Loss of Emergency Coolant Recirculation
- The crew is using Appendix M to makeup to the RWST from the Spent Fuel Pit
- Step 1 is verify Spent Fuel Pit pumps are OFF

Where will the operator go to stop the running 1-1 Spent Fuel Pool pump by pushing the STOP pushbutton?

A. At a panel in the AFW motor driven pump room

B. At the breaker on the appropriate Vital 480 V MCC

C. At a local panel near the Spent Fuel Pool

D. At the breaker on the appropriate Non-Vital 480 V MCC

Answer: A

Explanation:

A. Correct – controls are on a panel just outside the SFP pump room

B. Incorrect - only the breaker control there.

C. Incorrect – this has the indications for heat exchanger pressure (for adjusting dp)

D. Incorrect – power supply is vital G and H.

Technical References: EOP ECA-1.1 (Loss of Emergency Coolant Recirculation), Appendix M
References to be provided to applicants during exam: None
Learning Objective: 5275 - Identify the location of components associated with the SFP cooling system

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41 (5 and 7)	

Examination Outline Cross-Reference	**Level**	RO
Ability to determine and interpret the following as they apply to the (Loss of Secondary Heat Sink): Facility conditions and selection of appropriate procedures during abnormal and emergency operations.	**Tier #**	1
	Group #	1
	K/A #	W/E05 EA2.1
	Rating	3.4

Question 55

GIVEN:

- Reactor tripped and safety injection has actuated
- RCS press 1400 psig
- RCS Th is 425°F
- Steam Generator 1-1 pressure is 0 psig
- Steam Generator 1-1 Wide Range level is off scale low
- Intact steam generator Wide Range levels are 15%
- Intact steam generator pressures are 915 psig
- Crew has exited EOP E.0, Reactor Trip or Safety Injection, and entered EOP FR-H.1, Response to Loss of Secondary Heat Sink due to loss of all AFW flow

Which of the following describes the requirement of the Steam Generators and the operator action?

A. Required for RCS heat removal. Continue attempts to establish AFW flow in accordance with EOP FR-H.1.

B. Not Required for RCS heat removal. Trip RCPs and return to EOP E-0.

C. Required for RCS heat removal. Trip RCPs and initiate Bleed and Feed.

D. Not Required for RCS heat removal. Trip RCPs and enter EOP E-2, Faulted Steam Generator Isolation.

Answer: C

Explanation:

A. Incorrect – Steam Generators ARE required for RCS heat removal. Once AFW flow is established the next step will return you to the procedure and step in effect.

B. Incorrect – Steam Generators ARE required for RCS heat removal. In FR-H.1 after RCPs are tripped you will continue in this procedure in order to establish Bleed and Feed.

C. Correct – Steam Generators ARE required for RCS heat removal. Once RCPs are

tripped in step 5, the procedure continues to establish Bleed and Feed in order to remove decay heat.

D. **Incorrect – Steam Generators ARE required for heat removal. Once RCPs are tripped in step 5, the procedure continues to establish Bleed and Feed in order to remove decay heat.**

Technical References: EOP E.0 (Reactor Trip or Safety Injection0, EOP FR-H.1 (Response to Loss of Secondary Heat Sink)
References to be provided to applicants during exam: None
Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41 (5 and 7)	

Examination Outline Cross-Reference	**Level**	RO
Ability to operate and/or monitor the following as they apply to Generator Voltage and Electric Grid Disturbances: Turbine/generator controls.	**Tier #**	1
	Group #	1
	K/A #	077 AA1.02
	Rating	3.8

Question 56

GIVEN:

- Reactor power has been lowered to 42% due to grid stability problems
- PK12-06, TURBINE, alarms (Input 692, Bearing Oil Pressure Low)
- Bearing Oil Pressure on VB-4 at 5 psig and decreasing slowly
- Standby bearing oil pump is running
- The crew has entered OP AP-29, Main Turbine Malfunction

Which of the following describes the required operator action?

A. Trip the turbine and enter E-0, Reactor Trip or Safety Injection, based on the subsequent reactor trip.

B. Trip the turbine and continue with OP AP-29, Main Turbine Malfunction.

C. Trip the reactor and enter E-0, Reactor Trip or Safety Injection.

D. Continue to reduce turbine load in accordance with AP-29, Main Turbine Malfunction.

Answer: B

Explanation:

A. Incorrect – Tripping the turbine is correct, but with reactor power at less than 50% (P-9 Permissive) the reactor will not automatically trip upon a turbine trip, thus negating the need to transition to E-0.

B. Correct – An immediate trip for the turbine, per AP-29 foldout page, is necessary when bearing oil pressure falls below 8 psig. AP-29 is the guiding procedure, not E-0, as the reactor will not have tripped since power is below the P-9 setpoint.

C. Incorrect – This is the correct action for a 719 Input (PPC Lube Oil Pressure Alarm), but not for Input 692 (Bearing Oil Low Pressure), which directs you to trip the turbine with bearing oil pressure on VB-4 at less than 8 psig.

D. Incorrect – reducing turbine load will not address the lowering oil pressure

Technical References: AP-29 (Main Turbine Malfunction)
References to be provided to applicants during exam: None
Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41 (5 and 7)	

Examination Outline Cross-Reference	**Level**	RO
Conduct of operations: Ability to interpret and execute procedures.	**Tier #**	1
	Group #	2
	K/A #	028 G 2.1.20
	Rating	4.6

Question 57

The plant was stable at full power with all systems operating normally in automatic control when the following occurs:

- Pressurizer level is 60% and begins to rise
- Pressurizer pressure is 2235 psig and slowly increasing
- Letdown flow isolates
- Charging flow decreases
- PK 05-22, Pzr Level Hi/Lo Control alarms (input 544, PZR Lo Lvl Letdn Iso All Htrs Off)
- All other parameters are normal

Which of the following has occurred?

A. Excessive Reactor Coolant System Leakage

B. Pressurizer level channel failure

C. Loss of Charging

D. Letdown Line break

Answer: B

Explanation:

A. Incorrect – Rising pressurizer level is inconsistent with a RCS leak

B. Correct – The backup pressurizer level control channel has failed low.

C. Incorrect – Charging flow is decreasing as expected in the response to the operable primary controlling level channel.

D. Incorrect – Letdown has isolated as expected due to the backup pressurizer level control channel failing low.

Technical References: AR PK05-22 steps 5.1 and 5.7, STG A4A, OIM A-4-2b

References to be provided to applicants during exam: None
Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(10)	

Examination Outline Cross-Reference	**Level**	RO
Ability to determine and interpret the following as they apply to the Fuel Handling incidents: ARM system indications.	**Tier #**	1
	Group #	2
	K/A #	036 AA2.01
	Rating	3.2

Question 58

Refer to the figure of RM-58 Fuel Handling Building (FHB) Radiation Control Module for the question that follows:

Given the following indications:

- Red Light is ON for Trip 2
- Amber Light is ON for Trip 1
- Green Operate Light is ON

Based on the indications, what automatic actions should have occurred?

A. FHB Evacuation Alarm only.

B. FHB Ventilation swapped to Iodine removal Mode and FHB Evacuation Alarm

C. Both Auxiliary Building and FHB Ventilation swapped to Iodine Removal Mode.

D. Only Auxiliary Building Ventilation swapped to Iodine removal mode.

Answer: B

Explanation:

A. Incorrect. This condition would also cause the FHB ventilation system to swap to the Iodine removal mode.

B. Correct. Red light ON indicates a HI Alarm set point has been exceeded and the FHB evacuation alarm should be sounding and Iodine removal ventilation should be in service.

C. Incorrect. Auxiliary Building Ventilation is not swapped to Iodine Removal mode based on RM-58 input signal, only FHB Ventilation is shifted to Iodine Mode.

D. Incorrect. Auxiliary Building Ventilation is not swapped to Iodine Removal mode based on RM-58 input signal.

Technical References: STG G4A, Radiation Monitoring, Pages 2.2-18-23, Rev. 9.
References to be provided to applicants during exam: None

Learning Objective: To be determined

Question Source:	Bank #	X
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43(5) 55.45(13)	

Examination Outline Cross-Reference	**Level**	RO
Ability to operate and/or monitor the following as they apply to the Loss of Condenser Vacuum: Rod position.	**Tier #**	1
	Group #	2
	K/A #	051 AA1.04
	Rating	2.5

Question 59

Given the following conditions:

- The unit is at 90% and ramping up with "MW Feedback In"
- Control Bank D rods are at 210 steps in Auto
- All equipment is operable and in the proper alignment for power operations

Due to condenser seal degradation and subsequent air in-leakage, condenser vacuum begins to slowly degrade.

