Examination Outline Cross-reference:	**Level**	**RO**		
	Tier #	1		
	Group #	1		
	K/A #	007EA1.08		
	Importance Rating	4.4		
Ability to operate and monitor the following as they apply to a reactor trip: AFW System.				

Question #1

Given the following conditions:

- A Reactor Trip has occurred due to a loss of PA01.
- Steam Flow indicates zero (0) on all Steam Generators.
- The Crew is performing ES-0.1, Reactor Trip Response.
- You are the Reactor Operator performing Step 5, Check Shutdown Reactivity Status, when you notice RCS Tavg is 551° and lowering.

Which ONE (1) of the following actions needs to be taken by the crew to stabilize plant conditions?

A. Transfer Condenser Steam Dumps to the Steam Pressure Mode.

B. Throttle Auxiliary Feedwater Flow to allow the plant to heatup.

C. Fast Close all Main Steam Isolation Valves to stop the cooldown.

D. Actuate Safety Injection if RCS subcooling lowers to 40°.

Answer: B

Explanation:

A. Incorrect. This would be an option if all 4 RCPs had tripped, but not an option with just 2 RCPs tripped.
B. Correct. This is directed by Step 1 of ES-0.1 which is a continuous action step.
C. Incorrect. This action is only directed if throttling of AFW is not successful and temperature continues to lower.
D. Incorrect. Lowering temperature will actually cause subcooling to rise. Also wrong setpoint for initiating SI (Actual setpoint is 30°).

Technical Reference(s): ES-0.1

References to be provided to applicants during examination: None

Learning Objective: T61.003D 6, LP-6, Obj C

Question Source: Bank # ______

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

Modified Bank # ______
New ____X___

Question History: Last NRC Exam ____N/A_________

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

10 CFR Part 55 Content:
55.41.10

Comments:

Examination Outline Cross-reference:	**Level**	**RO**		
	Tier #	1		
	Group #	1		
	K/A #	008AK2.01		
	Importance Rating	2.7		
Knowledge of the interrelations between the Pressurizer Vapor Space Accident and the following: Valves.				

Question #2

Given the following conditions:

- Mode 1 at 100% power.
- Pressurizer Relief Tank (PRT) pressure is 5 psig.

Which ONE (1) of the following describes the condition of the steam entering the PRT if a Pressurizer PORV opens?

A. Superheated steam at 659°F.

B. Superheated steam at 653°F.

C. Saturated steam-water mixture at 228°F.

D. Saturated steam-water mixture at 162°F.

Answer: C

Explanation:

A. Incorrect. Assumes no throttling and uses 2350 psia.
B. Incorrect. Assumes no throttling and uses 2250 psia.
C. Correct
D. Incorrect. Uses 5 psig rather than 20 psia.

Technical Reference(s): Steam Tables

References to be provided to applicants during examination: Steam Tables

Learning Objective: T61.GFES, LP-25, Obj 22

Question Source: Bank # __1998 Callaway Exam____
Modified Bank # ______
New _______

Question History: Last NRC Exam ____N/A________

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

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

10 CFR Part 55 Content:
55.41.14

Comments:

Examination Outline Cross-reference:	**Level**	**RO**		
	Tier #	1		
	Group #	1		
	K/A #	009EA2.10		
	Importance Rating	3.1		
Ability to determine or interpret the following as they apply to a small break LOCA: Airborne activity.				

Question #3

The plant is at 100% power when you notice the following parameters trending as indicated:

- Volume Control Tank level lowering
- Pressurizer Level lowering
- Condenser Air Disch Rad Mon (RE-92) is stable
- Containment Airborne Activity is rising
- RCS Pressure is stable

Which ONE (1) of the following plant accidents would have the indications observed above?

A. Small Break LOCA Inside Containment

B. Steamline Break Inside Containment

C. Steam Generator Tube Rupture

D. Large Break LOCA Inside Containment

Answer: A

Explanation:

A. Correct
B. Incorrect. Would have no change in containment airborne activity.
C. Incorrect. Would have no change in containment parameters.
D. Incorrect. Would have change in containment parameters but RCS pressure would be lowering, not stable.

See Accident Identification Chart on next page from Lesson T61.003D 6, D-03, Accident Analysis

	Tavg	RCS Pressure	PZR Level	VCT Level	CTMT Pr, T, Humidity	CTMT Airborne Activity	Steam Flow	Feed Flow	Steam Pressure	S/G Level	Cond. Off Gas Activity
Small LOCA INSIDE CTMT		↓	↓	↓	↑	↑					
OUTSIDE CTMT											
Steam Break Inside CTMT Affected S/G	↓	↓	↓	↓	↑		↑	↑	↓	↑↓	
Non-Affected S/G's							↗	↗	↘	↑↓	
Steam Break Outside CTMT	↓	↓	↓	↓			↑	↑	↓	↑↓	
Feedwater Break Inside CTMT	↗	↗	↗		↑			↑	↗	↓	
Outside CTMT											
S/G Tube Leak		↓	↓	↓							↑

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

Technical Reference(s):

References to be provided to applicants during examination: None

Learning Objective: T61.003D 6, LP-D-03, Objective B & C

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:
55.41.5

Comments:

Examination Outline Cross-reference:	Level	RO		
	Tier #	1		
	Group #	1		
	K/A #	015/017AK1.04		
	Importance Rating	2.9		
Knowledge of the operational implications of the following concepts as they apply to Reactor Coolant Pump Malfunctions (Loss of RC Flow): Basic steady state thermodynamic relationship between RCS loops and S/Gs resulting from unbalanced RCS flow.				

Question #4

Given the following conditions:

- The plant is at 40% power, raising load at 10%/hour.
- All systems are in a normal alignment.
- Reactor Coolant Pump “B” trips due to a breaker malfunction.

Which ONE (1) of the following describes the INITIAL response of S/G “B” LEVEL AND STEAM FLOW after the pump trip?

	S/G “B” LEVEL	S/G “B” STEAM FLOW
A.	CHANGES	LOWERS
B.	Remains CONSTANT	LOWERS
C.	Remains CONSTANT	RAISES
D.	CHANGES	RAISES

Answer: A

Explanation:

A. Correct
B. Incorrect. Heat input removed from affected SG resulting in level changes and reduced steam flow.
C. Incorrect. Heat input removed from affected SG resulting in level changes and reduced steam flow.
D. Incorrect. Heat input removed from affected SG resulting in level changes and reduced steam flow.

Technical Reference(s): OTO-BB-00002

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

References to be provided to applicants during examination: None

Learning Objective: T61.003B 6, LP-43, Obj D

Question Source: Bank # __R12196____
Modified Bank # ______
New _______

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:
55.41.14

Comments:

Examination Outline Cross-reference:	Level	RO		
	Tier #	1		
	Group #	1		
	K/A #	022 2.4.11		
	Importance Rating	4.0		
Loss of Reactor Coolant Makeup: Knowledge of abnormal condition procedures.				

Question #5

The plant is at 20% power when the following annunciators are received:

- Annunciator 38A, Ltdn Regen Hx Temp Hi
- Annunciator 41A, Seal Inj To RCP Flow Lo
- Annunciator 42A, Chg Line Flow HiLo
- Annunciator 42E, Charging Pump Trouble

The Reactor Operator starts Centrifugal Charging Pump (CCP), PBG05A, as an IMMEDIATE ACTION IAW OTO-BG-00001, Pressurizer Level Control Malfunction.

Which ONE (1) of the following additional actions must also be performed by the operating crew after starting PBG05A?

A. Isolate letdown by closing all orifice isolation valves.

B. Ensure the CCPs are aligned to the Refueling Water Storage Tank.

C. Start the Normal Charging Pump, PBG04.

D. Ensure that Component Cooling Water Pump, PEG01A, is running.

Answer: D

Explanation:

A. Incorrect. Letdown is only isolated if a charging pump can not be started.
B. Incorrect. CCPs are only aligned to the RWST if the VCT is not available.
C. Incorrect. If CCP is started, the NCP would not be started.
D. Correct

Technical Reference(s): OTO-BG-00001,Pressurizer Level Control Malfunction

References to be provided to applicants during examination: None

Learning Objective:

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ____N/A________

Question Cognitive Level:
Memory or Fundamental Knowledge __X___
Comprehension or Analysis _____

10 CFR Part 55 Content:
55.41.10

Comments:

Examination Outline Cross-reference:	Level	RO		
	Tier #	1		
	Group #	1		
	K/A #	025AA2.01		
	Importance Rating	2.7		
Ability to determine and interpret the following as they apply to the Loss of Residual Heat Removal System: Proper amperage of running LPI/decay heat removal/RHR pump(s).				

Question #6

Improper filling and venting following maintenance on Residual Heat Removal Pump, PEJ01A, results in the pump being started for post maintenance testing with the suction piping completely air bound.

Which ONE (1) of the following describes pump current that the Reactor Operator starting PEJ01A will observe on the Main Control Board once the pump has been started?

A. Pump current indicating zero (0)

B. Pump current oscillating

C. Pump current higher than normal

D. Pump current less than normal

Answer: D

Explanation:

A. Incorrect. Pump current will be less than normal but not zero.
B. Incorrect. If the pump is cavitating, there will be intermittent fluid flow. Therefore, the motor current will fluctuate. It will be near normal during the period of good fluid flow and low during periods of gas movement through the pump.
C. Incorrect. Higher current indicates higher than normal flow-not the case if air bound.
D. Correct

Technical Reference(s):

References to be provided to applicants during examination: None

Learning Objective: T61.GFES, LP-27, Obj 33d

Question Source: Bank # ______
Modified Bank # ______
New ___X____

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:
55.41.14

Comments:

Examination Outline Cross-reference:	**Level**	**RO**		
	Tier #	1		
	Group #	1		
	K/A #	026AA2.06		
	Importance Rating	2.8		
Ability to determine and interpret the following as they apply to the Loss of Component Cooling Water: The length of time after the loss of CCW flow to a component before that component may be damaged.				

Question #7

Which ONE (1) of the following conditions would require the affected Reactor Coolant Pump(s) to be tripped?

A. The running Component Cooling Water Pump tripped eight (8) minutes ago.

B. EG HV-62, CCW From RCS In Ctmt Iso Hv (RCP Thermal Barriers), failed closed 15 minutes ago.

C. EG HV-60, CCW From RCS In Ctmt Iso Hv (RCP Motor Coolers), failed closed 12 minutes ago.

D. BB HV8351A, RCP A Seal Wtr Inj Vlv, fails closed.

Answer: C

Explanation:

A. Incorrect. Loss of flow to the RCPs is less than 10 minutes and the standby CCW pump should have started.
B. Incorrect. There is no time limit for loss of thermal barrier cooling as long as seal injection is maintained.
C. Correct. If cooling water is lost to the RCP motors for greater than 10 minutes, they are required to be secured.
D. Incorrect. There is no requirement to trip RCPs on a loss of seal injection as long as thermal barrier cooling is maintained by CCW.

Technical Reference(s): OTO-BB-00002

References to be provided to applicants during examination: None

Learning Objective: T61.0110 6, LP-10, Obj E

Question Source: Bank # ______
Modified Bank # ______
New ___N____

Question History: Last NRC Exam ____N/A________

Question Cognitive Level:

Memory or Fundamental Knowledge __X___

Comprehension or Analysis _____

10 CFR Part 55 Content:

55.41.8

Comments:

Examination Outline Cross-reference:	Level	RO		
	Tier #	1		
	Group #	1		
	K/A #	027AK3.04		
	Importance Rating	2.8		
Knowledge of the reasons for the following responses as they apply to the Pressurizer Pressure Control Malfunctions: Why, if PZR level is lost and then restored, that pressure recovers much more slowly?				

Question #8

Following a cooldown caused by a Steam Dump malfunction, pressurizer level fell below 17% and was rapidly restored by raising charging flow.

Pressurizer pressure lowered to a value of 2185 psig.

Which ONE (1) of the following correctly identifies the reason why the pressure recovery from 2185 psig takes a longer time for this event, than it does if a PORV fails open and the PORV block valve is closed at 2185 psig?

A. The volume of steam generation and cooling is greater with the level change.

B. Subcooled water insurge during refill reduced the pressurizer liquid space temperature.

C. When the PORV opens, only the steam space needs to be reheated to raise pressure.

D. The PZR heaters are less effective since they are operating at a reduced capacity on the level change.

Answer: B

Explanation:

A. Incorrect. Volume of steam generation is actually less with the level change.
B. Correct. The process of heating the water to the saturation temperature takes a longer time than for the case where the water inventory remains saturated but at a lower pressure.
C. Incorrect. Liquid space will also need to be heated up due to the insurge, but a smaller amount of volume of cold water will enter the pressurizer.
D. Incorrect. Heaters will have no effect since they trip off at 17%.

Technical Reference(s): T61. GFES, LP-24, Steam

References to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # _ Salem Exam - 1998_____
Modified Bank # ______
New _______

Question History: Last NRC Exam ____N/A________

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X___

10 CFR Part 55 Content:
55.41.5

Comments:

Examination Outline Cross-reference:	Level	RO		
	Tier #	1		
	Group #	1		
	K/A #	029EK2.06		
	Importance Rating	2.9		
Knowledge of the interrelations between the following and an ATWS: Breakers, relays, and disconnects.				