How is this event expected to affect control rod position?

A. Control rods will slowly step in due to T-avg vs. T-ref and/or power mismatch.

B. Control rods will slowly step out due to T-avg vs. T-ref and/or power mismatch.

C. A demand will exist for control rods to step out, but due to C-11, rods will remain at 210 steps.

D. Control rods remain at 210 steps as degraded condenser vacuum will not affect power or T-avg.

Answer: B

Explanation:

A. Incorrect – As vacuum degrades, efficiency is reduced, but DEHC will attempt to maintain the same load. This means that more MW's will be required from the RCS resulting in a reduced T-avg which results in a *"rods-out"* demand to restore T-avg

B. Correct – As vacuum degrades, efficiency is reduced, but DEHC will attempt to maintain the same load. This means that more MW's will be required from the RCS resulting in a reduced T-avg which results in a "rods-out" demand to restore T-avg

C. Incorrect – C-11 does not stop control bank D withdrawal until its position is 220 steps

D. Incorrect – turbine efficiency will affect reactor power and/or T-avg resulting in auto rod withdrawl.

Technical References: OP-AP-7, Rev. 34, page 2
References to be provided to applicants during exam: None
Learning Objective: To be determined

Question Source:	Bank #	X
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(7)	

–

Examination Outline Cross-Reference	Level	RO
Knowledge of the reasons for the following responses as they apply to the Area Radiation Monitoring system: Guidance contained in alarm response for ARM system.	**Tier #**	1
	Group #	2
	K/A #	061 AK3.02
	Rating	3.4

Question 60

Annunciator Response Procedure AR PK11-17, S.G. Blowdown Hi Rad, directs the operator to verify automatic blowdown isolation.

It then directs the operator to override this signal by placing the RE 19, 23 Hi Rad S/G Blowdown and Sample Valves (O.C.) isolation defeat cutout switch to the "CUT IN" position.

What is the reason for placing the defeat cutout switch in "CUT IN"?

A. Allow flushing the Radiation Monitor to eliminate nuisance alarms.

B. Allow repositioning the HASP#1/HASP#2 switch to be placed in HASP#2 (Leak based setpoint) position.

C. Allow re-flash capability for subsequent alarms.

D. Allow sampling to identify a ruptured Steam Generator.

Answer: D

Explanation:

A. Incorrect – nuisance alarms are eliminated by taking the HASP toggle to HASP#1 (effluent) setpoint position.

B. Incorrect – HASP#1/HASP#2 toggles is normally in HASP#2

C. Incorrect – this switch allows blowdown sample isolation valve to be re-opened

D. Correct – this switch allows blowdown sample isolation valve to be re-opened for chemisty sampling.

Technical References: Procedure AR PK11-17, STG G4a page 2.3-38
References to be provided to applicants during exam: None
Learning Objective: To be determined

Question Source: Bank #
(note changes; attach parent) Modified Bank #

	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(5 and 10)	

Examination Outline Cross-Reference	**Level**	RO
Knowledge of the interrelations between the Control Room evacuation and the following: Reactor trip system.	**Tier #**	1
	Group #	2
	K/A #	068 AK2.02
	Rating	3.7

Question 61

An explosion and lots of smoke caused an immediate control room evacuation.

Which of the following is the ***highest*** priority action that operators should take before evacuating the control room?

A. Manually close MSIVs and bypass valves.

B. Trip the main turbine.

C. Trip the main unit generator.

D. Trip the reactor.

Answer: D

Explanation:

A. Incorrect – True, but this is step 5 of AP-8A. RNO can be done locally.

B. Incorrect – True, but this is step 3 of AP-8A. RNO can be done locally.

C. Incorrect – True, but this is step 4 of AP-8A. RNO can be done locally.

D. Correct – Tripping reactor is highest priority. The RNO of steps of 3- 5 can be done locally. See background of AP-8A for basis for performing only the RNO's.

Technical References: AP-8A background
References to be provided to applicants during exam: None
Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(7)	

Examination Outline Cross-Reference	Level	RO
Knowledge of the operational implications of the following concepts as they apply to the (Reactor Trip or Safety Injection/Rediagnosis): components, capacity, and function of emergency systems.	**Tier #**	1
	Group #	2
	K/A #	W/E01& 02 EK1.1
	Rating	3.1

Question 62

The following plant conditions exist after stabilizing from a small break LOCA:

- RCS Pressure = 1750 psig
- Pressurizer Level = 20% stable
- Subcooling = 45°F
- AFW flow = 500 gpm
- 1 CCP running
- 2 SI pumps running

How will pressurizer level respond if both SI pumps are shut down?

A. Increase

B. Decrease

C. No change

D. Initial sharp decrease, then slowly stabilize

Answer: C

Explanation:

A. Incorrect. Turning off the SI pumps will not result in more flow and subsequent re-pressurization.

B. Incorrect. RCS pressure is greater than shutoff head. Pressure should not change.

C. Correct. RCS pressure is above SI pump shutoff head. When the SI pumps are secured, there will not be an RCS pressure response or temperature response.

D. Incorrect. This is a typical response when pumps are secured during ECCS reduction sequence

Technical References: STG B3
References to be provided to applicants during exam: None

Learning Objective: 6743 Explain PZR response during ECCS reduction sequence

Question Source:	Bank #	X
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(8 & 10)	

Examination Outline Cross-Reference	**Level**	RO
Knowledge of the interrelations between the (Steam generator Overpressure) and the following: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.	**Tier #**	1
	Group #	2
	K/A #	W/E13 EK2.1
	Rating	3.0

Question 63

Unit 1 experienced a Steam Generator Tube Rupture (SGTR).

The crew is at the step in E-3, Steam Generator Tube Rupture, which checks to see if ECCS flow should be terminated.

Which of the following would be the effect on the plant, if Safety Injection Termination is delayed beyond the time assumed in the FSAR analysis for a SGTR?

A. RWST inventory depletion

B. Pressurized Thermal Shock

C. Steam Generator Overfill

D. Reactor Vessel Head bubble formation

Answer: C

Explanation:

A. Incorrect – This is an ECA 1.1, Loss of Emergency Coolant Recirculation, concern.

B. Incorrect – This is a red or orange path FR-P.1, “Response to Imminent Pressurized Thermal Shock” concern.

C. Correct – If SI Flow is not terminated in a timely fashion, leakage into the steam generator will eventually fill the SG with water and potentially lift the SG Safety valves which greatly increases off-site dose

D. Incorrect –This is a yellow path FR-I.1 concern only implemented after SI is terminated.

Technical References: WOG E-3 Background Information
References to be provided to applicants during exam: None
Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(7)	

Examination Outline Cross-Reference	**Level**	RO
Knowledge of the reasons for the following responses as they apply to the (Containment Flooding): RO or SRO function as a team as appropriate to the assigned position, in such a way that procedures are adhered to and the limitations in the facility license and amendments are not violated.	**Tier #**	1
	Group #	2
	K/A #	W/E 15 EK3.4
	Rating	2.9

Question 64

A reactor trip and SI have occurred as a result of a large break LOCA.

E-1.3, Transfer to Cold Leg Recirculation, has just been completed.

The WCSFM reports the following conditions associated with the Containment critical safety function:

- Containment pressure 2.0 psig.
- Containment radiation 1400 R/hr.
- Containment sump level 98 ft.

Which of the following is an immediate containment concern?

A. Containment structural integrity. Go to FR-Z.1, Response to High Containment Pressure.

B. Flooding vital equipment in containment. Go to FR-Z.2, Response to High Containment Flooding.

C. Erroneous instrumentation readings. Go to FR-Z.3, Response to High Containment Radiation.

D. Inadequate suction to the RHR pumps. Go to ECA-1.3, Sump Blockage Guideline.

Answer: B

Explanation:

A. Incorrect – FR-Z.3 is a RED path but containment pressure is <22 psig. Entry condition not met

B. Correct – FR-Z.2 entry condition met and is MAGENTA Path

C. Incorrect – FR-Z.3 is a YELLOW path

D. Incorrect – ECA-1.3 entry condition not met E-1.3 step 13 ACTION/EXPECTED RESPONSE met

Technical References:
References to be provided to applicants during exam: EOP F-0.5
Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(5 and 10)	

Examination Outline Cross-Reference	**Level**	RO
Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.	**Tier #**	1
	Group #	2
	K/A #	W/E16 G2.4.50
	Rating	4.2

Question 65

While observing the containment purge radiation monitor (RM44A) radiation display unit (RDU), you notice that the HIGH ALARM and CVI BYP status lights on the panel are both ON.

Based solely on the indications on the RDU, which of the following describes the containment purge CVI status?

A. Status is normal; high radiation on R-44A will cause a CVI.

B. A CVI signal has been sensed and a CVI has occurred.

C. A CVI signal has NOT been sensed, but the CVI actions will occur when it is sensed.

D. A CVI signal is sensed, but the CVI function is bypassed and it will NOT occur.

Answer: D

Explanation:

A. Incorrect – This a true but inconsistent with the CVI BYP status

B. Incorrect – The CVI will NOT actuate

C. Incorrect – The HIGH ALARM status light means the signal is sensed

D. Correct – CVI BYP is light when key-switch is in Bypass – The CVI will NOT actuate

Technical References: STG G-4B pages 2-43 and 2-44
References to be provided to applicants during exam: None
Learning Objective: 3281 Explain the conditions that effect Digital Radiation Monitoring system radiation monitor indications

Question Source:	Bank #	X
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	No

Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(10)	

Examination Outline Cross-Reference	**Level**	RO
Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup.	**Tier #**	3
	Group #	1
	K/A #	2.1.31
	Rating	4.6

Question 66

Given the following:

- A plant heatup is in progress
- The plant is in MODE 3
- PZR pressure is 1975 psig
- RCS temperature is 525° F

Which of the following describes the status of PK08-06, P-11 Pressurizer SI Permissive?

A. PK ON; PZR Low Pressure SI is enabled.

B. PK ON; PZR Low Pressure SI is blocked.

C. PK OFF; PZR Low Pressure SI is enabled.

D. PK OFF; PZR Low Pressure SI is blocked.

Answer: C

Explanation:

A. Incorrect

B. Incorrect

C. Correct – Pressurizer pressure is above the 1915 psig setpoint, therefore P-11 light on VB1 is OFF and Low Pressure SI is enabled.

D. Incorrect

Technical References: System Lesson Guide B-6A (Reactor Protection System)
References to be provided to applicants during exam: None
Learning Objective: To be determined

Question Source: Bank #

(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41 (5, 7, 8)	

Examination Outline Cross-Reference	Level	RO
Knowledge of conduct of operations requirements.	**Tier #**	3
	Group #	1
	K/A #	2.1.1
	Rating	3.8

Question 67

An event has occurred that will cause a safety injection actuation (SI) setpoint to be exceeded, and automatic actuation is deemed to be unavoidable.

In accordance with OP1.DC10, Conduct of Operations, operators should ________.

A. initiate a reactor trip and at step 4 of E-0, Reactor Trip or Safety Injection, verify SI actuation

B. monitor the plant and verify that the reactor trips and SI actuates automatically

C. inform the Shift Foreman and wait for direction to trip the reactor and initiate SI

D. initiate SI and verify the reactor automatically trips

Answer: D
Per AP1 (step 1) SI is actuated.
Explanation:

A. Incorrect – the operator should actuate the ESF signal.

B. Incorrect - operator should actuate the ESF signal.

C. Incorrect - operator should actuate the ESF signal.

D. Correct – Per OP1.DC10, step 4.6.2 - Licensed Operators are expected to manually initiate Engineered Safety Feature (ESF) actions, (Reactor Trips and Safety Injections), under the following circumstances:T35692
   - When directed by procedure.
   - When a plant parameter is approaching an automatic set point, such that the automatic actuation is judged to be unavoidable.
   - When in the judgment of the Operator or supervisor, initiation of the ESF signal, and use of the EOP set, will better allow for stabilization of the plant and diagnosis of the situation

Technical References: OP1. DC10 page 4
References to be provided to applicants during exam: None
Learning Objective: To be determined

Question Source:	Bank #	#70 2002 Exam
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41 (10)	

Examination Outline Cross-Reference	**Level**	RO
Knowledge of the process for making changes to procedures.	**Tier #**	3
	Group #	2
	K/A #	2.2.6
	Rating	3.0

Question 68

While performing a procedure, it is noted that a minor typographical change is required. The change will not change the intent of the procedure and nuclear safety is NOT affected.

Which of the following is the correct procedure revision process?

A. Editorial correction

B. On the spot change

C. Expedited procedure revision

D. Normal revision

Answer: A

ASSOCIATED INFORMATION:

Associated objective(s):

9735	Explain procedure for making on the spot changes (OTSC)

Reference Id: P-1551

Answer: A

Explanation:

A. Correct - per AD2.ID1, R19, page 20. EC is used to update or correct editorial information that is obviously wrong.

B. Incorrect

C. Incorrect

D. Incorrect

Technical References: OP1.DC10, Conduct of Operations

References to be provided to applicants during exam: None
Learning Objective: To be determined

Question Source:	Bank #	P-1551
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41 (10)	

Examination Outline Cross-Reference	**Level**	RO
Knowledge of tagging and clearance procedures.	**Tier #**	3
	Group #	2
	K/A #	2.2.13
	Rating	4.1

Question 69

GIVEN:

- A turbine building sump pump has been cleared for routine maintenance.
- The clearance has been reported on, and a maintenance red tag has been hung.
- No work has been done to the pump.
- A problem has developed with the other sump pump, making it necessary to place the cleared pump back in service.
- The clearance holder can NOT be located.

Which of the following individuals may remove this red tag?

A. Any Maintenance Foreman familiar with the clearance

B. The Senior Control Operator

C. The Nuclear Operator removing the clearance with the concurrence of the Senior Control Operator

D. The Unit Shift Foreman

Answer: D

Explanation:

A. Incorrect

B. Incorrect

C. Incorrect

D. Correct. It is the reasonability of the SFM and cannot be delegated.

Technical References: OP2. ID2, Tagging Requirements, step 5.2
References to be provided to applicants during exam: None
Learning Objective: To be determined

Question Source:	Bank #	#95 2002 Exam
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41 (10)	

Examination Outline Cross-Reference	Level	RO
Knowledge of the process for managing troubleshooting activities.	**Tier #**	3
	Group #	2
	K/A #	2.2.20
	Rating	2.6

Question 70

In accordance with Operations Policy, B-16, Operations Troubleshooting Activities, whose approval is needed prior to any troubleshooting activities commencing on a unit?

A. Shift Foreman

B. Shift Manager

C. Work Control Lead

D. Operations Manager

Answer: A

Explanation:

A. Correct per B-16, The SFM must approve any preplanned troubleshooting activities.

B. Incorrect

C. Incorrect

D. Incorrect

Technical References: B-16, Operations Troubleshooting Activities
References to be provided to applicants during exam: None
Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41 (10)	

Examination Outline Cross-Reference	**Level**	RO
Knowledge of radiation exposure limits under normal or emergency conditions.	**Tier #**	3
	Group #	3
	K/A #	G2.3.4
	Rating	3.2
		41.12

Question 71

Which of the following describes the 10CFR20 Limits and the Diablo Canyon Administrative Limit for radiation exposure for a calendar year?

	10CFR20 Limit	DCPP Admin Limit
A.	4500 mREM	2000 mREM
B.	4500 mREM	4000 mREM
C.	5000 mREM	2000 mREM
D.	5000 mREM	4500 MREM

Answer: D

Explanation:

A. Incorrect –10CFR20 limit is 5000 mREM and not 4500 mREM. 2000 mREM is the DCPP Admin *Guideline*. 4500 mREM is the Admin **Limit**.

B. Incorrect –.10CFR20 limit is 5000 MREM and not 4500 mREM. 4000 mREM is 90% of DCPP Admin **Limit** (4500 mREM).

C. Incorrect – 10CFR20 limit is 5000 mREM but 2000mREM is the DCPP Admin *Guideline* not the Admin **Limit**.

D. Correct –10CFR20 limit is 5000 mREM and the DCPP Admin **Limit** is 4500 mREM.

Technical References: RP1.ID6, Personnel Dose Limits and Monitoring
References to be provided to applicants during exam: None
Learning Objective: To be determined

Question Source:	Bank #	X
(note changes; attach parent)	Modified Bank #	
	New	

Question History:	Last NRC Exam	No

Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.12	

Examination Outline Cross-Reference	**Level**	RO
Ability to control radiation releases.	**Tier #**	3
	Group #	3
	K/A #	G2.3.11.
	Rating	3.8
		41.11

Question 72

GIVEN:
- Several Auxiliary Building radiation alarms are received
- It is confirmed that Liquid Hold-Up Tank 1-1 has ruptured, and is leaking in the Auxiliary Building
- The crew has entered OP AP-14, Tank Ruptures

What action must be taken to prevent the offsite release of radioactive particulate and iodine?