Question #9

Given the following conditions:

- The plant is at 100%.
- PAE01, Main Feedwater Pump "A", trips.
- The Reactor Operator and Balance of Plant Operator attempt to trip the reactor from SB HS-1 and SB HS-42, respectively, IAW OTO-AE-00001, Feedwater System Malfunction.
- NO Rod Bottom Lights are LIT.

Which ONE (1) of the following Immediate Actions will be taken by the Crew in response to the above indications?

A. OPEN PG HIS-16 and PG HIS-18, supply breakers for Load Centers PG19 and PG20.

B. OPEN the Rod Control Generator Circuit Breaker Control switches.

C. OPEN the Rod Control Motor Circuit Breaker Control switches.

D. DIRECT the Primary OT to push the breaker Trip buttons on the Reactor Trip and Reactor Trip Bypass Breakers.

Answer: A

Explanation:

A. Correct. Immediate Action in Step 1 of FR-S.1.
B. Incorrect. Is directed by FR-S.1, but not until Step 6. Only Steps 1 and 2 are Immediate Action Steps. Also, would only be performed in Trip Breakers can not be opened.
C. Incorrect. Is directed by FR-S.1, but not until Step 6. Only Steps 1 and 2 are Immediate Action Steps. Also, would only be performed in Trip Breakers can not be opened.
D. Incorrect. Is directed by FR-S.1, but not until Step 6. Only Steps 1 and 2 are Immediate Action Steps.

Technical Reference(s): FR-S.1

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

References to be provided to applicants during examination: None

Learning Objective: T61.003D 6, LP-29, Obj C

Question Source: Bank # ______
Modified Bank # ______
New ____X___

Question History: Last NRC Exam ____N/A________

Question Cognitive Level:
Memory or Fundamental Knowledge __X___
Comprehension or Analysis _____

10 CFR Part 55 Content:
55.41.10

Comments:

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

Examination Outline Cross-reference:	**Level**	**RO**		
	Tier #	1		
	Group #	1		
	K/A #	038EK1.03		
	Importance Rating	3.9		
Knowledge of the operational implications of the following concepts as they apply to the SGTR: Natural circulation.				

Question #10

Given the following conditions:

- A Steam Generator Tube Rupture has occurred in S/G “D”.
- ALL Reactor Coolant Pumps (RCPs) tripped due to a loss of the Startup Transformer.
- The Crew has completed E-3, Steam Generator Tube Rupture, and transitioned to ES-3.1, Post-SGTR Cooldown Using Backfill.
- Conditions have been established that allow restarting the “D” RCP at Step 1 in ES-3.1.

Which ONE (1) of the following is the greatest concern with starting “D” RCP?

A. Pressurized thermal shock of the reactor vessel.

B. Radioactive release via the “D” S/G atmospheric steam dump.

C. Secondary depressurization could initiate Steam Line Isolation Signal.

D. Localized boron dilution in the reactor core region.

Answer: D

Explanation:

A. Incorrect. There is no concern with PTS during a SGTR with no uncontrolled cooldown.
B. Incorrect. Starting an RCP will not raise secondary pressure. E-3 has already lowered affected S/G pressure to RCS pressure.
C. Incorrect. Starting an RCP will not lower secondary pressure. Having forced flow will allow the control room crew to better control pressure.
D. Correct. Caution at beginning of ES-3.1 and basis allude to localized boron dilution and possible inadvertent criticality.

Technical Reference(s): ES-3.1 and BD-ES-3.1

References to be provided to applicants during examination: None

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

Learning Objective: T61.0110 6, LP D-18, Obj F

Question Source: Bank # __R11963____
Modified Bank # ______
New _______

Question History: Last NRC Exam ____N/A________

Question Cognitive Level:
Memory or Fundamental Knowledge __X___
Comprehension or Analysis _____

10 CFR Part 55 Content:
55.41.10

Comments:

Examination Outline Cross-reference:	Level	RO		
	Tier #	1		
	Group #	1		
	K/A #	054 2.1.31		
	Importance Rating	4.6		
Loss of Main Feedwater (MFW): Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup.				

Question #11

Given the following conditions:

- A power ascension is in progress.
- The Crew has just verified that permissive P-9 bistable lights are illuminated.
- PAE01B, Main Feedwater Pump "B" trips.
- The Control Room Supervisor directs the Balance of Plant Operator to verify that the Feedwater to Steam Header differential pressure is accurate for the current plant conditions.

Which ONE (1) of the following is the expected differential pressure for the current plant conditions and what is a reason for maintaining this differential pressure?

	Differential Pressure	Reason
A.	97 psid	Maintain Main Feedwater Control Valves in a more linear operating range
B.	75 psid	Maintain Main Feedwater Control Valves in a more linear operating range
C.	97 psid	Efficiency is improved by reducing pump power requirements at high power
D.	75 psid	Efficiency is improved by reducing pump power requirements at high power

Answer: A

Explanation:

A. Correct.
B. Incorrect. Dp is calculated using minimum value of 0 rather than 45.
C. Incorrect. Efficiency is improved at low power, not high power.
D. Incorrect. Dp is calculated using minimum value of 0 rather than 45 and efficiency is improved at low power, not high power.

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

Technical Reference(s): OTN-AE-00001

References to be provided to applicants during examination: None

Learning Objective: T61.0110 6, LP-23, Obj E

Question Source: Bank # ______
Modified Bank # ______
New ____X___

Question History: Last NRC Exam ____N/A________

Question Cognitive Level:
Memory or Fundamental Knowledge __X___
Comprehension or Analysis _____

10 CFR Part 55 Content:
55.41.4

Comments:

Examination Outline Cross-reference:	Level	RO		
	Tier #	1		
	Group #	1		
	K/A #	055EK3.02		
	Importance Rating	4.3		
Knowledge of the reasons for the following responses as they apply to the Station Blackout: Actions contained in EOP for loss of offsite and onsite power.				

Question #12

A CAUTION provided to the Control Room Crew prior to commencing a secondary depressurization in ECA-0.0, Loss of All AC Power, states a minimum pressure that the Steam Generators should be depressurized to.

Which ONE (1) of the following is the minimum pressure given in ECA-0.0 and the reason given to not go below this pressure?

	Pressure	Reason
A.	160 PSIG	Prevent a pressurized thermal shock concern for the reactor vessel
B.	260 PSIG	Prevent injection of accumulator nitrogen into the RCS
C.	160 PSIG	Prevent injection of accumulator nitrogen into the RCS
D.	260 PSIG	Prevent a pressurized thermal shock concern for the reactor vessel

Answer: C

Explanation:

A. Incorrect. PTS concern is addressed by giving a minimum RCS temperature to not go below during the depressurization.
B. Incorrect. This is the pressure to stop the depressurization, not the minimum pressure given for the depressurization.
C. Correct
D. Incorrect. This is the pressure to stop the depressurization, not the minimum pressure given for the depressurization.

Technical Reference(s): ECA-0.0

References to be provided to applicants during examination: None

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

Learning Objective: T61.0110 6, LP D-22, Obj Q

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ____N/A________

Question Cognitive Level:
Memory or Fundamental Knowledge __X___
Comprehension or Analysis _____

10 CFR Part 55 Content:
55.41.10

Comments:

Examination Outline Cross-reference:	Level	RO		
	Tier #	1		
	Group #	1		
	K/A #	056AK1.04		
	Importance Rating	3.1		
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.				

Question #13

Given the following conditions:

- A Reactor Trip and Safety Injection have occurred from 100% power due to a seismic event.
- Offsite power has been lost.
- Pressurizer PORV, BB PCV-456A, has lifted and stuck open.
- Pressurizer PORV Block Valve, BB HV-8000B, has failed open and can not be closed.
- Pressurizer level is 65% and rising.
- RCS temperature is 550°F.
- RCS pressure is 1015 psig.

Which ONE (1) of the following describes the condition of the RCS and primary concern associated with this condition?

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

B. Saturated, boiling is now occurring in the RCS which would preclude starting RCPs.

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

D. 25°F subcooled, 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 subcooling does 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 starting the RCPs.
C. Correct
D. Incorrect. Subcooling does not exist when the RCS is at saturation.

Technical Reference(s): Steam Tables

References to be provided to applicants during examination: Steam Tables

Learning Objective:

Question Source: Bank # __2010 Diablo Canyon Exam____
Modified Bank # ______
New _______

Question History: Last NRC Exam ____N/A________

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X___

10 CFR Part 55 Content:
55.41.5

Comments:

Examination Outline Cross-reference:	**Level**	**RO**		
	Tier #	1		
	Group #	1		
	K/A #	057AA1.04		
	Importance Rating	3.5		
Ability to operate and/or monitor the following as they apply to the Loss of Vital AC Instrument Bus: RWST and VCT valves.				

Question #14

Given the following conditions:

- Reactor Power is 80%.
- Annunciator 47D, RWST Lev HiLo, is in alarm.
- Reactor Coolant Temperature is lowering.

Which ONE (1) of the following conditions has occurred in the plant?

A. Loss of NK03

B. Loss of NN01

C. RWST Level Channel, BN LT-932, has failed high

D. VCT Level Channel, BG LT-185, has failed high

Answer: B

Explanation:

A. Incorrect. Loss of NK03 will not affect suction supplies to the CCPs.
B. Correct. Loss of NN01 will cause the CCP suction source to switch from the VCT to the RWST due to VCT level channel failing low.
C. Incorrect. Loss of 932 will give ANN 47D, but will not affect suction supplies to the CCPs.
D. Incorrect. Failure high will not affect suction supplies to the CCPs; CCP suction would be swapped if failure was low.

Technical Reference(s): OTO-NN-00001

References to be provided to applicants during examination: None

Learning Objective: T61.003B 6, LP-27, Obj D

Question Source: Bank # ______
Modified Bank # ______
New ____X___

Question History: Last NRC Exam ____N/A________

Question Cognitive Level:

Memory or Fundamental Knowledge ______

Comprehension or Analysis __X___

10 CFR Part 55 Content:

55.41.7

Comments:

Examination Outline Cross-reference:	Level	RO		
	Tier #	1		
	Group #	1		
	K/A #	058AA1.01		
	Importance Rating	3.4		
Ability to operate and/or monitor the following as they apply to the Loss of DC Power: Cross-tie of the affected dc bus with the alternate supply.				

Question #15

Battery Charger NK21 has failed.

Which ONE (1) of the following is NOT a source that can supply power to 125 VDC Bus NK01?

A. Battery NK11

B. Breaker NG0109

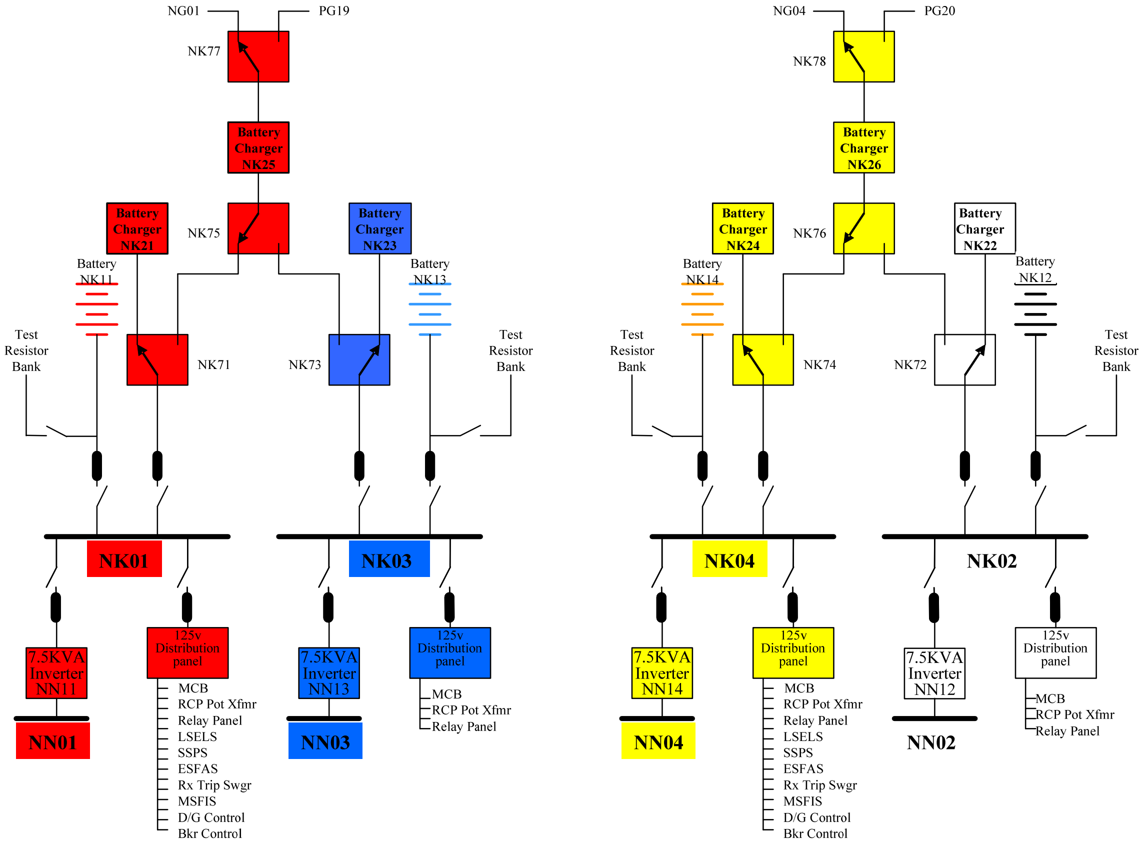
C. Breaker NG0303

D. Breaker PG1910

Answer: C

Explanation:

A. Incorrect. Can supply power to NK01 (See Drawing Below).
B. Incorrect. Can supply power to NK01 (See Drawing Below).
C. Correct. NG03 can not supply power to NK01 (See Drawing Below).
D. Incorrect. Can supply power to NK01 (See Drawing Below).



Technical Reference(s): OTN-NK-00001

References to be provided to applicants during examination: None

Learning Objective: T61.0110 6, LP-06, Obj B.5

Question Source: Bank # ______
Modified Bank # ______
New ____X___

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:
55.41.7

Comments:

Examination Outline Cross-reference:	Level	RO		
	Tier #	1		
	Group #	1		
	K/A #	W/E05 G2.1.19		
	Importance Rating	3.9		
Loss of Secondary Heat Sink: Ability to use plant computers to evaluate system or component status.				

Question #16

A Reactor Trip and Safety Injection have occurred from 100% power.

“A” Centrifugal Charging Pump failed to start and cannot be started.

As the Reactor Operator you observe plant conditions as given on the plant computer screens attached.

Which ONE (1) of the following will be the FIRST action to be taken?

(REFERENCES PROVIDED)

A. Align AFW suction to ESW

B. Initiate RCS Feed and Bleed

C. Reset Safety Injection

D. Stop all Reactor Coolant Pumps

Answer: D

Explanation:

A. Incorrect. This would only be performed if CST level lowers to <36%.
B. Incorrect. Do not meet any of the Feed and Bleed criteria of SG low levels, high PZR pressure or no CCPs in service.
C. Incorrect. This would only be done after initiating a Feed and Bleed operation.
D. Correct. This will be the first action taken in FR-H.1 to minimize heat input into the RCS.

Technical Reference(s): CSF-1, FR-H.1

References to be provided to applicants during examination: Yes, two computer screen printouts

Learning Objective: T61.003D 6, LP-01, Obj T; T61.003D 6, LP-26, Obj Q

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

Question Source: Bank # ______
Modified Bank # ______
New ____X___

Question History: Last NRC Exam ____N/A________

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X___

10 CFR Part 55 Content:
55.41.10

Comments:

Examination Outline Cross-reference:	Level	RO		
	Tier #	1		
	Group #	1		
	K/A #	W/E11EK3.2		
	Importance Rating	3.5		
Knowledge of the reasons for the following responses as they apply to the (Loss of Emergency Coolant Recirculation): Normal, abnormal and emergency operating procedures associated with (Loss of Emergency Coolant Recirculation).				

Question #17

Which ONE (1) of the following is a reason for implementing ECA-1.1, Loss of Emergency Coolant Recirculation?

A. A Loss of Coolant Accident has occurred outside of containment.

B. Depressurizing the RCS to minimize break flow and cause SI accumulator injection.

C. Raising main steam flow to the condenser to reduce the need for RCS cooling from ECCS components is required.

D. ECCS recirculation flow cannot be maintained due to the affects of sump blockage in containment.

Answer: B

Explanation:

A. Incorrect. Plausible as mass is being loss from the RWST which would affect the ability to transfer to cold leg recirculation, but this condition would be addressed by a transition to ECA-1.2.
B. Correct.
C. Incorrect. This is a plausible reason for ECA-1.1 as the need for ECCS cooling would be reduced if the RCS temp were lowered, however ECA-1.1 does not perform this action
D. Incorrect. This is a transition from ECA-1.1 to ECA-1.3 if this condition exists..

Technical Reference(s): ECA-1.1

References to be provided to applicants during examination: None

Learning Objective: T61.003D 6, LP-13, Obj A

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ____N/A________

Question Cognitive Level:
Memory or Fundamental Knowledge __X___
Comprehension or Analysis _____

10 CFR Part 55 Content:
55.41.10

Comments:

Examination Outline Cross-reference:	Level	RO		
	Tier #	1		
	Group #	1		
	K/A #	W/E12EK2.2		
	Importance Rating	3.6		
Knowledge of the interrelations between the (Uncontrolled Depressurization of all Steam Generators) and the following: Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.				

Question #18

Given the following plant conditions:

- ECA-2.1, Uncontrolled Depressurization of All Steam Generators, is being performed by the Crew
- The operators have reduced AFW flow to all steam generators (SG) to minimum as they continue attempts to isolate the SGs

Which ONE (1) of the following describes the plant response to the AFW flow reduction and what actions are to be taken as SG pressures lower?

	Plant Response	Crew Actions
A.	RCS hot leg temperature will eventually begin to rise	Crew will terminate Safety Injection using ES-1.1, SI Termination
B.	The SGs will eventually become completely depressurized	Crew will terminate Safety Injection using ECA-2.1, Uncontrolled Depressurization of All steam Generators
C.	RCS hot leg temperature will eventually begin to rise	Crew will terminate Safety Injection using ECA-2.1, Uncontrolled Depressurization of All steam Generators
D.	The SGs will eventually become completely depressurized	Crew will terminate Safety Injection using ES-1.1, SI Termination

Answer: C

Explanation:

A. Incorrect. There is no transition to ES-1.1 from ECA-2.1.

B. Incorrect. The SGs will not be allowed to completely depressurize if control of AFW flow is available.
C. Correct
D. Incorrect. The SGs will not be allowed to completely depressurize if control of AFW flow is available and no transition to ES-1.1.

Technical Reference(s): ECA-2.1 and BD-ECA-2.1

References to be provided to applicants during examination: None

Learning Objective: T61.003D 6, LP-16, Obj I

Question Source: Bank # __Callaway 2005 Exam____
Modified Bank # ______
New _______

Question History: Last NRC Exam ____N/A________

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X___

10 CFR Part 55 Content:
55.41.10

Comments:

Examination Outline Cross-reference:	Level	RO		
	Tier #	1		
	Group #	2		
	K/A #	003AK1.19		
	Importance Rating	2.8		
Knowledge of the operational implications of the following concepts as they apply to Dropped Control Rod: Differential rod worth.				

Question #19

Given the following conditions:

- Reactor Power is 100%
- A control rod drops into the core from 215 steps

In which ONE (1) of the following conditions would the dropped rod add the largest amount of negative reactivity to the reactor?

A. Beginning of Life from the periphery of the core.

B. End of Life from the periphery of the core.

C. End of Life from the center of the core.

D. Beginning of Life from the center of the core.

Answer: C

Explanation:

$$DRW = C\left(\frac{\phi_{tip}}{\phi_{avg}}\right)\Psi$$

Where:

DRW	=	differential rod worth
C	=	constant based on rod size, shape and material
ϕ_{tip}	=	neutron flux near rod tip
ϕ_{avg}	=	average neutron flux in core

ψ	=	flux importance factor

A. Incorrect. See equation above – low importance factor and more competition at BOL.
B. Incorrect. See equation above – low importance factor.
C. Correct
D. Incorrect. See equation above – more competition at BOL.

Technical Reference(s):

References to be provided to applicants during examination: None

Learning Objective: T61.GFES 6, LP-24, Objs 5 and 9

Question Source: Bank # ______
Modified Bank # ______
New ____X___

Question History: Last NRC Exam ____N/A________

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X___

10 CFR Part 55 Content:
55.41.1

Comments:

Examination Outline Cross-reference:	Level	RO		
	Tier #	1		
	Group #	2		
	K/A #	005AK2.01		
	Importance Rating	2.5		
Knowledge of the interrelations between the Inoperable/Stuck Control Rod and the following: Controllers and positioners.				

Question #20

Given the following conditions:

- Reactor Power is 100%.
- The Reactor Operator (RO) commences a planned plant shutdown by manually inserting the Control Rods.
- Control Rod D12 in Control Bank D does not move with the other Control Bank D rods when inserted by the RO.
- Annunciator 79A, ROD CTRL URG FAIL, is in alarm.

Which ONE (1) of the following would be the FIRST indication available to the RO that Control Rod D12 is not moving **AND** is the rod OPERABLE or INOPERABLE?

	INDICATION	OPERABLE/INOPERABLE
A.	Deviation of rod position indicated on SB-074, DRPI Rod Position Indication	OPERABLE
B.	Deviation of rod position indicated on SC CB-D2, Step Counter CTRL Bank Indication	INOPERABLE
C.	Deviation of rod position indicated on SC CB-D2, Step Counter CTRL Bank Indication	OPERABLE
D.	Deviation of rod position indicated on SB-074, DRPI Rod Position Indication	INOPERABLE

Answer: A

Explanation:

A. Correct
B. Incorrect. Step Counters do not give individual positions but give the demand position for the rod and does not receive feedback on individual rod position. It gives a more accurate, but less reliable indication of rod position. The rod is considered operable unless it is determined to be untripable and there is no information given that implies the rod would not trip if required.
C. Incorrect. Step Counters do not give individual positions but give the demand position for the rod and does not receive feedback on individual rod position. It gives a more accurate, but less reliable indication of rod position.
D. Incorrect. The rod is considered operable unless it is determined to be untripable and there is no information given that implies the rod would not trip if required.

Technical Reference(s): Tech Specs

References to be provided to applicants during examination: None

Learning Objective: T61.0110 6, LP-26, Obj N

Question Source: Bank # ______
Modified Bank # ______
New ____X___

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:
55.41.6

Comments:

Examination Outline Cross-reference:	Level	RO		
	Tier #	1		
	Group #	2		
	K/A #	028AA2.12		
	Importance Rating	3.1		
Ability to determine and interpret the following as they apply to the Pressurizer Level Control Malfunctions: Cause for PZR level deviation alarm: controller malfunction or other instrumentation malfunction.				

Question #21

Given the following conditions:

- Reactor Power is 100%.
- Pressurizer level is 56% and rising.
- Pressurizer pressure is 2235 psig and slowly rising.
- Letdown flow isolates.
- Charging flow lowers.
- Annunciator 32B, PZR 17% HTRS OFF, alarms.
- Annunciator 32C, PZR LO LEV DEV, alarms.
- All other parameters are normal.

Which ONE (1) of the following has occurred?

A. Pressurizer Steam Space Leak

B. Pressurizer Level Channel Failure

C. Pressurizer Pressure Channel Failure

D. Letdown Line Break

Answer: B

Explanation:

A. Incorrect. Rising PZR level and lowering charging are indicators of a steam space leak but other given conditions do not support a steam space leak.
B. Correct
C. Incorrect. A failed pressure instrument could affect pressure and level depending on the failure but all conditions are not consistent for a pressure failure.
D. Incorrect. Letdown has isolated as expected due to the controlling level channel failing.

Technical Reference(s): OTO-BG-00001

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

References to be provided to applicants during examination: None

Learning Objective: T61.003B 6, LP-43, Obj B

Question Source: Bank # _Diablo Canyon 2010_____
Modified Bank # ______
New _______

Question History: Last NRC Exam ____N/A________

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X___

10 CFR Part 55 Content:
55.41.10

Comments:

Examination Outline Cross-reference:	Level	RO		
	Tier #	1		
	Group #	2		
	K/A #	032 2.1.43		
	Importance Rating	4.1		
Loss of Source Range Nuclear Instrumentation: Conduct of Operations: Ability to use procedures to determine the effects or reactivity of plant changes, such as reactor coolant system temperature, secondary plant, fuel depletion, etc.				

Question #22

Given the following conditions:

- Callaway Plant has just finished a refueling.
- The Plant is in Mode 3.
- The Reactor Trip Breakers are closed.
- RCS temperature is stable at 557°.
- RCS pressure is stable at 2235 psig.
- Annunciator 42B, VCT LEV HILO, alarms.

The Reactor Operator notices that RCS boron is rising as indicated by Boron Concentration Monitoring System (BCMS) on Panel RL001.

Which ONE (1) of the following has occurred to cause RCS boron concentration to increase **AND** what action must be taken to recover from the failure without causing a Reactor Trip?

A. Source Range (SR) NIS Channel N31 failed LOW — Place Normal/Test switch on SR Drawer N31 in Test

B. Source Range (SR) NIS Channel N31 failed HIGH — Remove the Control Power Fuses for N31

C. Source Range (SR) NIS Channel N32 failed HIGH — Place Normal/Test switch on SR Drawer N32 in Test

D. Source Range (SR) NIS Channel N32 failed LOW — Remove the Control Power Fuses for N32

Answer: C

Explanation:

VCT outlet valves will swap to the RWST on a flux doubling when the reactor is shutdown. This will result in water from the RWST being supplied to the RCS, thus raising RCS boron.

A. Incorrect. Valves will swap on a flux doubling, not a failure low.
B. Incorrect. Removing the control power fuses will result in a reactor trip.
C. Correct
D. Incorrect. Valves will swap on a flux doubling, not a failure low. Removing the control power fuses will result in a reactor trip.