A. Select "S" signal test, secure one Aux Bldg Ventilation train, and energize charcoal heaters.

B. Push "Status Reset" at POV1 and POV2, and reset the "S" signal.

C. Stop all Aux Bldg supply and exhaust fans, and energize charcoal heaters.

D. Locally close dampers that isolate the Waste Gas Decay Tank rooms.

Answer: A

Explanation:

A. Correct. AP-14, step 2 provides direction for placing ABVS in SFGDS Only. This isolates ventilation to general aux Bldg and ensures all other exhaust is filter by the Iodine Removal filter prior to exhaust to atmosphere.

B. Incorrect. This will reset ventilation logic. System will be reposition to normal based on selected mode and selected equipment. This does not minimize release to environment.

C. Incorrect. This will not provide for Iodine removal. Exhaust Fan E-1 needs to be running.

D. Incorrect. This will isolate ventilation supply to Liquid Hold up tanks but will not provide for Iodine removal.

Technical References: OP-AP-14, LPA-14 page 6, STG H-1
References to be provided to applicants during exam: None
Learning Objective:

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.11	

Examination Outline Cross-Reference	Level	RO
Ability to comply with radiation work permit requirements during normal or abnormal conditions.	**Tier #**	3
	Group #	3
	K/A #	2.3.7
	Rating	3.5

Question 73

Both units are at full power.

What type of Radiation Work Permit (RWP) would the operator sign on to for an oil addition to a Reactor Coolant Pump?

A. Routine RWP.

B. Job Specific RWP.

C. Special WP.

D. Planned Special Exposure.

Answer: B

Explanation:

A. Incorrect – issued for up to a year for considered routine in stable radiological conditions. Not to be used for entry into areas where general area dose rates exceed 1000 mr/hr.

B. Correct. Issued for specific non routine work. Job specific RWPs such as adding oil to RCPs may be authorized for extended periods of time.

C. Incorrect – no longer used.

D. Incorrect – used for emergency exposures.

Technical References: RCP D-201, Writing Radiation Work Requests
References to be provided to applicants during exam: None
Learning Objective: To be determined

Question Source: Bank #
(note changes; attach parent) Modified Bank #

	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41 (12)	

Examination Outline Cross-Reference	**Level**	RO
Ability to identify post accident instrumentation.	**Tier #**	3
	Group #	4
	K/A #	2.4.3
	Rating	3.7

Question 74

Which of the following is monitored on PAM1?

A. Wide Range Containment Sump Water Level

B. Pressurizer Level

C. Steam Line radiation monitors

D. Auxiliary Feedwater Flow

Answer: A

Explanation:

A. Correct – Wide Range Containment Sump Water Level is monitored by PAM1

B. Incorrect – Not monitored by PAMS

C. Incorrect – Monitored separate rad monitor panel

D. Incorrect – Not monitored by PAMS

Technical References: System Training Guide, B-10, Post-Accident Monitoring System
References to be provided to applicants during exam: None
Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No

Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41 (7)	

Examination Outline Cross-Reference	**Level**	RO
Knowledge of EOP layout, symbols, and icons.	**Tier #**	3
	Group #	4
	K/A #	2.4.19
	Rating	3.4

Question 75

When proceeding through Emergency Operating Procedures, if the reader comes across a step number enclosed in a box (e.g. $\boxed{1.}$ step instruction…), this denotes to the reader…

A. Continuous Action Step

B. Refer to Foldout Page

C. Refer to the Note/Caution of that step

D. Immediate Action Step

Go to e-0 and see if it can be expanded like the example above)
Answer: D

Explanation:

A. Incorrect – Continuous Action Steps are bordered by a box around the entire step.

B. Incorrect – Instructions within the step will refer the reader to the Foldout Page.

C. Incorrect – Notes and Cautions are identified by their descriptors, NOTE or CAUTION, with the text extending across the entire page.

D. Correct

Technical References: LPE-Rule (EOP Rules of Usage)
References to be provided to applicants during exam: None
Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No

Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41 (10)	

Examination Outline Cross-Reference	**Level**	SRO
Ability to Interpret and execute procedure steps.	**Tier #**	1
	Group #	1
	K/A #	009 G2.1.20
	Rating	4.6

Question 76

Following a small break LOCA on Unit 2 the following conditions exist:

- The crew is performing the actions of E-1, Loss of Reactor or Secondary Coolant
- RCS temperature 525° F
- RWST level 32%
- RHR pumps are stopped by the operator
- RCS pressure 290 psig
- Hydrogen Concentration 0.4%

Which of the following procedures provides the required actions that mitigate these plant conditions?

A. EOP E-1.4, Transfer to Hot Leg Recirculation

B. EOP E-1.2, Post LOCA Cooldown and Depressurization

C. OP H-9, Inside Containment Hydrogen Recombination System

D. EOP E-1.3, Transfer to Cold Leg Recirculation

Answer: D

Explanation:

A. Incorrect – Transfer to Hot Leg Recirculation occurs at step 20 of EOP E-1, but due to the current conditions, this procedure is exited at step 13 to Cold Leg Recirculation with RWST level less than 33% and RHR flow initially greater than 100 GPM before being stopped by the operator.

B. Incorrect – Entrance into E-1.2, Post LOCA Cooldown and Depressurization, does not occur due to RCS pressure being less than 300 PSIG in step 12 of E-1.

C. Incorrect – Entrance into OP H-9 would occur if Hydrogen concentration was greater than .5%.

D. Correct – Transfer to Cold Leg Recirculation occurs in step 13 of EOP E-1 with

RWST level less than 33% and RHR flow greater than 100 GPM before being stopped by the operator.

Technical References: EOP E-1, Loss of Reactor or Secondary Coolant
References to be provided to applicants during exam: None
Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43 (5)	

Examination Outline Cross-Reference	**Level**	SRO
Ability to determine and interpret the following as they apply to the Loss of Residual Heat Removal System: Leakage of reactor coolant from RHR into closed cooling water system or into reactor building atmosphere.	**Tier #**	1
	Group #	1
	K/A #	025 AA2.02
	Rating	3.8

Question 77

GIVEN:

- Unit 1 is performing a core offload
- Both trains of RHR are in service

PK11-21, High Radiation alarms. A few minutes later, PK02-16, RHR System and PK02-17, RHR Pumps, also goes into alarm.

Which of the following procedures should the SFM utilize to address the current plant conditions?

A. AP-1, Excessive Reactor Coolant System Leakage

B. AP SD-2, Loss of RCS Inventory

C. AP-16, Malfunction of the RHR System

D. AP-24, Shutdown LOCA

Answer: B

Explanation:

A. Incorrect – Only appropriate in MODES 1-4

B. Correct

C. Incorrect – Not applicable in MODE 6 or if there is a loss of RCS inventory

D. Incorrect – Not applicable in MODE 6

Technical References: OP AP-24, OP AP SD-2, OP AP-1, OP AP-16, PK02-16, PK02-17, PK11-21
References to be provided to applicants during exam: None
Learning Objective: To be determined

Question Source:	Bank #	#78 2005 Exam
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43 (5)	

Examination Outline Cross-Reference	**Level**	SRO
Knowledge of the purpose and function of major system components and controls.	**Tier #**	1
	Group #	1
	K/A #	026 G2.1.28
	Rating	4.1

Question 78

Given the following conditions:

- Unit 1 reactor is at 100% power in MODE 1
- A 200 gpm CCW leak occurs
- CCW makeup is not available

Which of the following describes the minimum time the surge tank is designed to provide system make-up based and what the time is based on?

A. 40 minutes; for operators to locate and isolate the leak before the system becomes impaired due to water loss.

B. 20 minutes; for operators to locate and isolate the leak before the system becomes impaired due to water loss.

C. 40 minutes; to cross-tie the units.

D. 20 minutes; to cross-tie the units.

Answer: B
Reworded question to focus on one theme
Explanation:

A. Incorrect

B. Correct - Tech Spec Bases 3.7.7 states that 20 minutes based on a non-mechanistic leakage rate of 200 gpm, for operators to locate and isolate the leak or realign the CCW system into two separate vital loops before the system becomes impaired due to water loss.