Technical Reference(s): OTO-SE-00001

References to be provided to applicants during examination: None

Learning Objective: T61.003B 6, LP-42, Obj D
T61.0110 6, LP-11, Obj B-14

Question Source: Bank # ______
Modified Bank # ______
New ____X___

Question History: Last NRC Exam ____N/A________

Question Cognitive Level:
Memory or Fundamental Knowledge ____
Comprehension or Analysis __X___

10 CFR Part 55 Content:
55.41.10

Comments:

Examination Outline Cross-reference:	Level	RO		
	Tier #	1		
	Group #	2		
	K/A #	051AK3.01		
	Importance Rating	2.8		
Knowledge of the reasons for the following responses as they apply to the Loss of Condenser Vacuum: Loss of steam dump capability upon loss of condenser vacuum.				

Question #23

A circulating water pump tripped with the plant at full power three minutes ago.

The following plant conditions exist:

• Auct High T_{ave}	590°F and lowering
• Reactor Power	75% and stable
• Turbine Load	880 MWe and stable
• LP 'A' Cond Press	4.7" HgA
• IP 'B' Cond Press	5.0" HgA
• HP 'C' Cond Press	6.6" HgA

What is the expected condenser steam dump automatic operation under these conditions?

A. <12 dumps available and dumps closed

B. 12 dumps available and tripped open

C. 12 dumps available and some modulated open

D. <12 dumps available and dumps open

Answer: D

Explanation:

A. Incorrect. All available steam dumps would be armed from the turbine setback and open at this temperature following a CW pump trip.
B. Incorrect. Condenser interlock for steam dumps is 6.0 HgA, therefore only the dumps in A and B condensers are available, not all dumps.
C. Incorrect. Condenser interlock for steam dumps is 6.0 HgA, therefore only the dumps in A and B condensers are available, not all dumps.
D. Correct

Technical Reference(s): OTO-AD-00001

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

References to be provided to applicants during examination: None

Learning Objective: T61.0110 6, LP-20, Obj I

Question Source: Bank # __L7368____
Modified Bank # ______
New _______

Question History: Last NRC Exam ____N/A________

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X___

10 CFR Part 55 Content:
55.41.4

Comments:

Examination Outline Cross-reference:	Level	RO		
	Tier #	1		
	Group #	2		
	K/A #	059AK2.01		
	Importance Rating	2.7		
Knowledge of the interrelations between the Accidental Liquid Radwaste Release and the following: Radioactive-liquid monitors.				

Question #24

A liquid Radwaste (RW) release from Discharge Monitor Tank (DMT) 'A' is in progress.

Which ONE (1) of the following conditions would AUTOMATICALLY terminate the release?

A. Cooling Tower blowdown flow rate is REDUCED to 6000 gpm

B. RW Building Discharge Rad Monitor, HB RE-18 FAILS, resulting in a HI HI alarm

C. A valve misalignment causes a HI HI LEVEL alarm in DMT 'A'

D. A Hi Hi alarm on S/G discharge rad monitor causes BM FV-54 to CLOSE

Answer: B

Explanation:

A. Incorrect. Setpoint on CT blowdown flow to secure a DMT discharge is adjustable from 3000 to 5000 gpm.
B. Correct
C. Incorrect. This would cause the inlet valve to the DMT to close but would not close the outlet valve to stop the release.
D. Incorrect. This would stop a plant discharge from the SG blowdown system but not from the DMT.

Technical Reference(s): OTA-SP-RM011

References to be provided to applicants during examination: None

Learning Objective: T61.0110 6, LP-16, Obj G

Question Source: Bank # __L7084____
Modified Bank # ______
New ________

Question History: Last NRC Exam ____N/A________

Question Cognitive Level:
Memory or Fundamental Knowledge __X___
Comprehension or Analysis _____

10 CFR Part 55 Content:
55.41.11

Comments:
New Distractor C

Examination Outline Cross-reference:	**Level**	**RO**		
	Tier #	1		
	Group #	2		
	K/A #	W/E03EA1.3		
	Importance Rating	3.7		
Ability to operate and / or monitor the following as they apply to the (LOCA Cooldown and Depressurization): Desired operating results during abnormal and emergency situations.				

Question #25

A Loss of Coolant Accident (LOCA) has occurred and the Crew has transitioned to ES-1.2, Post LOCA Cooldown and Depressurization.

Prior to commencing a depressurization of the Reactor Coolant System (RCS), the procedure states that the upper head region may void during RCS depressurization.

Which ONE (1) of the following conditions could cause the upper head region to void **AND** what would be the result if voiding occurs?

	CAUSE	RESULT
A.	Reactor Coolant Pumps NOT RUNNING	Rapidly rising Pressurizer Level
B.	RCS is not COUPLED to the Steam Generators	Loss of RCS Subcooling
C.	RCS is not COUPLED to the Steam Generators	Rapidly rising Pressurizer Level
D.	Reactor Coolant Pumps NOT RUNNING	Loss of RCS Subcooling

Answer: A

Explanation:

A. Correct
B. Incorrect. The steam generators are not required for an RCS cooldown in this procedure. They can be used if they are available. If available, the RHR system can be used. The voiding occurs in the upper head region due to the lack of flow in this region which would be supplied by the RCPs, not from whether the RCS is coupled to the secondary plant or not.
C. Incorrect. The steam generators are not required for an RCS cooldown in this procedure. They can be used if they are available. If available, the RHR system can be used. The voiding

occurs in the upper head region due to the lack of flow in this region which would be supplied by the RCPs, not from whether the RCS is coupled to the secondary plant or not.
D. Incorrect. Cause is correct but result is not. As described in the basis for the step to depressurize, subcooling may be lost, but it would be lost from the depressurization, not the voiding in the upper head region.

Technical Reference(s): ES-1.2

References to be provided to applicants during examination: None

Learning Objective: T61.003D 6, LP-10, Obj F

Question Source: Bank # ______
Modified Bank # ______
New ____X___

Question History: Last NRC Exam ____N/A________

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41.10
Comments:

Examination Outline Cross-reference:	Level	RO		
	Tier #	1		
	Group #	2		
	K/A #	W/E07EA1.1		
	Importance Rating	3.6		
Ability to operate and / or monitor the following as they apply to the (Saturated Core Cooling): Components, and functions of control and safety systems, including instrumentation signal, interlocks, failure modes, and automatic and manual features.				

Question #26

Following a LOCA with subsequent ECCS failures, the Crew is performing the actions in FR-C.2, Response To Degraded Core Cooling.

Current conditions are:
- RCS pressure is rising
- Core Cooling has NOT been restored

Which ONE (1) of the following describes the required operation of the Pressurizer PORVs in this event?

A. Manually OPEN to depressurize the RCS to facilitate SI accumulator injection.

B. Closed and ISOLATED until required to establish a vent path prior to RCP restart.

C. Closed and ISOLATED to prevent further loss of RCS inventory.

D. Closed and UNISOLATED for automatic overpressure protection as necessary.

Answer: D

Explanation:

A. Incorrect. This is plausible as it would allow the accumulators to inject but this is not a method utilized in FR-C.2.
B. Incorrect. Possible action to be taken in FR-C.1, not an action in FR-C.2.
C. Incorrect. PORVs are always left available for RCS overpressure protection. They would only be isolated if a valve failure occurred.
D. Correct

Technical Reference(s): FR-C.2

References to be provided to applicants during examination: None

Learning Objective:

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

Question Source: Bank # __Callaway 2005 Exam____
Modified Bank # ______
New _______

Question History: Last NRC Exam ____N/A________

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X___

10 CFR Part 55 Content:
55.41.10

Comments:

Examination Outline Cross-reference:	Level	RO		
	Tier #	1		
	Group #	2		
	K/A #	W/E16EA2.1		
	Importance Rating	2.9		
Ability to determine and interpret the following as they apply to the (High Containment Radiation): Facility conditions and selection of appropriate procedures during abnormal and emergency operations.				

Question #27

A Reactor Trip and Safety Injection have occurred from 100% power.
The Control Room Crew is currently performing ES-1.1, SI Termination.

Which ONE (1) of the following groups of indications has revised limits to use during adverse containment conditions **AND** what containment parameters are used to determine adverse containment?

	Indications With Adverse Containment Values	Adverse Containment Parameters
A.	Pressurizer Level RCS Subcooling SG Narrow Range Level	Containment Pressure Containment Radiation
B.	Pressurizer Level Pressurizer Pressure SG Narrow Range Level	Containment Temperature Containment Radiation
C.	Pressurizer Level RCS Subcooling SG Narrow Range Level	Containment Temperature Containment Radiation
D.	Pressurizer Level Pressurizer Pressure SG Narrow Range Level	Containment Pressure Containment Radiation

Answer: A

Explanation:

A. Correct
B. Incorrect. Prz pressure does not have adverse value and ctmt temp is not used for adverse conditions.
C. Incorrect. Ctmt temp is not used for adverse conditions.

D. Incorrect. Prz pressure does not have adverse value.

Technical Reference(s): ES-1.1

References to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # ______
Modified Bank # ______
New ____X___

Question History: Last NRC Exam ____N/A________

Question Cognitive Level:
Memory or Fundamental Knowledge __X___
Comprehension or Analysis _____

10 CFR Part 55 Content:
55.41.10

Comments:

Examination Outline Cross-reference:	Level	RO		
	Tier #	2		
	Group #	1		
	K/A #	003K6.02		
	Importance Rating	2.7		
Knowledge of the effect of a loss or malfunction on the following will have on the RCPS: RCP seals and seal water supply.				

Question #28

Callaway Plant has been operating at 100% when the RO notices that 'A' RCP #1 seal leakoff flow has dropped 1 gpm.

Which ONE of the following is the most probable cause for the low seal leakoff flow?

A. Failure of the #2 seal.

B. High seal water injection temperature.

C. Low VCT Pressure.

D. Loss of seal injection.

Answer: A

Explanation:

A.Correct
B.Incorrect. High seal injection temperature would not affect the leakoff of #1 seal.
C.Incorrect. Low VCT pressure would cause #1 seal leakoff to rise.
D.Incorrect RCP #1 seal would continue to function with flow supplied from the RCS being cooled by the thermal barrier hex.

Technical Reference(s): OTA-RK-00018 Addendum 41A, OTA-RK-00022 Addendum 72B and 73A

References to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # ___X___
Modified Bank # ______
New _______

Question History: Last NRC Exam ____N/A________

Question Cognitive Level:
Memory or Fundamental Knowledge _____

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

Comprehension or Analysis __X___

10 CFR Part 55 Content:
55.41.10

Comments:

Examination Outline Cross-reference:	Level	RO		
	Tier #	2		
	Group #	1		
	K/A #	004K5.27		
	Importance Rating	2.6		
Knowledge of the operational implications of the following concepts as they apply to the CVCS: Reason for nitrogen purge of CVCS.				

Question #29

During mechanical degasification of the Volume Control Tank (VCT), VCT pressure must be maintained between 15 and 70 psig.

Which ONE (1) of the following describes the reason for this requirement?

A. Maintains proper back pressure on RCP seals and maintains an inert environment in the CVCS.

B. Reduces oxygen in the waste gas header during plant shutdown and ensures sufficient NPSH for the charging pumps.

C. Adjusts hydrogen concentration in RCS and causes a controlled crud burst.

D. Maintains proper RCP seal operation and prevents lifting the VCT relief valve.

Answer: D

Explanation:

A.Incorrect. Nitrogen helps maintain an inert environment, a back pressure of 15 psig is for RCP operation not for the seals regardless of RCP operation.
B.Incorrect. Nitrogen Charging pumps do not require a cover gas pressure for NPSH, after RCPs are secured VCT pressure is lowered to 5psig to prevent oxygen intrusion.
C.Incorrect. Nitrogen is added to the VCT to scrub H2, the pressure band is for RCP operation and relief setpoint.
D.Correct.

Technical Reference(s): OTN-BG-00004 Add04, VCT – Nitrogen Atmosphere – Rapid and Mechanical Degasification

References to be provided to applicants during examination: None

Learning Objective: T61.011 6, LP-11, Obj AA

Question Source: Bank # ______
Modified Bank # ______
New __X_____

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

Question History: Last NRC Exam ____________

Question Cognitive Level:
Memory or Fundamental Knowledge __X___
Comprehension or Analysis ____

10 CFR Part 55 Content:
55.41.10

Comments:

Examination Outline Cross-reference:	Level	RO		
	Tier #	2		
	Group #	1		
	K/A #	005K5.09		
	Importance Rating	3.2		
Knowledge of the operational implications of the following concepts as they apply the RHRS: Dilution and boration considerations.				

Question #30

During refueling operations, RHR flow rate must be greater than or equal to 1000 gpm.

Which ONE (1) of the following is the reason for this minimum flow?

This ensures…

A. adequate flow through the RHR pumps to prevent pump damage.

B. an indication of flow on the lower end of the flow meter.

C. there is adequate flow to prevent boron stratification.

D. RHR vortexing will not occur.

Answer: C

Explanation:

A.Incorrect. These limits are 500 as a minimum anytime and 1700 for more than 2 hours and 15 minutes.
B.Incorrect. There is no minimum flow to ensure indication.
C.Correct.
D.Incorrect. Level to prevent vortexing is not a set amount but is a variable amount based on reactor vessel level..

Technical Reference(s): OSP-ZZ-00001

References to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ____N/A________

Question Cognitive Level:
Memory or Fundamental Knowledge __X___

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

Comprehension or Analysis ______

10 CFR Part 55 Content:
55.41.10

Comments:

Examination Outline Cross-reference:	Level	RO		
	Tier #	2		
	Group #	1		
	K/A #	006A1.08		
	Importance Rating	2.8		
Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ECCS controls including: Temperature, high motor and bearing.				

Question #31

Given the following conditions:

- WPA tagging was removed from 'A' SI pump.
- When performing Post Maintenance Testing, the 'A' SI pump was started and stopped twice in a 5 minute period.

What is the required time the pump must be idle before a subsequent start can be attempted AND the reason for this time limit?

A. 45 minutes; Prevent rotor damage due to excessive cyclic stresses on the shaft.

B. 15 minutes; Prevent rotor damage due to excessive cyclic stresses on the shaft.

C. 45 minutes; Prevent overheating of the windings due to high starting currents.

D. 15 minutes; Prevent overheating of the windings due to high starting currents.

Answer: C

Explanation:

A. Incorrect. Wait time has no effect on cyclic stresses the pump shaft will experience.
B. Incorrect. 15 min would be correct for the pump to be at operating temperature.
C. Correct.
D. Incorrect. See a and b.

Technical Reference(s): OTN-EM-00001

References to be provided to applicants during examination: None

Learning Objective: T61.GFES, Motors and Generators, Obj 9

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam __

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

Question Cognitive Level:

Memory or Fundamental Knowledge __X__

Comprehension or Analysis _____

10 CFR Part 55 Content:

55.41.10

Comments:

Examination Outline Cross-reference:	Level	RO		
	Tier #	2		
	Group #	1		
	K/A #	007K3.01		
	Importance Rating	3.3		
Knowledge of the effect that a loss or malfunction of the PRTS will have on the following: Containment				

Question #32

Given the following conditions:

- Callaway was at 100% power when a Reactor Trip and SI occurred.
- Following the SI actuation the following conditions were observed:
    - Containment radiation monitors showed a rapid rise in rad levels.
    - Containment pressure started rising.
    - Containment humidity spiked to 100%.
    - 60E CTMT SUMP A/B LEV HI actuated.
    - 60F CTMT SUMP C/D LEV HI actuated.
    - 34D PRT TEMP HI is actuated.
    - 34E PRT PRESS HI clear.
    - 34D PRT LVL HI is actuated.

Assuming NO operator action, the failure of which ONE (1) of the following would result in these conditions?

A. ‘B’ RCP #1 seal

B. BBPCV455A, ‘A’ PORV

C. Inner reactor vessel flange o-ring

D. CRDM pressure housing causing a LOCA

Answer: B

Explanation:

A.Incorrect. A seal failure would cause PRT conditions to rise after leakoff is isolated, but would not cause containment radiation, pressure or humidity to rise.
B.Correct.
C.Incorrect. The inner vessel flange leak would be directed to the RCDT. The outer o-ring would prevent the RCS from depressurizing and causing a Rx trip or SI.
D.Incorrect. A CRDM pressure housing failure would cause containment parameters to rise, but

would not impact the PRT.

Technical Reference(s): OTA-RK-00018 ADD34E

References to be provided to applicants during examination: None

Learning Objective: T61.0110 6, LP-09, Obj B

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ____________

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X___

10 CFR Part 55 Content:
55.41.10

Comments:

Examination Outline Cross-reference:	Level	RO		
	Tier #	2		
	Group #	1		
	K/A #	008A3.03		
	Importance Rating	3.0		
Ability to monitor automatic operation of the CCWS, including: All flow rate indications and the ability to evaluate the performance of this closed-cycle cooling system.				

Question #33

Which ONE (1) of the following lists the control interlock signals that will cause Radwaste Component Cooling Water Isolation Valves, EG HV-70A and 70B, to automatically close?

A. CISA, low-low level in CCW Train "A" surge tank, or Low flow

B. SIS, low-low level in CCW Train "A" surge tank, or High flow

C. SIS, low-low level in CCW Train "B" surge tank, or High flow

D. CISA, low-low level in CCW Train "B" surge tank, or Low flow

Answer: C

Explanation:

A.Incorrect. EGHV 70A and 70B are 'B' train components and would not close on 'A' surge tank lvl, low flow or CIS A.
B.Incorrect. EGHV 70A and 70B are 'B' train components and would not close on 'A' surge tank lvl,
C.Correct.
D.Incorrect. CIS A would not close the EGHV70A &70B.

Technical Reference(s): dwg E-23EG08

References to be provided to applicants during examination: None

Learning Objective: T61.0110 6, LP-09, Obj B

Question Source: Bank # __ Wolf Creek _Q35 ML042080215____
Modified Bank # ______
New ___ _____

Question History: Last NRC Exam ____N/A_______

Question Cognitive Level:
Memory or Fundamental Knowledge __X___
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41.7

Comments:

Examination Outline Cross-reference:	Level	RO		
	Tier #	2		
	Group #	1		
	K/A #	010K2.01		
	Importance Rating	3.0		
Knowledge of bus power supplies to the following: PZR heaters				

Question #34

Which ONE (1) of the following is the correct power supply to the Group 'A' Pressurizer Backup Heaters?

A. PG 21

B. PG 22

C. PG 24

D. PG 25

Answer: A

Explanation:

A..Correct.
B.Incorrect. PG22 is the power supply to the Group B Pzr BU heaters.
C.Incorrect. PG24 powers Group C cycling heaters.
D.Incorrect. PG25 is a 480V load center in the Fuel Bldg.

Technical Reference(s): dwg E-21PG06

References to be provided to applicants during examination: None

Learning Objective: T61.0110 6, LP-09, Obj B

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ___________

Question Cognitive Level:
Memory or Fundamental Knowledge __X___
Comprehension or Analysis _____

10 CFR Part 55 Content:
55.41.7

Comments:

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

Examination Outline Cross-reference:	**Level**	**RO**		
	Tier #	2		
	Group #	1		
	K/A #	012K4.02		
	Importance Rating	3.9		
Knowledge of RPS design feature(s) and/or interlock(s) which provide for the following: Automatic reactor trip when RPS setpoints are exceeded for each RPS function; basis for each.				

Question #35

Which ONE (1) of the following automatic Reactor Trips will protect the reactor from the Departure from Nucleate Boiling Ratio (DNBR)?

A. Overpower ΔT

B. Overtemperature ΔT

C. Pressurizer Pressure - High

D. Power Range Neutron Flux – High Positive Rate

Answer: B

Explanation:

A.Incorrect. OPΔT is a reactor trip, but protects integrity of the fuel.
B.Correct
C.Incorrect. Pzr Press high is a reactor trip, but protects rcs integrity, low pressure protects against DNBR.
D.Incorrect. High positive rate is a reactor trip, but protects against an ejected rod.

Technical Reference(s): Tech Specs Basis 3.3.1

References to be provided to applicants during examination: None

Learning Objective: T61.0110 6, LP-27, Obj C

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ___________

Question Cognitive Level:
Memory or Fundamental Knowledge __X___
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41.7
Comments:

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

Examination Outline Cross-reference:	**Level**	**RO**		
	Tier #	2		
	Group #	1		
	K/A #	013K4.11		
	Importance Rating	3.2		
Knowledge of ESFAS design feature(s) and/or interlock(s) which provide for the following: Vital power load control				

Question #36

The plant is at 100% power with all systems in their normal lineups. Annunciator 14A, S/U XFMR LOCKOUT, alarms due to failure of the Startup Transformer (SUT).

Which ONE (1) of the following occurs as a result of the event?

A. A blackout load shed occurs to both safety and non-safety loads on NB02. When the ‘B’ EDG output breaker closes selected loads will be sequenced on.

B. A blackout load shed occurs to the safety loads on NB01. When the ‘A’ EDG output breaker closes selected loads will be sequenced on.

C. A blackout load shed occurs to the non-safety loads on NB02. When the ‘B’ EDG high speed relay is actuated all shed loads will be sequenced on.

D. A blackout load shed occurs to both safety and non-safety loads on NB01. When the ‘A’ EDG high speed relay is actuated all shed loads will be sequenced on.

Answer: A

Explanation:

A Correct.
B.Incorrect The SUT powers NB02, NB01 is powered from the ‘B’ Safeguards transformer.
C.Incorrect. A blackout load shed occurs to both safety and non-safety, and the HSR relay is a permissive for the EDG output breaker to shut.
D.Incorrect. NB01 is the non affected bus.

Technical Reference(s): dwg, E-21005 and E-23NG01

References to be provided to applicants during examination: None

Learning Objective: T61.0110 6, LP-51, Obj B and D

Question Source: Bank # ______
Modified Bank # __X____
New ________

Question History: Last NRC Exam ___________

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

10 CFR Part 55 Content: 55.41.7

Comments:

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

Examination Outline Cross-reference:	**Level**	**RO**		
	Tier #	2		
	Group #	1		
	K/A #	022K2.01		
	Importance Rating	3.0		
Knowledge of power supplies to the following: Containment cooling fans.				

Question #37

The following plant conditions exist:

- The unit is at 100% power with all systems aligned normally.
- The switchyard is in its preferred lineup with Ring Bus Breakers 52-2 and 52-3 CLOSED.
- A lockout occurs on 345KV Swyd Bus 'B'.

Which ONE (1) of the following describes the response of the Containment Cooling Fans?

A. All fans CONTINUE to RUN in the PRESELECTED speed.

B. A & C fans are SHIFTED to the PRESELECTED speed by the shutdown sequencer.

C. B & D fans DE-ENERGIZE and are RESTARTED in FAST speed by the shutdown sequencer.

D. A & C fans DE-ENERGIZE and are RESTARTED in SLOW speed by the shutdown sequencer.

Answer: D

Explanation:

A Incorrect. Incorrect because A & C de-energize.
B.Incorrect. Incorrect because shutdown sequencer does shift fan speeds.
C.Incorrect. Incorrect because B & D fans remain running.
D.Correct.

Technical Reference(s): T61.0110 6 LP-40 and T61.0110 6 LP-1

References to be provided to applicants during examination: None

Learning Objective: T61.0110 6, LP-40, Obj D

Question Source: Bank # ___Callaway 2002 NRC Exam___
Modified Bank # ______
New ______

Question History: Last NRC Exam ___N/A________

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

10 CFR Part 55 Content: 55.41.7

Comments:

Examination Outline Cross-reference:	Level	RO		
	Tier #	2		
	Group #	1		
	K/A #	028K6.01		
	Importance Rating	2.6		
Knowledge of the effect of a loss or malfunction on the following will have on the HRPS: Hydrogen recombiners.				

Question #38

The following plant conditions exist:

- Currently in E-1 after a large break LOCA from 100% power.
- ‘A’ Hydrogen Recombiner is tagged out of service.
- ‘B’ Hydrogen Recombiner trips on overcurrent after it is placed in service.
- Current containment hydrogen concentration is 4.5% and rising slowly.
- The TSC recommends using the Hydrogen Purge System to reduce containment hydrogen concentration.

Which ONE (1) of the following describes how the hydrogen concentration will be reduced in containment?

A. The containment is vented to the Normal Fuel and Aux Building exhaust upstream of the Filter Adsorber Unit and makeup is provided by Service Air.

B. The containment is vented directly to the Unit Vent and makeup air is provided by Instrument Air.

C. The containment is vented to the Emergency Exhaust System upstream of the Emergency Filter Adsorber Unit and makeup air is provided by Instrument Air.

D. The containment is vented directly to the Unit Vent and makeup air is provided by Service Air.

Answer: C

Explanation:

A.Incorrect. Incorrect because makeup air comes from IA.
B.Incorrect. Incorrect because the containment would be vented through the emergency exhaust units, not the unit vent.
C.Correct.
D.Incorrect. Incorrect because of vent location and wrong makeup air source.

Technical Reference(s): T61.0110 6 LP-40, OTN-GS-00001

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

References to be provided to applicants during examination: None

Learning Objective: T61.0110 6, LP-40, Obj J

Question Source: Bank # ______
Modified Bank # __ ____
New ___X____

Question History: Last NRC Exam ___________

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

10 CFR Part 55 Content: 55.41.8

Comments:

Examination Outline Cross-reference:	**Level**	**RO**		
	Tier #	2		
	Group #	1		
	K/A #	026A2.04		
	Importance Rating	3.9		
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 operations: Failure of spray pump.				

Question #39

The following plant conditions exist:

- A large break LOCA has occurred from 100% power.
- E-1, Loss of Reactor or Secondary Coolant, has just been entered.
- ‘A’ Containment Spray Pump tripped after starting.
- Containment Pressure is 15 PSIG and SLOWLY LOWERING.
- RWST level is 45% and SLOWLY LOWERING.
- All four (4) Containment Coolers are in service.

Which ONE (1) of the following describes the impact to the Containment Safety Function and the required actions in accordance with E-1?