C. Incorrect

D. Incorrect –

Technical References: T.S. 3.7.7, Bases 3.7.7, System Lesson Guide LF-1 (Component Cooling Water)
References to be provided to applicants during exam: None
Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43 (2)	

Examination Outline Cross-Reference	Level	SRO
Ability to determine or interpret the following as they apply to a Station Blackout: Actions necessary to restore power	**Tier #**	1
	Group #	1
	K/A #	055 EA2.03
	Rating	4.7

Question 79

Current Plant Conditions:

- The crew is performing the actions of ECA-0.0, Loss of All Vital AC Power
- No ECCS pumps are running
- No RCPs are running
- RCS subcooling is 15°F
- Pressurizer level is 11% and slowly decreasing
- RCS pressure is 2050 psig and slowly decreasing

The crew is implementing ECA-0.3, Restore 4kV Buses, to restore power to a vital 4 kV bus as directed by ECA-0.0.

Which of the following actions will be taken by the crew in order to load vital equipment and stabilize the plant once power is restored to a vital 4 kV bus?

A. Transition to ECA-0.1 (Loss of All AC Power Without SI Required).

B. Continue in ECA-0.0 and continue to implement EOP ECA-0.3 without transitioning.

C. Transition to ECA-0.2 (Loss of All AC Power With SI Required).

D. Return to E-0 (Reactor Trip or Safety Injection).

Answer: C
Lowered subcooling and pressurizer level, a little. Reworded setup.
Explanation:

A. Incorrect – Transitioning to ECA-0.1 would occur in step 29 of ECA-0.0 if RCS subcooling were greater than 20°F, Pressurizer level were greater than 12% and no ECCS pumps were running. As current conditions indicate that RCS subcooling is less than 20°F, the correct transition would be to ECA-0.2

B. Incorrect

C. Correct – Transition to ECA-0.2 from step 29 in ECA-0.0 with RCS subcooling less than 20°F.

D. Incorrect

Technical References: ECA-0.0, ECA-0.1, ECA-0.2, ECA-0.3, EOP E-0
References to be provided to applicants during exam: None
Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43 (5)	

Examination Outline Cross-Reference	Level	SRO
Loss of offsite power: Knowledge of abnormal condition procedures.	**Tier #**	1
	Group #	1
	K/A #	056 2.4.11
	Rating	4.2

Question 80

GIVEN:
- The RCS temperature is 235°F
- L-1, Plant Heatup from Cold Shutdown to Hot Standby, is in progress

A loss of offsite power (230 kV and 500 kV) occurs.

What procedure will the Shift Foreman enter?

A. OP SD-1, Loss of AC Power

B. OP AP-26, Loss of Offsite Power

C. ECA-0.0, Loss of All AC Power

D. OP SD-5, Loss of Residual Heat Removal

Answer: B

Explanation:

A. Incorrect – used in MODE 5, to restore power to vital buses. The vital buses are energized.

B. Correct – Used to restore power to non-vital 4 kV and/or 12 kV buses. Procedure also will have the RHR pump restarted

C. Incorrect – does not apply in MODE 5, or if vital buses are energized.

D. Correct – used in MODE 5 to restore power to decay heat removal.

Technical References: OP AP SD-5
References to be provided to applicants during exam: None
Learning Objective: 3478 - Given initial conditions, assumptions, and symptoms, determine the correct abnormal operating procedure to be used to mitigate an operational event

Question Source:	Bank #	X – P-45897
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43 (5)	

Examination Outline Cross-Reference	**Level**	SRO
Ability to determine and interpret the following as they apply to the Loss of vital AC Instrument Bus: That substitute power sources have come on line on a loss of initial AC.	**Tier #**	1
	Group #	1
	K/A #	057 AA2.14
	Rating	3.6

Question 81

Unit 1 is at full power.

AR PK19-19, UPS Failure, is now alarming due to inputs 1503, Instr AC UPS 1-4 Inverter Failure, and 1505, Instr AC UPS 1-4 on Bypass. 120 VAC Bus 1-4 is energized from its alternate source.

Which of the following describes the status of inverter 1-4 and the vital AC instrument bus 14?

A. Both the inverter and the vital AC instrument bus are OPERABLE.

B. The inverter is OPERABLE; the vital AC instrument bus is de-energized and inoperable.

C. The inverter is inoperable; the vital AC instrument bus is OPERABLE.

D. Both the inverter and the vital AC instrument bus are inoperable.

Answer: C

Explanation:

A. Incorrect – The inverter will be OPERABLE when it is powered from its normal source or from the DC source.

B. Incorrect – The inverter has transferred to its backup supply (input 1505), therefore the vital AC instrument bus is OPERABLE.

C. Correct – The vital instrument bus is powered from a class 1E CVT (backup supply) and OPERABLE. The inverter is inoperable.

D. Incorrect – The vital AC bus is currently energized.

Technical References: T.S. 3.8.7 and T.S. Bases 3.8.7, AR PK19-19
References to be provided to applicants during exam: T.S. 3.8.7, TS 3.8.9 and T.S. Bases

3.8.9
Learning Objective: To be determined

Question Source:	Bank #	#81 2005 Exam
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43 (2)	

Examination Outline Cross-Reference	Level	SRO
Ability to determine and interrupt the following as they apply to the Pressurizer Control Malfunctions: leak in PZR.	**Tier #**	1
	Group #	2
	K/A #	028 AA2.11
	Rating	3.6

Question 82

The plant was stable at full power with all systems operating normally in automatic control when the following occurs:

- Pressurizer level increased slightly but is now decreasing
- Pressurizer pressure is 2220 psig and decreasing
- Pressurizer backup heaters energized
- Charging flow is increasing
- PRT pressure is stable
- PK01-16, Containment Environment PPC alarms
- Containment Temperature is rising

Which of the following describes the procedure the Shift Foreman will use for the given conditions?

A. OP AP-1, Excessive Reactor Coolant System Leakage, due to a Pressurizer vapor space leak.

B. OP AP-5, Malfunction of Protection or Control Channel, due to a Pressurizer pressure channel failure.

C. OP AP-13, Malfunction of Reactor Pressure Control System, due to a malfunction of the Pressurizer pressure controller.

D. OP AP-1, Excessive Reactor Coolant System Leakage, due to a leaking PORV or Pressurizer safety.

Answer: A

Explanation:

A. Correct –Pressure Vapor Space leak has occured

B. Incorrect – Elevated Containment Temperature is inconsistent with a Pzr instrument level or pressure instrument failure.

C. Incorrect – Pressure response is as expected.

D. Incorrect – Leaking PORV/safety would not cause containment temperature to increase at this time.

Technical References: ARP 01-16, Simulator Malfunction MALFPZR, OIM-A-4-2B
References to be provided to applicants during exam: None
Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43(5 and 13)	

Examination Outline Cross-Reference	**Level**	SRO
Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.	**Tier #**	1
	Group #	2
	K/A #	051 G 2.1.7
	Rating	4.7

Question 83

A turbine load increase is in progress with all systems operating normally in automatic control when the following occurs:

- Turbine Load is 600 MWE.
- Pressurizer pressure is 2235 psig and stable
- Tave/Tref is matched.
- Condenser Pressure PI-44 reads 4.0"Hg Abs.
- Condenser Pressure Recorder PR-11A and B both show condenser pressure is slowly rising.
- Condenser differential pressure is 7 psid and stable on all quadrants
- PK10-11 COND PRESS/LEVEL is in ALARM

Which of the following describes the action that will be taken by the Shift Foreman?

A. Direct the operator to trip the reactor, then the turbine and go to E-0, Reactor Trip or Safety Injection.

B. Direct the operator to trip the turbine and go to AP-29, Main Turbine Malfunction.

C. Go to OP AP-7, Degraded Condenser, section A (Loss of Condenser Vacuum), and reduce load increase necessary to restore condenser pressure to within operating limits.

D. Go to OP AP-7, Degraded Condenser, section B (Condenser Fouling) and reduce load to remove a Circulating Water pump from operation to lower condenser differential pressure.

Answer: C

Explanation:

A. Incorrect. No indication that a turbine trip required at this time.

B. Incorrect. Condenser pressure is within acceptable operating region of Op AP-7 Attachment 6.2.

C. Correct. Normal condenser parameter should re be restored.

D. Incorrect. There are no abnormal CWP indications or indications of tube fouling.

Technical References: OP-L-3 PK10-11 OP-AP-7
References to be provided to applicants during exam: None
Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(7)	

Examination Outline Cross-Reference	**Level**	SRO
Ability to operate and/or monitor the following as they apply to the dropped rod: Rod position indication to actual rod position	**Tier #**	1
	Group #	2
	K/A #	003 AA2.01
	Rating	3.9

Question 84

The plant is at full power, all rods out, when the following events occur:

- DRPI rod bottom light for a Control Bank D rod lights
- PK03-21, DRPI Failure/Rod Bottom, alarms
- PK03-17, Rod Cont Urgent Failure, alarms
- PK03-13, Rod Lo Insertion Limit Alarm and PK03-14, Rod Lo Lo Insertion Limit Alarm, alarm
- Reactor power and Tave decrease then begin to stabilize
- Group step counters for Control Bank D read 231 steps

Which of the following actions will be taken by the Shift Foreman?