A. Iodine and/or fission products in containment will exceed limits.
Maintain ‘B’ Containment Spray pump in service.

B. Iodine and/or fission products in containment will exceed limits.
Secure the ‘B’ Containment Spray pump.

C. Iodine and/or fission products in containment will not exceed limits.
Maintain ‘B’ Containment Spray pump in service.

D. Iodine and/or fission products in containment will not exceed limits.
Secure the ‘B’ Containment Spray pump.

Answer: C

Explanation:

A. Incorrect. Only one train of CS is req’d to reduce Iodine and Fission Products below limits.
B. Incorrect. Per table in E-1 if containment pressure is >4.5 psig and lowering 1 CS pump must be in service.
C. Correct.
D. Incorrect. Per table in E-1 if containment pressure is >4.5 psig and lowering 1 CS pump must be in service.

Technical Reference(s): T61.0110 6 LP-18, E-1, FSAR 6.5.2

References to be provided to applicants during examination: None

Learning Objective: T61.0110 6, LP-18, Obj A

Question Source: Bank # ______
Modified Bank # __ ____
New ___X____

Question History: Last NRC Exam ___________

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

10 CFR Part 55 Content: 55.41.8

Comments:

Examination Outline Cross-reference:	Level	RO		
	Tier #	2		
	Group #	1		
	K/A #	039A4.01		
	Importance Rating	2.9		
Ability to manually operate and/or monitor in the control room: Main steam supply valves.				

Question #40

Following a small break LOCA, E-0, Reactor Trip or Safety Injection, was entered due to a reactor trip. The main turbine failed to automatically trip and the operator attempted a manual turbine trip that was also unsuccessful.

Which ONE (1) of the following immediate actions should the operator perform FIRST?

A. Place both EHC pumps in Pull-To-Lock using CH HIS-1A and CH HIS-1B.

B. Manually run back the turbine using the Load Limit Potentiometer.

C. Dispatch an operator to locally trip the turbine at the Front Standard.

D. Fast Close the MSIVs using AB HS-79 / AB HS-80.

Answer: D

Explanation:

A Incorrect. This will cause a turbine trip, but E-0 does not address operation of EHC pumps.
B Incorrect. This would shut the Control Valves, but has no effect on Stop Valves.
C Incorrect. This would take too long and is not addressed in the immediate actions.
D Correct.

Technical Reference(s): E-0

References to be provided to applicants during examination: None

Learning Objective: T61.003D, D-4, Obj C

Question Source: Bank # ___L2458___
Modified Bank # ______
New ___ _____

Question History: Last NRC Exam ___________

Question Cognitive Level:

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

Memory or Fundamental Knowledge _X____
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41.7

Comments:

Examination Outline Cross-reference:	Level	RO		
	Tier #	2		
	Group #	1		
	K/A #	039A1.06		
	Importance Rating	3.0		
Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MRSS controls including: Main steam pressure				

Question #41

The following Callaway plant conditions exist:

- Mode 3
- RCS pressure is 1000 psig
- 'A' Atmospheric Steam Dump has failed open

Which ONE (1) of the following describes the condition of the MSIVs and reason for closure?

	MSIVs Closed	Signal
A.	A	Steam Pressure Rate
B.	A,B,C,D	Low Steam Line Pressure
C.	A	Low Steam Line Pressure
D.	A,B,C,D	Steam Pressure Rate

Answer: D

Explanation:

A Incorrect. Low steam pressure rate on 2 of 3 pressure inst on any S/G will close all MSIVs.
B Incorrect. Below P-11 (1970 psig) low steam pressure is blocked.
C Incorrect. Below P-11 (1970 psig) low steam pressure is blocked.
D Correct.

Technical Reference(s): T61.0110 6, LP-20, DWG 7250D64 sht 7

References to be provided to applicants during examination: None

Learning Objective: T61.0110 6, LP-20, Obj F

Question Source: Bank # ______
Modified Bank # __ ____
New ___X____

Question History: Last NRC Exam ___________

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X___

10 CFR Part 55 Content: 55.41.7

Comments:

Examination Outline Cross-reference:	Level	RO		
	Tier #	2		
	Group #	1		
	K/A #	059A2.04		
	Importance Rating	2.9		
Ability to (a) predict the impacts of the following malfunctions or operations on the MFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Feeding a dry S/G.				

Question #42

Callaway is currently performing actions in FR-H.1, Response to a Loss of Heat Sink, following an ATWS. The following plant conditions exist:

- All AFW has been lost
- RCS Bleed and Feed is in progress
- Ctmt pressure is 4.5 psig rising slowly
- CETC are reading 650°F and rising slowly
- SG levels
  - SG A WR level 9%
  - SG B WR level 8%
  - SG C WR level 12%
  - SG D WR level 7%

The 'B' Condensate pump has just been made available to feed steam generators.

(1) What is the appropriate feeding rate to the steam generators and
(2) What is the possible effect of feeding a dry steam generator?

A. (1) Feed ANY SG at the maximum rate.
(2) Excessive thermal stress on SG U-tubes resulting in failure.

B. (1) Feed ONLY 'C' SG at the maximum rate.
(2) Water hammer at J-tubes on the feedwater ring resulting in reduced feed.

C. (1) Feed ANY SG at <40,000 lbm/hr.
(2) Excessive thermal stress on SG U-tubes resulting in failure.

D. (1) Feed ONLY 'C' SG at <40,000 lbm/hr.
(2) Water hammer at J-tubes on the feedwater ring resulting in reduced feed.

Answer: A

Explanation:

A Correct.
B Incorrect. All SGs are dry so any SG can be fed, Water hammer damage could happen, but is not credited.
C Incorrect. Would be a correct feeding rate if CETCs were lowering.
D Incorrect. Water hammer damage could happen, but is not credited.

Technical Reference(s): FR.H-1

References to be provided to applicants during examination: None

Learning Objective: T61.003D, D-4, Obj C

Question Source: Bank # ___ ___
Modified Bank # __ ____
New ___X____

Question History: Last NRC Exam ___________

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X___

10 CFR Part 55 Content: 55.41.7

Comments:

Examination Outline Cross-reference:	Level	RO		
	Tier #	2		
	Group #	1		
	K/A #	061K5.03		
	Importance Rating	2.6		
Knowledge of the operational implications of the following concepts as the apply to the AFW: Pump head effects when control valve is shut.				

Question #43

Callaway has just had a manual reactor trip due to loss of one (1) Main Feed Pump while greater than 80% power. All systems have operated correctly.

If the I/P converter for ALHV0006, TDAFP Flow Control Valve to 'D' SG, FAILS to 100% air output, how would the pump discharge head change for each AFW pump?

	TDAFP	**'A' MDAFP**	**'B' MDAFP**
A.	Stable	Lower	Rise
B.	Rise	Stable	Lower
C.	Lower	Rise	Rise
D.	Rise	Lower	Stable

Answer: B

Explanation:

A Incorrect. TDAFP pump head would rise due to ALHV0006 going closed.
B Correct.
C Incorrect. ALHV0005 would open to maintain 300gpm flow to the 'D' SG causing 'B'MDAFP pump head to lower; 'A' MDAFP pump head would not change do to no effects to its flow stream.
D Incorrect. ALHV0005 would open to maintain 300gpm flow to the 'D' SG causing 'B'MDAFP pump head to lower.

Technical Reference(s): T61.003D, LP-25, Obj D

References to be provided to applicants during examination: None

Learning Objective: T61.003D, LP-25, Obj D

Question Source: Bank # ___ ___
Modified Bank # __ ____
New ___X____

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

Question History: Last NRC Exam _____N/A______

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X___

10 CFR Part 55 Content: 55.41.14

Comments:

Examination Outline Cross-reference:	Level	RO		
	Tier #	2		
	Group #	1		
	K/A #	062K1.04		
	Importance Rating	3.7		
Knowledge of the physical connections and/or cause-effect relationships between the ac distribution system and the following systems: Off-site power sources.				

Question #44

The plant is in Mode 1 at 100% power with all systems in their normal lineups.
A lockout occurs on the Startup Transformer.

Which ONE (1) of the following statements correctly describes what occurs first?

A. Both the NB02 normal and alternate feeder breakers receive an undervoltage trip.

B. An automatic Reactor Trip and Turbine Trip actuate due to the loss of all Reactor Coolant Pumps.

C. An automatic Reactor Trip and Turbine Trip actuate due to loss of power to NB02.

D. Both emergency diesels NE01 and NE02 start due to undervoltage condition.

Answer: A

Explanation:

A Correct. Loss of SUT results in an UV condition on NB02 which sends a trip signal to both normal and alternate NB02 feeder breakers.
B Incorrect. In Mode 1 100% the SUT only powers XNB02. Below 25% the SUT powers PA01 and PA02 which causes all RCPs to trip and a Reactor Trip.
C Incorrect. Loss of NB02 would cause perturbations, but would not cause a Reactor Trip.
D Incorrect. NE02 would start due to the UV on NB02, but NB01 does not lose power, and NE01 would not start.

Technical Reference(s): E-22NF01

References to be provided to applicants during examination: None

Learning Objective: T61.0110, LP-6, Obj D

Question Source: Bank # ____R12216__

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

Modified Bank # __ ____
New _______

Question History: Last NRC Exam ___________

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X___

10 CFR Part 55 Content: 55.41.7

Comments:

Examination Outline Cross-reference:	Level	RO		
	Tier #	2		
	Group #	1		
	K/A #	063 2.2.42		
	Importance Rating	3.9		
Ability to recognize system parameters that are entry-level conditions for Technical Specifications.				

Question #45

Which ONE (1) of the following NK01 bus electrical lineups meets the requirements for continuous operation while in Mode 1?

	Norm Charger NK21	Alt Charger NK25	Battery NK11
A.	Disconnected	Disconnected	Connected
B.	Disconnected	Connected From NG Bus	Connected
C.	Connected	Disconnected	Disconnected
D.	Disconnected	Connected From PG Bus	Connected

Answer: B

Explanation:

A Incorrect. Lineup not allowed by Tech Specs for operability.
B Correct.
C Incorrect. Lineup not allowed by Tech Specs for operability.
D Incorrect. Lineup not allowed by Tech Specs for operability.

Technical Reference(s): Tech Specs 3.8.4

References to be provided to applicants during examination: None

Learning Objective: T61.0110, LP-6, Obj G

Question Source: Bank # ___X_ __
Modified Bank # __ ____
New ___ ______

Question History: Last NRC Exam ___________

Question Cognitive Level:
Memory or Fundamental Knowledge _____

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41.7

Comments:

Examination Outline Cross-reference:	Level	RO		
	Tier #	2		
	Group #	1		
	K/A #	063K3.01		
	Importance Rating	3.7		
Knowledge of the effect that a loss or malfunction of the DC electrical system will have on the following: ED/G.				

Question #46

NE02, 'B' Emergency Diesel Generator, is operating in parallel with an off-site source.

What is the immediate effect on Diesel Generator 'B' by the loss of NK04?

A. Diesel will continue to run, the diesel can be stopped from the local STOP PB but not the Main Control Board STOP switch, and only the vital engine shutdowns are enabled.

B. Diesel will continue to run, the diesel can be stopped from the local STOP PB but not the Main Control Board STOP switch, and the vital engine shutdowns are disabled.

C. Diesel will continue to run, the diesel can not be stopped from the local STOP PB or the Main Control Board STOP switch, and the vital engine shutdowns are disabled.

D. Diesel will immediately trip due to the vital engine shutdowns being actuated.

Answer: C

Explanation:

A Incorrect. The diesel cannot be stopped from either control switch and no protective trips are enabled.
B Incorrect. The diesel cannot be stopped from either control switch and no protective trips are enabled.
C Correct.
D Incorrect. The diesel will not immediately trip.

Technical Reference(s): E-23KJ03A, 3B

References to be provided to applicants during examination: None

Learning Objective: T61.0110, LP-3, Obj B

Question Source: Bank # ____ __
Modified Bank # __ ____
New ___X____

Question History: Last NRC Exam ___________

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41.7

Comments:

Examination Outline Cross-reference:	**Level**	**RO**		
	Tier #	2		
	Group #	1		
	K/A #	064A4.09		
	Importance Rating	3.2		
Ability to manually operate and/or monitor in the control room: Establishing power from the ring bus (to relieve ED/G).				

Question #47

At 10:00 Callaway experienced a loss of off-site power.
At 10:30 Off-site power has been restored and is ready to power safety related busses.

The RO is performing OTN-NE-0001A, Addendum 006, Transferring Bus NB01 from NE01 to Normal or Alternate Source, to restore normal off-site power to NB01. The synchroscope is placed in the Main Feeder Position and is rotating slowly in the SLOW direction. Incoming voltage is 4185 VAC NB01 and bus voltage is 4130 VAC.

What actions must the RO take to parallel the EDG with the off-site source?

A. Lower EDG speed, Lower EDG voltage.

B. Raise EDG speed, Raise EDG voltage.

C. Lower EDG speed, Raise EDG voltage.

D. Raise EDG speed, Lower EDG voltage.

Answer: C

Explanation:

A Incorrect. The EDG voltage is supplying NB01, by lowering voltage the incoming voltage would be greater than 50V above bus voltage.
B Incorrect. By raising EDG speed, the synchroscope would rotate faster in the slow direction, the OTN has the synchroscope rotate slowly in the fast direction.
C Correct.
D Incorrect. By raising EDG speed, the synchroscope would rotate faster in the slow direction, the OTN has the synchroscope rotate slowly in the fast direction.

Technical Reference(s): OTN-NE-0001A, Add 6

References to be provided to applicants during examination: None

Learning Objective: T61.0110, LP-3, Obj B

Question Source: Bank # ____ __
Modified Bank # __ ____
New ___X____

Question History: Last NRC Exam ___________

Question Cognitive Level:
Memory or Fundamental Knowledge __ ___
Comprehension or Analysis __X___

10 CFR Part 55 Content: 55.41.7

Comments:

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

Examination Outline Cross-reference:	**Level**	**RO**		
	Tier #	2		
	Group #	1		
	K/A #	064K4.05		
	Importance Rating	2.8		
Knowledge of ED/G system design feature(s) and/or interlock(s) which provide for the following: Incomplete-start relay.				

Question #48

An emergency start has occurred on NE02 due to a Safety Injection.

The High Speed Relay (HSR) senses a RPM of 450.

Which ONE (1) of the following will NOT occur?

A. Run Time Meter enabled

B. Jacket Water Keep Warm Pump stops

C. Air Shutoff Relay de-energizes the starting air solenoids

D. Automatic operation of Rocker Arm Prelube Pump enabled

Answer: A

Explanation:

A Correct. HSR energizes at 471 rpm – this would enable the run time meter.
B Incorrect. Accomplished when the Low Speed Relay is energized at 125 rpm.
C Incorrect. Occurs at 85 rpm.
D Incorrect. Accomplished when the Low Speed Relay is energized at 125 rpm.

Technical Reference(s): T61.0110 LP-3

References to be provided to applicants during examination: None

Learning Objective: T61.0110, LP-3, Obj K

Question Source: Bank # ___ __
Modified Bank # __ ____
New ___X____

Question History: Last NRC Exam ___________

Question Cognitive Level:

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

Memory or Fundamental Knowledge __X___
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41.7

Comments:

Examination Outline Cross-reference:	Level	RO		
	Tier #	2		
	Group #	1		
	K/A #	073K1.01		
	Importance Rating	3.6		
Knowledge of the physical connections and/or cause effect relationships between the PRM system and the following systems: Those systems served by PRMs.				

Question #49

A Hi Hi Radiation signal from BM RE-52, Steam Generator Blowdown Discharge Pumps Discharge Radiation Monitor, will automatically close which ONE (1) of the following valves?

A. BM HV-38, S/G 'D' Blowdown Nuclear Sampling System Lower Isolation Valve

B. BM HV-21, S/G 'C' Blowdown Nuclear Sampling System Upper Isolation Valve

C. BM HV-65, S/G 'A' Sample Isolation Valve

D. BM HV-6, S/G 'B' Blowdown Nuclear Sampling System Line Downstream Isolation Valve

Answer: D

Explanation:

A Incorrect. BMHV 38 is closed by a SGBSIS, BMRE52 generates a BSPIS
B Incorrect. BMHV 21 is closed by a SGBSIS, BMRE52 generates a BSPIS
C Incorrect. BMHV 65 is closed by a SGBSIS, BMRE52 generates a BSPIS
D Correct.

Technical Reference(s): OTO-SA-00001, M-22BM01

References to be provided to applicants during examination: None

Learning Objective: T61.0110, LP-12, Obj D

Question Source: Bank # ___0110120D09A___
Modified Bank # __ ____
New ___ ____

Question History: Last NRC Exam ___________

Question Cognitive Level:

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

Memory or Fundamental Knowledge __X___
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41.7

Comments:

Examination Outline Cross-reference:	Level	RO		
	Tier #	2		
	Group #	1		
	K/A #	073 2.2.23		
	Importance Rating	3.1		
Process Radiation Monitoring (PRM) System: Ability to track Technical Specification limiting conditions for operations.				

Question #50

Which ONE (1) of the following conditions associated with Process Radiation Monitors (PRM) would require a separate Equipment Out Of Service Log (EOSL) entry, assuming all control room log entries have been made?

A. OSP-SP-00001, Radiation Monitor Source Check, will be performed on BM RE-10B, Radiation Building Vent Gas Channel.

B. The RM-80 for GT RE-33, CTMT Purge Gas Detector, has failed, projected to be replaced on same shift.

C. A filter change will be performed on GT RE-31, CTMT Atmosphere Rad Monitor.

D. No EOSL entry is required for PRMs if control room log entries are made.

Answer: B

Explanation:

A Incorrect. A surveillance where the equip can be returned to service would not require an EOSL.
B Correct.
C Incorrect. A filter change is specifically excluded from an EOSL entry.
D Incorrect. EOSL entries are required for PRMs, even if control room log entries are made.

Technical Reference(s): ODP-ZZ-00002

References to be provided to applicants during examination: None

Learning Objective: T61.0110, LP-74, Obj B

Question Source: Bank # __ _ __
Modified Bank # __ ____
New ___X____

Question History: Last NRC Exam ___________

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

Question Cognitive Level:
Memory or Fundamental Knowledge __X___
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41.10

Comments:

Examination Outline Cross-reference:	Level	RO		
	Tier #	2		
	Group #	1		
	K/A #	076A2.02		
	Importance Rating	2.7		
Ability to (a) predict the impacts of the following malfunctions or operations on the SWS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Service water header pressure.				

Question #51

Callaway is in Mode 4 performing a plant cooldown.

- 'A' and 'C' Service Water pumps are in operation.
- 13:00 'A' ESW Pump was placed in operation in a manual alignment.
- 13:45 'B' ESW Pump was placed in operation in a manual alignment to support RHR operations.

(1) At what time MUST one (1) Service Water pump be secured?

(2) What is the reason for securing the Service Water pump?

	(1)	(2)
A.	13:50	Excessive cavitation
B.	13:50	High bearing loading
C.	13:55	Excessive cavitation
D.	13:55	High bearing loading

Answer: B

Explanation:

A Incorrect. Cavitation would be a problem with high flow rates, not low flow rates.
B Correct.
C Incorrect. For pump start limitations the ESW pump is considered at operating temperature after 10 minutes of operation.
D Incorrect. For pump start limitations the ESW pump is considered at operating temperature after 10 minutes of operation and cavitation would be a problem with high flow rates not low flow rates.

Technical Reference(s): OTN-EF-00001 and OTN-EA-00001

References to be provided to applicants during examination: None

Learning Objective: T61.0110, LP-4, Obj E

Question Source: Bank # __ _ __
Modified Bank # __ ____
New ___X____

Question History: Last NRC Exam ___________

Question Cognitive Level:
Memory or Fundamental Knowledge __X___
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41.10

Comments:

Examination Outline Cross-reference:	Level	RO		
	Tier #	2		
	Group #	1		
	K/A #	076 2.4.20		
	Importance Rating	3.8		
Service Water System: Knowledge of the operational implications of EOP warnings, cautions, and notes.				

Question #52

When performing ECA-0.0, Loss of All AC Power, all loads that would be sequenced on by the LOCA sequencer are taken to Pull-To-Lock with the exception of one load.

(1) Which load is left in NORMAL and (2) what is the reason for this?

A. (1) MDAFP
(2) Provide feedwater to the steam generators.

B. (1) CCP
(2) Provide RCP seal injection.

C. (1) ESW pump
(2) Provide diesel generator cooling.

D. (1) CCW pump
(2) Provide cooling to safety related components.

Answer: C

Explanation: Major loads are placed in PTL to defeat automatic loading of large loads on the AC emergency bus. Defeating automatic shutdown or SI loading of as many large loads as practical is intended to avoid potential overload of the energized AC emergency bus.

A Incorrect. Providing for a heat sink is important, but the TDAFP should be running to provide feedwater. Also, see above.
B Incorrect. CCPs normally start at step 0 of the LOCA sequencer, but RCPs are isolated. Also, see above.
C Correct.
D Incorrect. CCW pumps provide cooling to safety related components during LOCA sequencer actuation. Also, see above.

Technical Reference(s): ECA-0.0

References to be provided to applicants during examination: None

Learning Objective: T61.003D, D-22, Obj L

Question Source: Bank # __ ___
Modified Bank # __ ____
New ____X___

Question History: Last NRC Exam ___________

Question Cognitive Level:
Memory or Fundamental Knowledge __X___
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41.10

Comments:

Examination Outline Cross-reference:	Level	RO		
	Tier #	2		
	Group #	1		
	K/A #	078A4.01		
	Importance Rating	3.1		
Ability to manually operate and/or monitor in the control room: Pressure gauges.				

Question #53

Callaway is at 100% steady state when a loss of Instrument Air occurs.

The Reactor Operator checks KA PI-40, Inst Air Hdr Press, and reports that it is indicating 75 psig.

Which ONE (1) of the following statements describe the IMMEDIATE effect of this failure on the Reactor Coolant System (RCS)?

A. Pressurizer pressure will lower due to both pressurizer PORVs failing open.

B. Pressurizer level will lower due to BG HCV-182, CVCS CHG PMPS To Regen HX HCV, failing shut.

C. Pressurizer level will rise due to BG FCV-124, NCP Disch HDR FCV, failing open.

D. Pressurizer pressure will rise due to both pressurizer Spray Valves failing shut.

Answer: C

Explanation:

A Incorrect. The PORVs are pilot actuated valves and fail closed on a loss of electrical power, air does not affect them.
B Incorrect. BGHCV182 fails open on a loss of IA, PZR lvl would rise.
C Correct.
D Incorrect. The Pzr spray valves fail shut on a loss of IA, but they would not be the cause of pressure rise since they are normally shut at 100% power.

Technical Reference(s): OTO-KA-00001

References to be provided to applicants during examination: None

Learning Objective: T61.0110, LP-11, Obj B

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

Question Source: Bank # ____
Modified Bank # __ ____
New __X_____

Question History: Last NRC Exam ___________

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X___

10 CFR Part 55 Content: 55.41.7

Comments:

Examination Outline Cross-reference:	**Level**	**RO**		
	Tier #	2		
	Group #	1		
	K/A #	078A3.01		
	Importance Rating	3.1		
Ability to monitor automatic operation of the IAS: Air pressure.				

Question #54

The Secondary Operator reports that the inlet solenoid valve for the IN-SERVICE Instrument Air Dryer train has failed closed.

Which ONE (1) of the following statements describes the system response?

A. Service Air Header Isolation Valve KA PV-11 closes.

B. All air compressors are running.

C. The First Backup air compressor loads.

D. Inlet/outlet valves for the Standby Air Dryer train fail open.

Answer: D

Explanation:

A Incorrect. KA PV-11 senses pressure downstream of KA PV-11, not Instrument Air.
B Incorrect. The pressure to operate Air Compressors is from Service air, which is not affected by an IA dryer failure.
C Incorrect. The pressure to operate Air Compressors is from Service air, which is not affected by an IA dryer failure.
D Correct.

Technical Reference(s): OTO-KA-00001

References to be provided to applicants during examination: None

Learning Objective: T61.0110, LP-14, Obj B

Question Source: Bank # ___L6789__
Modified Bank # __ ____
New ___ _____

Question History: Last NRC Exam ___________

Question Cognitive Level:

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

Memory or Fundamental Knowledge __X___
Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41.7

Comments:

Examination Outline Cross-reference:	**Level**	**RO**		
	Tier #	2		
	Group #	1		
	K/A #	103K3.02		
	Importance Rating	3.8		
Knowledge of the effect that a loss or malfunction of the containment system will have on the following: Loss of containment integrity under normal operations.				

Question #55

Which ONE (1) of the following conditions represents a loss of containment integrity per Technical Specifications?

A. With RCS temperature 250°F, Containment Shutdown Purge is initiated to cool containment.

B. On exiting the Reactor Building after an at power entry the outer Containment Personnel Hatch could not be closed, the inner door was verified locked.

C. With RCS temperature 180°F, an inner containment isolation valve failed to isolate during a slave relay test.

D. With reactor power at 25%, an electrician opens the outer Containment Personnel Hatch without equalizing pressure.

Answer: A

Explanation:

A Correct.
B Incorrect. TS 3.6.2 states if one door is inoperable the other must be locked.
C Incorrect. TS 3.9.4 ctmt only required to be operable less than Mode 4 if core alts are in progress. At 180 F Callaway would not be moving fuel.
D Incorrect. Equalizing pressure is a procedural requirement, not TS requirement.

Technical Reference(s): Tech Specs

References to be provided to applicants during examination: None

Learning Objective: T61.0110, LP-40, Obj O

Question Source: Bank # ____
Modified Bank # ______
New __X_____

Question History: Last NRC Exam ___________

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

Question Cognitive Level:

Memory or Fundamental Knowledge __X___

Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41.7

Comments:

Examination Outline Cross-reference:	Level	RO		
	Tier #	2		
	Group #	2		
	K/A #	002A1.04		
	Importance Rating	3.9		
Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RCS controls including: Subcooling Margin.				

Question #56

Given the following conditions:

- A Steam Generator (SG) Tube Rupture has occurred in SG 'B'.
- The Crew is commencing an RCS depressurization IAW E-3, Steam Generator Tube Rupture, Step 16, Depressurize RCS To Minimize Break Flow And Refill PZR.
- RCS conditions at the start of depressurization are:
  - RCS Pressure 1500 psig
  - RCS Subcooling 60° subcooled
  - Pressurizer (PZR) Level 5%
  - SG 'B' NR Level 62%
  - SG 'B' Pressure 1020 psig

The RCS depressurization will be stopped under which ONE (1) of the following set of conditions?