A. Direct a reactor trip and go to E-0, Reactor Trip or Safety Injection.

B. Contact maintenance for troubleshooting a DRPI failure.

C. Go to OP AP-12B, Control Rod Misalignment.

D. Go to OP AP-12C, Dropped Control Rod.

Answer: D

Explanation:

A. Incorrect. Only 1 rod has dropped, no call for reactor trip at this time.

B. Incorrect. DRPI is functioning properly. Group step counters will not change and the urgent failure could be an expected alarm.

C. Incorrect. With the rod bottom light lit, the appropriate section is AP-12C

D. Correct. A single rod has dropped.

Technical References: OP AP-12C
References to be provided to applicants during exam: None
Learning Objective: 3477 Given an abnormal condition, summarize the major actions of the abnormal operating procedure to mitigate an event in progress.
5024 Explain the effect of dropped rod(s) on reactor operation

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43(5 and 13)	

Examination Outline Cross-Reference	**Level**	SRO
Knowledge of the limiting conditions for operation and safety limits	**Tier #**	1
	Group #	2
	K/A #	WE08 G2.2.22
	Rating	4.7

Question 85

The plant is at 3% power when a steam break occurs outside containment.

As the crew reaches the diagnostic steps of E-0 Rector Trip or Safety Injection the following plant conditions exist:

- RCS temperature 280°F and rising after an initial decrease to 230°F
- RCS pressure 1750 psig and rising
- Pressurizer level 15% and rising
- Steam Generator pressures:

S/G 2-1	820 psig, rising
S/G 2-2	0 psig, stable
S/G 2-3	900 psig, rising
S/G 2-4	890 psig, rising

Which of the following procedures will the crew transition to from E-0?

A. E-1.1 Safety Injection Termination.

B. E-2 Faulted Steam Generator Isolation

C. ECA-2.1 Uncontrolled Depressurization of All Steam Generators

D. FR-P.1 Response to Imminent Pressurized Thermal Shock Condition

Answer: B

Explanation:

A. Incorrect. Transition to E-1.1 will occur at E-2 Step 8.

B. Correct. Transition to E-2 is required at E-0 step 11

C. Incorrect. Transition to ECA-2.1 could occur at E-2 step 2 but condition is not met.

D. Incorrect. Magenta FR-P.1 condition initially met but is now Yellow. Per EOP F-0 section 4.1.1, operator is not required to perform action of FR-P.1 since adequate time was not allowed for thermal stresses to affect the integrity of the vessel wall.

Technical References:
References to be provided to applicants during exam: None
Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	X
	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(5) 43(2) and 45(2)	

Examination Outline Cross-Reference	**Level**	SRO
Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.	**Tier #**	2
	Group #	1
	K/A #	007 G2.2.44
	Rating	4.2

Question 86

The plant is at full power.

The following events occur:
- PK05-25, PRT Press/Lvl/Temp alarms, input 545 – Pzr Relief Tk Press Hi and Vent Hdr Isol. alarms
- PRT temperature is 110°F and rising at approximately 1°F every 5 minutes
- PRT pressure is 10 psig and rising at approximately 1 psig every 5 minutes
- AR PK05-25 has been entered by the Shift Foreman

What action will be taken by the Shift Foreman?

A. Implement AP-1, Excessive RCS Leakage to aid in identifying the source and magnitude of the leakage.

B. Direct the operator to trip the reactor and enter E-0, Reactor Trip or Safety Injection due to an unidentified loss of reactor coolant.

C. Per AR PK05-25, contact maintenance to troubleshoot a possible failure of the PRT N_2 Supply regulator, RCS-1-PCV-3035.

D. Per AR PK05-25, vent the PRT to the Waste Gas Header to less than 3 psig using PCV-472, PRT Vent to Vent Hdr, and verify RCS-1-8045, PRT N_2 Supply Isolation valve is closed.

Answer: A. Implement AP-1, Excessive RCS Leakage to aid in identifying the source and magnitude of the leakage.

Explanation:
Answer A correct – increase in pressure with a corresponding increase in level/temperature is an indication of leakage into the PRT. AP-1 is implemented to aid in finding the leakage.

Answer B incorrect – there is no indication that a reactor trip is required at this time.

Answer C incorrect – a failure of the regulator is suspected if only pressure is increasing (no corresponding increase in level or temperature)

Answer D incorrect – PCV-472 cannot be used when PRT pressure is 10 psig or higher.
Technical References: AR PK05-25
References to be provided to applicants during exam: None
Learning Objective: 3552 - Given initial conditions, assumptions, and symptoms, determine the correct abnormal operating procedure to be used to mitigate an operational event

Question Source:	Bank #	X
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	2009 DCPP #87
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43(5)	

Examination Outline Cross-Reference	**Level**	SRO
Ability to (a) predict the impacts of the following malfunctions or operations on the CSS; and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operation: Failure of spray pump	**Tier #**	2
	Group #	1
	K/A #	026 A2.04
	Rating	4.2

Question 87

A large break LOCA occurred, resulting in a reactor trip and safety injection.

EOP E-1, "Loss of Reactor Coolant or Secondary Coolant" is in progress.

The following conditions exist:

- RCS Pressure is 60 psig and trending down slowly
- RCS Temperature is 300°F and trending down slowly
- RWST level is 50% and trending down
- RHR flow is 650 gpm
- Containment pressure is 21 psig and trending down slowly
- PK01-18 Containment Spray Actuation is ON
- Both trains of Containment Spray are in service

Containment Spray Pump 1-1 trips and cannot be manually started.

Based on the above, the Shift Forman will:

A. Continue in E-1, "Loss of Reactor Coolant or Secondary Coolant."

B. GO TO E-1.3, "Transfer to Cold Leg Recirculation."

C. GO TO ECA-1.1, "Loss of Emergency Coolant Recirculation."

D. GO TO FR-Z.1, "Response to High Containment Pressure."

Answer: A

Explanation:

A. Correct. Crew has transitioned out of E-O to E-1 already. Status Monitoring is in effect per EOP F-0 background document.

B. Incorrect. E-1.3 would be implemented if RWST level is < 33%. RWST level is above this transition criteria.

C. Incorrect. ECA-1.1 Entry condition from E-1 Step 11 NOT met since RHR is available and running.

D. Incorrect. Yellow PATH on Containment should be implemented per EOP F-0 rule of Usage 3.3.

Technical References:
References to be provided to applicants during exam: None
Learning Objective:

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	X
	New	
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(5) 43(5)	
	55.45(3 & 13)	

Examination Outline Cross-Reference	**Level**	SRO
Ability to prioritize and interpret the significance of each annunciator or alarm.	**Tier #**	2
	Group #	1
	K/A #	005G2.4.45
	Rating	4.3

Question 88

GIVEN:

- The crew is performing E-1, Loss of Reactor or Secondary Coolant
- RCS pressure is 200 psig
- Readings on RM-13, RHR Exhaust Duct, radiation monitor are increasing
- RWST level is 40% and decreasing
- RHR pump amps are fluctuating
- The following alarms are received in the Control Room:

PK02-16	RHR SYSTEM
PK02-17	RHR PUMPS

Which of the following procedures will the Shift Foreman transition to from E-1?

A. E-1.2, Post-LOCA Cooldown and Depressurization

B. ECA-1.1, Loss of Emergency Coolant Recirculation

C. ECA-1.2, LOCA Outside Containment

D. ECA-1.3, Sump Blockage Guideline

Answer: C

Explanation:

A. Incorrect – RCS pressure is below 300 psig.

B. Incorrect – this is entered if both trains are unable to be aligned to the containment sump.

C. Correct – increasing radiation levels, coupled with the RHR alarms indicate a LOCA outside containment.

D. Incorrect – the sump is not blocked.

Technical References: E-1, steps 11 - 13
References to be provided to applicants during exam: None
Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(10), 43(5) and 55.45(3)	

Examination Outline Cross-Reference	Level	SRO
Ability to (a) predict the impacts of the following malfunctions or operations on the DC electrical systems; and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operation: Loss of ventilation during battery charging.	**Tier #**	2
	Group #	1
	K/A #	063 A2.02
	Rating	3.1

Question 89

Unit 1 is at 100% power.

Maintenance Service requests to work on both Auxiliary Building Switchgear Ventilation System supply and exhaust fans (S-27 and E-27)

What is the operational concern and the action the SFM should take prior to approving the maintenance request?

A. Reduced Battery Capacity; Implement compensatory measures for blocked open doors per ECG80.1, Doors Required for HELB, HVAC ECCS Function or Flood Protection.

B. Reduced Battery Capacity; Declare Batteries Inoperable per Tech Spec 3.8.4, DC Sources - Operating

C. Elevated Hydrogen concentration in the Vital Battery Rooms; Implement compensatory measures for blocked open doors per ECG80.1, Doors Required for HELB, HVAC ECCS Function or Flood Protection.

D. Elevated Hydrogen concentration in the Vital Battery Rooms; Declare Batteries Inoperable per Tech Spec 3.8.4, DC Sources - Operating

Answer: C

Explanation:

A. Incorrect - Battery Capacity will not be diminished but ECG is correct.

B. Incorrect – Battery Capacity will not be diminished, T/S is incorrect

C. Correct – Hydrogen concentration will increase in battery rooms without adequate ventilation, blocked open doors are controlled by ECG 80.1.

D. Incorrect – Hydrogen concentration will increase in battery rooms without adequate ventilation, but T/S is incorrect.

Technical References:

References to be provided to applicants during exam: None
Learning Objective: To be determined

Question Source:	Bank #	X
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(5), 55.43(5) and 55.45(3 & 13)	

Examination Outline Cross-Reference	**Level**	SRO
Ability to interpret control room indications to verify the status and operations of a system, and understand how operator actions and directives affect plant and system conditions.	**Tier #**	2
	Group #	1
	K/A #	003 G2.2.44
	Rating	4.4

Question 90

Unit 1 is carrying out actions of an Emergency Operating Procedure, and all RCPs are off.

The crew is making preparations to start a RCP using Attachment B, Restart of Reactor Coolant Pump. Seal cooling was lost to RCP No. 2.

The following RCP conditions exist:

	RCP No. 1	RCP No. 2	RCP No. 3	RCP No. 4
Seal D/P	Slightly low	SAT	SAT	SAT
Seal Injection Flow	SAT	Slightly high	Slightly low	SAT
Seal Leak-off	SAT	SAT	SAT	Slightly high

What direction would the SFM give to the operator that is starting the RCP?

A. Start RCP No. 1 after lowering VCT pressure to restore Seal D/P.

B. Start RCP No. 2 after lowering VCT pressure to restore Seal Injection Flow.

C. Start RCP No. 3 after lowering VCT pressure to restore Seal Injection Flow.

D. Start RCP No. 4 after lowering VCT pressure to restore Seal Leak-off.

Answer: A

Explanation:

A. Correct – Pump is second in order of preference for normal PZR Spray.

B. Incorrect – Preferred pump but RCP seal cooling was lost and should not be started WITHOUT an RCP Seal Status Evaluation.

C. Incorrect – Pump is behind RCP No 1 and No.2 in starting preference.

D. Incorrect – lowering VCT pressure will raise (not lower) seal leak off flow to normal.

Technical References: EOP FR-C.1 Appendix B
References to be provided to applicants during exam: None

Learning Objective:

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	X
	New	
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(5), 55.43(5) and 55.45(12)	

Examination Outline Cross-Reference	**Level**	SRO
Ability to (a) predict the impacts of the following malfunctions or operations on the NIS: and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Power supply loss or erratic operation.	**Tier #**	2
	Group #	2
	K/A #	015 A2.01
	Rating	3.9

Question 91

A plant startup is in progress. Power is 3%.

The operator reports all Channel I bistable lights are out and Channel IV bistable lights are lit.

Which of the following actions will be taken by the Shift Foreman?

A. Enter E-0, Reactor Trip or Safety Injection due to an Intermediate Range high flux trip.

B. Go to OP AP-4, Loss of Vital or Non-vital Instrument AC, due to the loss of PY-11.

C. Enter E-0, Reactor Trip or Safety Injection due to the loss of the Source Range block signal, causing the Source Ranges to energize and cause a trip on high flux.

D. Go to OP AP-4, Loss of Vital or Non-vital Instrument AC, due to the loss of PY-14.

Answer: D.

Explanation:

- **A. Incorrect –** no IR channels off PY-14. Powered from PY-11 and 12.
- **B. Incorrect** – bistable lights on Channel I are out because of a loss of power from Channel IV. PY-11 is still powered.
- **C. Incorrect** – PY-14 loss will cause Source Ranges to energize, however, the trip is blocked.
- **D. Correct**- PY-14 powers PY-11 bistable lights, causes PR N44 to fail low, however, no trip occurs. The appropriate action is to go to AP-4 and take action to re-energize the PY.

Technical References: OP AP-4, LPA-4
References to be provided to applicants during exam: None
Learning Objective: 4274 - Explain the consequences of loss of vital instrument bus.

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43 (5)	

Examination Outline Cross-Reference	**Level**	SRO
Malfunctions or operations on the HRPS; and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations: LOCA condition and related concern over hydrogen	**Tier #**	2
	Group #	2
	K/A #	028 A2.02
	Rating	3.9

Question 92

GIVEN:

- A large break LOCA occurred 2 hours ago
- Hydrogen concentration inside Containment as measured on PAM 1 is 1.6%
- EOP E-1, Loss of Reactor or Secondary Coolant, is being performed

Which of the following supplemental procedures to E-1 will be used to place the Containment Hydrogen Recombination System in service and who is responsible for directing this system be placed in service?

NOTE:
OP H-8, Containment Hydrogen Purge System
OP H-9, Inside Containment Hydrogen Recombination System

A. OP H-8; Shift Foreman

B. OP H-9; TSC (Site Emergency Coordinator)

C. OP H-8; TSC (Site Emergency Coordinator)

D. OP H-9; Shift Foreman

Answer: B

Explanation:

A. Incorrect

B. Correct – EOP E-1 step 19 directs operators to OP H-9 if hydrogen concentration is more than .5%. OP H-9 states that the IHRS is ONLY placed in service upon direction of the TSC and the Site Emergency Coordinator.

C. Incorrect

D. Incorrect

Technical References: EOP E-1 (Loss of Reactor or Secondary Coolant), OP H-9 (Inside Containment H2 Recombination System)
References to be provided to applicants during exam: None
Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43 (5)	

Examination Outline Cross-Reference	**Level**	SRO
Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.	**Tier #**	2
	Group #	2
	K/A #	068 2.2.36
	Rating	4.2

Question 93

The liquid radwaste effluent radiation monitor is to be removed from service for calibration. No discharges are in progress.

Which of the following describe the action, if any, that will be taken by the crew?

A. Declare the monitor inoperable when it is removed from service.

B. No action is required because there are no releases planned or in progress.

C. No action required unless the monitor fails its calibration, then it must be declared inoperable.

D. No action is required because ECG 39.3, NOTE A allows releases to occur for 14 days before having to declare the monitor inoperable and applying the Required Actions of Condition A.

Answer: A

Explanation:
This question is designed to test a problem that occurred at the plant. A gaseous rad monitor was removed from service multiple times without being declared inoperable because of the mistaken understanding that if there was no release planned/in progress the ECG did not apply. (Notification 50254364)

A. Correct – the 14 day clock starts when the monitor is removed from service, regardless of the status of any planned release.

B. Incorrect – the monitor is inoperable immediately.

C. Incorrect – the monitor is inoperable when removed from service.

D. Incorrect- its possible to read the note and think that as long as samples are processed correctly for 14 days there is no further action, and the 14 completion time of Condition A would start after the 14 days of Note A.

Technical References: ECG 39.3 pages 1 and 4, notification 50254364 (07/2009)
References to be provided to applicants during exam: ECG 39.3

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43 (2)	

Examination Outline Cross-Reference	**Level**	SRO
Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc.	**Tier #**	3
	Group #	1
	K/A #	2.1.5
	Rating	3.9

Question 94

Unit 1 is in MODE 3 and Unit 2 is in MODE 4.

The Shift Foreman is developing the list of qualified personnel for the upcoming shift.

Are a Health Physics Technician and Shift Technical Advisor required for the current plant conditions?

	Health Physics Technician	Shift Technical Advisor
A.	No	Yes
B.	Yes	No
C.	Yes	Yes
D.	No	No

Answer: C

Explanation:

A. Incorrect

B. Incorrect

C. Correct – T.S. 5.2.2 (Unit Staff) requires a Health Physics Technician any time that fuel is in the core, which meets the current plant condition. A Shift Technical Advisor is required in MODES 1, 2, 3, and 4 only.

D. Incorrect

Technical References: T.S. 5.2.2
References to be provided to applicants during exam: None
Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43 (2)	

Examination Outline Cross-Reference	**Level**	SRO
Ability to make accurate, clear, and concise verbal reports.	**Tier #**	3
	Group #	1
	K/A #	2.1.17
	Rating	4.0

Question 95

If the Shift Foreman has deviated from a license condition or Technical Specification in order to protect public health and safety in the event of an emergency, what would be the time requirement to report such the deviation to the NRC?

A. 15 minutes

B. 30 minutes

C. 1 hour

D. 4 hours

Answer: C

Explanation:

A. Incorrect

B. Incorrect

C. Correct – 10CFR50.72(5)(b) directs a licensee to report within one hour if that licensee has departed from a license condition or technical specification pursuant to 10CFR50.54(x).

D. Incorrect

Technical References: 10CFR50.72(5)(b)
References to be provided to applicants during exam: None
Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No

Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43 (5)	

Examination Outline Cross-Reference	**Level**	SRO
Knowledge of the process for controlling equipment configuration or status.	**Tier #**	3
	Group #	2
	K/A #	2.2.14
	Rating	4.3

Question 96

Prior to the Shift Foreman approving an Emergent Temporary Modification (TMOD) which of the following must occur?

A. A LBIE must be performed

B. Engineering approval obtained

C. An Operability Determination must be performed

D. Engineering Director (or designee) approval obtained

Answer: A

Explanation:

A. Correct – per step 5.2.4, the Emergent TMOD process is used when requested by the SFM. The process requires the SFM document why the TMOD is required prior to engineering evaluation (attachment 4). An LBIE is performed prior to approval

B. Incorrect – engineering reviews as soon as possible.

C. Incorrect – Oper. Determination not applicable to TMODs.

D. Incorrect – Engineering director responsible for reviewing and obtaining approval for any TMOD to remain installed past a refueling outage.

Technical References: CF4.ID7 pages 20, attachment 2 and 4
References to be provided to applicants during exam: None
Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No

Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43 (3)	

Examination Outline Cross-Reference	**Level**	SRO
Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.	**Tier #**	3
	Group #	2
	K/A #	2.2.25
	Rating	4.2

Question 97

Which of the following is the reason the minimum CST level required to satisfy Technical Specification 3.7.6, Condensate Storage Tank, is higher for one unit than it is for the other unit?

A. Unit 1 requires a higher level because the assumed natural circulation cooldown rate to 350°F is half of the cooldown rate assumed for Unit 2.

B. Unit 2 requires a higher level because the assumed natural circulation cooldown rate to 350°F is half of the cooldown rate assumed for Unit 1.

C. Unit 1 requires a higher level because the assumed natural circulation cooldown rate to less than 200°F is half of the cooldown rate assumed for Unit 2.

D. Unit 2 requires a higher level because the assumed natural circulation cooldown rate to 200°F is half of the cooldown rate assumed for Unit 1.

Answer: A

Explanation:

A. Correct – Bases for 3.7.6 states: For Unit 1 with a Thot upper head design, the analysis for CST minimum required storage assumes the unit is held in MODE 3 for 1 hour followed by an 8-hour cooldown to RHR entry conditions at a reduced cooldown rate of 25°F/hour. For Unit 2 with a Tcold upper head design, the analysis for CST minimum required storage assumes the unit is held in MODE 3 for 2 hours followed by a 4-hour cooldown to RHR entry conditions at a cooldown rate of 50°F/hour

B. Incorrect – The opposite is true.

C. Incorrect – cooldown is only until RHR can be placed in service.

D. Incorrect – cooldown only until RHR and higher for Unit 1.

Technical References: T.S. 3.7.6. B3.7.6
References to be provided to applicants during exam: None

Learning Objective: 9694G, Discuss 3.7 Technical Specification bases

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43 (2)	

Examination Outline Cross-Reference	**Level**	SRO
Ability to approve release permits.	**Tier #**	3
	Group #	3
	K/A #	2.3.6
	Rating	3.8

Question 98

A discharge of Gas Decay Tank 1-2 is planned.

GIVEN:

- Current time and date - 2200, 17 July
- RE-22 declared inoperable at 0100 on 3 July
- The planned discharge will take 4 hours
- 2 samples have been independently drawn and analyzed
- Release rate calculations have been independently verified

Which of the following describes whether the planned discharge may or may not occur?

A. The planned discharge may proceed in its entirety.

B. The discharge may occur, but only for 3 hours, then it must be terminated.

C. The planned discharge may not occur until RE-22 is restored to OPERABLE status.

D. The discharge may not proceed because during the discharge, the allowable time RE-22 may be inoperable will expire.

Answer: C

Explanation:

A. Incorrect –Discharge not allowed by OP-G-2:V step 6.1.7

B. Incorrect – The allowable outage time (14 days) has ALREADY expired. This is a credible distracter because if RE-22 was declared inoperable at 0100 on 4 July, OP G-2:V step 6.1.6 would allow 3 hours of discharge.

C. Correct – as of 0100 on 17 July, the 14 days allowed by ECG (and procedure OP G- 2:V step 6.1.7) has been exceeded. A discharge is not allowed.

D. Incorrect –The allowable outage time (14 days) has ALREADY expired.

Technical References: ECG 39.4 , OP G-2:V, Gaseous Radwaste System – Gas Decay Tank Discharge

References to be provided to applicants during exam: OP G-2:V ECG 39.4
Learning Objective: To be determined

Question Source:	Bank #	X
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(13), 43(4), 45(10)	

Examination Outline Cross-Reference	Level	SRO
Knowledge of EOP terms and definitions.	**Tier #**	2
	Group #	4
	K/A #	2.4.17
	Rating	4.3

Question 99

What is meant by the term "Hot Dry" Steam Generator?

A. Primary Side Temperature >543°F and Wide Range level is less than 10% (18%).

B. Primary Side Temperature >550°F and Wide Range level is less than 10% (18%).

C. Primary Side Temperature >581°F and AFW flow has been isolated to the steam generator for an hour or longer.

D. Primary Side Temperature >635°F and AFW flow has been isolated to the steam generator for an hour or longer.

Answer: B

Explanation:

A. Incorrect. This is Lo Tave. Second part correct.

B. Correct. Section 2.4 EOP background for FR-H.1.

C. Incorrect. This is DNB limit (T/S 3.4.1/COLR) temperature and Feed & Bleed initiation criteria for level.

D. Incorrect. This 100% Tave Safety Limit and Feed & Bleed initiation criteria for level.

Technical References: FR-H.1 Background, FR-H.1 foldout page
References to be provided to applicants during exam: None
Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(10) 43(13)	

Examination Outline Cross-Reference	**Level**	SRO
Knowledge of the basis for prioritizing safety functions during abnormal/emergency operations.	**Tier #**	3
	Group #	4
	K/A #	2.4.22
	Rating	4.4

Question 100

The control room operators have entered EOP FR-H.1 "Response to Loss of Secondary Heat Sink."

The Emergency Evaluation Coordinator (EEC) identifies a RED path on the "Integrity" Critical Safety Function Status Tree.

The Shift Foreman should:

A. Continue with FR-H.1, it is a higher priority than RCS Integrity.

B. GO TO FR-P.1, RCS Integrity is a higher priority than H.1.

C. Implement FR-P.1 while continuing in FR-H.1; to minimize cooldown caused by FR-H.1 actions.

D. Immediately return to Step 1 of FR-H.1, to reassess secondary conditions.

Answer: A

Explanation:

A. Correct. FR H.1 is a higher priority than FR-P.1, higher priority CSFSTs are always continued unless a higher priority challenge is identified. Fuel and Cladding Integrity are a higher priority for protection against radiation releases..

B. Incorrect. FR P.1 is not a higher priority FRG, it is lower. FR-P.1 actions will be addressed when FR-H.1 actions are completed.

C. Incorrect. FR-P.1 actions are not done in parallel with FR-H.1 actions; this is not allowed per rules of usage.

D. Incorrect. Returning to step 1 is not required when FR-P.1 conditions are met.

Technical References: F-0 Critical Safety Functions Background Section 1.2
References to be provided to applicants during exam: None
Learning Objective: To be determined

Question Source:	Bank #	X

(note changes; attach parent)	Modified Bank # New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(7 and 10), 43(5) and 45(12)	