A. PZR level – 70%
SG 'B' NR level – 68%
RCS subcooling - 35°

B. RCS pressure – 1310 psig
SG 'B' pressure – 1010 psig
RCS subcooling - 28°

C. PZR level – 50%
SG 'B' NR level – 75%
RCS subcooling - 35°

D. PZR level – 8%
RCS pressure – 1000 psig
SG 'B' pressure – 1010 psig

Answer: B

Explanation:
Parameters to stop depressurization are:
RCS pressure less than ruptured SG pressure and pzr level >9%
or
Pzr level >74%
or
RCS subcooling <30 deg

A. Incorrect
B. Correct
C. Incorrect
D. Incorrect

Technical Reference(s): E-3

References to be provided to applicants during examination: None

Learning Objective: T61.003D 6, LP-17, Obj J

Question Source: Bank # ______
Modified Bank # ______
New ____X___

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:
55.41.10

Comments:

Examination Outline Cross-reference:	Level	RO		
	Tier #	2		
	Group #	2		
	K/A #	011K3.01		
	Importance Rating	3.2		
Knowledge of the effect that a loss or malfunction of the PZR LCS will have on the following: CVCS.				

Question #57

The plant is in MODE 1, 100% reactor power.

All CVCS and Pressurizer (PZR) level control systems are in AUTOMATIC, with the Normal Charging Pump (NCP) in service.

The following plant parameters exist:

- LETDOWN HX OUTLET FLOW (BG FI-132) 120 GPM
- CHARGING HDR FLOW (BG FI-121A) 132 GPM
- TOTAL RCP SEAL INJECTION FLOW (BG FI-215) 32 GPM

The controlling PZR level control channel fails HIGH to an indicated 100% level.

Which ONE (1) of the following describes the response of BG FCV-0124, NCP Flow Control Valve? (Assume NO operator action is taken.)

A. BG FCV-0124 throttles CLOSED until BG FI-121A indicates 45 GPM

B. BG FCV-0124 throttles OPEN until BG HV-8109, NCP RECIRC HV, receives a close signal

C. BG FCV-0124 throttles OPEN until ANN 41F, NCP FLOW HI/LO, alarms

D. BG FCV-0124 throttles CLOSED until BG HV-8109, NCP RECIRC HV, receives an open signal

Answer: A

Explanation:

A. Correct
B. Incorrect. BGFCV124 throttles closed on a high pressurizer level.
C. Incorrect. BGFCV124 throttles closed on a high pressurizer level.
D. Incorrect. BGFCV124 throttles closed to 45 gpm to ensure proper seal injection flow which is above the auto opening setpoint of 41 gpm for 8109.

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

Technical Reference(s): Drawing 7250D64, Sheet 11

References to be provided to applicants during examination: None

Learning Objective: T61.0110 6, LP-11, Obj B-18

Question Source: Bank # __R12234____
Modified Bank # ______
New _______

Question History: Last NRC Exam ____N/A________

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X___

10 CFR Part 55 Content:
55.41.7

Comments:

Examination Outline Cross-reference:	Level	RO		
	Tier #	2		
	Group #	2		
	K/A #	015K6.03		
	Importance Rating	2.6		
Knowledge of the effect of a loss or malfunction on the following will have on the NIS: Component interconnections.				

Question #58

The plant is in MODE 2 commencing warmup of the main turbine.

Which ONE (1) of the following is a direct result of a loss of Vital AC Instrument Bus NN02?

A. Charging Pumps Suction Swaps from the VCT to the RWST

B. Source Range High Flux Reactor Trip

C. Intermediate Range High Flux Reactor Trip

D. Loss of Cooling Water to 'A' Air Compressor

Answer: C

Explanation:

A. Incorrect. Only occurs on loss of NN01 or NN04.
B. Incorrect. SR trips are blocked following the reactor startup and would not be enabled during warmup of the turbine.
C. Correct
D. Incorrect. Only occurs on loss of NN01.

Technical Reference(s): OTO-NN-00001

References to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # __L6807____
Modified Bank # ______
New _______

Question History: Last NRC Exam ____N/A________

Question Cognitive Level:
Memory or Fundamental Knowledge __X___
Comprehension or Analysis ____

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

10 CFR Part 55 Content:
55.41.7

Comments:
Changed one distractor.

Examination Outline Cross-reference:	Level	RO		
	Tier #	2		
	Group #	2		
	K/A #	017A2.02		
	Importance Rating	3.6		
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.				

Question #59

Given the following conditions:

- A Loss of Coolant Accident has occurred.
- The Crew is currently in E-1, Loss of Reactor or Secondary Coolant, performing Step 11, Check Ultimate Heat Sink-Normal.
- Incore Thermocouples indicate that RCS temperature is 1215°F.
- RCS Subcooling indicates 125°F Superheat.

Which ONE (1) of the following will be the preferred order of actions taken to address plant conditions?

A. Depressurize SGs to Depressurize the RCS
Establish ECCS Flow to the RCS
Start RCPs and Open RCS Vent Paths

B. Establish ECCS Flow to the RCS
Start RCPs and Open RCS Vent Paths
Depressurize SGs to Depressurize the RCS

C. Start RCPs and Open RCS Vent Paths
Establish ECCS Flow to the RCS
Depressurize SGs to Depressurize the RCS

D. Establish ECCS Flow to the RCS
Depressurize SGs to Depressurize the RCS
Start RCPs and Open RCS Vent Paths

Answer: D

Explanation:

A. Incorrect. Order of actions are not correct IAW FR-C.1, Response to Inadequate Core Cooling.

B. Incorrect. Order of actions are not correct IAW FR-C.1, Response to Inadequate Core Cooling.
C. Incorrect. Order of actions are not correct IAW FR-C.1, Response to Inadequate Core Cooling.
D. Correct

Technical Reference(s): FR-C.1

References to be provided to applicants during examination: None

Learning Objective: T61.003D 6, LP-25, OBJs A and K

Question Source: Bank # ______
Modified Bank # ______
New ____X___

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:
55.41.10

Comments:

Examination Outline Cross-reference:	**Level**	**RO**		
	Tier #	2		
	Group #	2		
	K/A #	033A3.02		
	Importance Rating	2.9		
Ability to monitor automatic operation of the Spent Fuel Pool Cooling System including: Spent fuel leak or rupture.				

Question #60

Given the following conditions:

- The Plant is in Mode 6.
- EC-V995, Fuel Transfer Tube Isolation Valve, is open for fuel movement.
- A leak develops on the Refueling Pool at EC-V7129, RFP To RCDT Pump.
- Containment Normal Sump levels are rising.

Which ONE (1) of the following automatic actions could occur?

A. A Fuel Building Isolation Signal due to high rad on RE-27 or 28.

B. Trip of the running Spent Fuel Pool Cooling Pump.

C. Trip of the running Spent Fuel Cleanup Pump aligned to the Refueling Pool.

D. Makeup from Essential Service Water to the Spent Fuel Pool.

Answer: B

Explanation:

A. Incorrect. Leak is in the Containment, not in the Fuel Building, which is monitored by 27 and 28.
B. Correct
C. Incorrect. Cleanup pumps do not have trip signals on low pool level.
D. Incorrect. Makeup from ESW is manual only, no automatic feature.

Technical Reference(s): OTA-RK-00022, Addendum 76D

References to be provided to applicants during examination: None

Learning Objective: T61.0110 6, LP-24, Obj D

Question Source: Bank # ______
Modified Bank # ______

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

New ____X___

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:
55.41.10

Comments:

Examination Outline Cross-reference:	Level	RO		
	Tier #	2		
	Group #	2		
	K/A #	035K1.09		
	Importance Rating	3.8		
Knowledge of the physical connections and/or cause-effect relationships between the S/Gs and the following systems: RCS.				

Question #61

A reactor had been operating at a constant power level for the last two weeks when a loss of all AC power occurred, causing a reactor trip.

Natural circulation reactor coolant flow developed and stabilized 30 minutes after the reactor trip.

Which ONE (1) of the following combinations of INITIAL reactor power and POST-TRIP steam generator pressure will result in the HIGHEST stable natural circulation flow rate 30 minutes after the reactor trip?

	INITIAL REACTOR POWER	POST-TRIP STEAM GENERATOR PRESSURE
A.	100%	1100 psia
B.	25%	1100 psia
C.	100%	1000 psia
D.	25%	1000 psia

Answer: C

Explanation:

A. Incorrect. Lower delta T due to larger Tsat from higher SG pressure, thus lower natural circ flow.
B. Incorrect. Lower decay heat and lower delta T due to larger Tsat from higher SG pressure.
C. Correct. The largest amount of natural circ flow will be developed by having the most decay heat available (100%) and the lowest SG pressure (lowest Tsat).
D. Incorrect. Lower decay heat results in less natural circ flow.

Technical Reference(s): Steam Tables

References to be provided to applicants during examination: Steam Tables

Learning Objective: T61.GFES 6, LP-45, Obj 24

Question Source: Bank # __P1591____
Modified Bank # ______
New _______

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:
55.41.14

Comments:

Examination Outline Cross-reference:	Level	RO		
	Tier #	2		
	Group #	2		
	K/A #	041 2.2.1		
	Importance Rating	4.5		
Equipment Control: Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity.				

Question #62

Given the following conditions:

- A Reactor Trip occurs from 100% at End of Life.
- An Estimated Critical Position (ECP) is calculated for 24 hours after the Reactor Trip.
- The following conditions were used in the ECP:
  - RCS temperature 557°F
  - RCS pressure 2235 psig
  - RCS boron concentration 225 ppm

Which ONE (1) of the following would result in the Control Rods at a higher position than calculated in the ECP when the Reactor goes critical?

A. BB PK-455A, PZR Press Master CTRL, is set at 7.0 TURNS.

B. Boron concentration is actually 215 ppm.

C. The startup is delayed by four (4) hours.

D. AB PK-507, Steam Hdr Press CTRL, is set at 7.4 TURNS.

Answer: D

Explanation:

A. Incorrect. This controller misposition will result in a higher RCS pressure but will have no affect on RCS reactivity, thus control rod position will not be affected.
B. Incorrect. Lower boron adds positive reactivity, thus rods will be further in, not out.
C. Incorrect. Xenon will be less, thus adding positive reactivity, rods will be further in, not out.
D. Correct. Steam dumps will be less open, RCS temp will be higher, adding negative reactivity, thus rods will have to be further out for criticality.

Technical Reference(s): OOA-RL-00004

References to be provided to applicants during examination: None

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

Learning Objective: T61.003A 6, LP-12, Obj D

Question Source: Bank # ______
Modified Bank # ______
New ____X___

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:
55.41.5

Comments:

Examination Outline Cross-reference:	Level	RO		
	Tier #	2		
	Group #	2		
	K/A #	072A4.01		
	Importance Rating	3.0		
Ability to manually operate and/or monitor in the control room: Alarm and interlock setpoint checks and adjustments.				

Question #63

The Radwaste Pipe Tunnel Area Radiation Monitor (ARM) has just caused ANNUN 62A, AREA RAD HIHI, to alarm in the Control Room.

Which ONE (1) of the following will alert the Control Board Monitor to subsequent ARM alarms (with the exception of Containment High Range ARM)?

A. ARM annunciator reflash with NO audible alarm

B. ARM annunciator reflash with audible alarm

C. RM-11, RADIATION MONITOR CONTROL PANEL, alarm

D. SD055A, CTRL PNL-AREA RADN MONITOR, alarm

Answer: B

Explanation:

A. Incorrect. ARM annunciator reflashes with audible alarm.
B. Correct
C. Incorrect. This panel will give alarm indication for process rad monitors, not area rad monitors.
D. Incorrect. Does give alarm status for area rad monitors but is located in the back of the control room, not in the At the Control Area where the CR operators are located..

Technical Reference(s):

References to be provided to applicants during examination: None

Learning Objective: T61.0110 6, LP-36, Obj C

Question Source: Bank # __L5480____
Modified Bank # ______
New _______

Question History: Last NRC Exam ____N/A________

Question Cognitive Level:

Memory or Fundamental Knowledge __X___

Comprehension or Analysis _____

10 CFR Part 55 Content:

55.41.12

Comments:

Bank question enhanced by changing 3 distractors.

Examination Outline Cross-reference:	Level	RO		
	Tier #	2		
	Group #	2		
	K/A #	075K4.01		
	Importance Rating	2.5		
Knowledge of circulating water system design feature(s) and interlocks(s) which provide for the following: Heat sink.				

Question #64

Outside air temperature has been below freezing for the past two weeks with the weather forecast predicting below freezing temperatures the next several days.

HSDA3102, COOL TWR DISTRIBUTION VLVS, is placed in FREEZE PROTECT.

Which ONE (1) of the following set of conditions would require operation of the cooling tower in AUTO-BYPASS?

(REFERENCE PROVIDED)

A. Outside Air Temperature 10°F, Basin Temperature 40°F

B. Outside Air Temperature 0°F, Basin Temperature 50°F

C. Outside Air Temperature -20°F, Basin Temperature 60°F

D. Outside Air Temperature -10°F, Basin Temperature 55°F

Answer: A

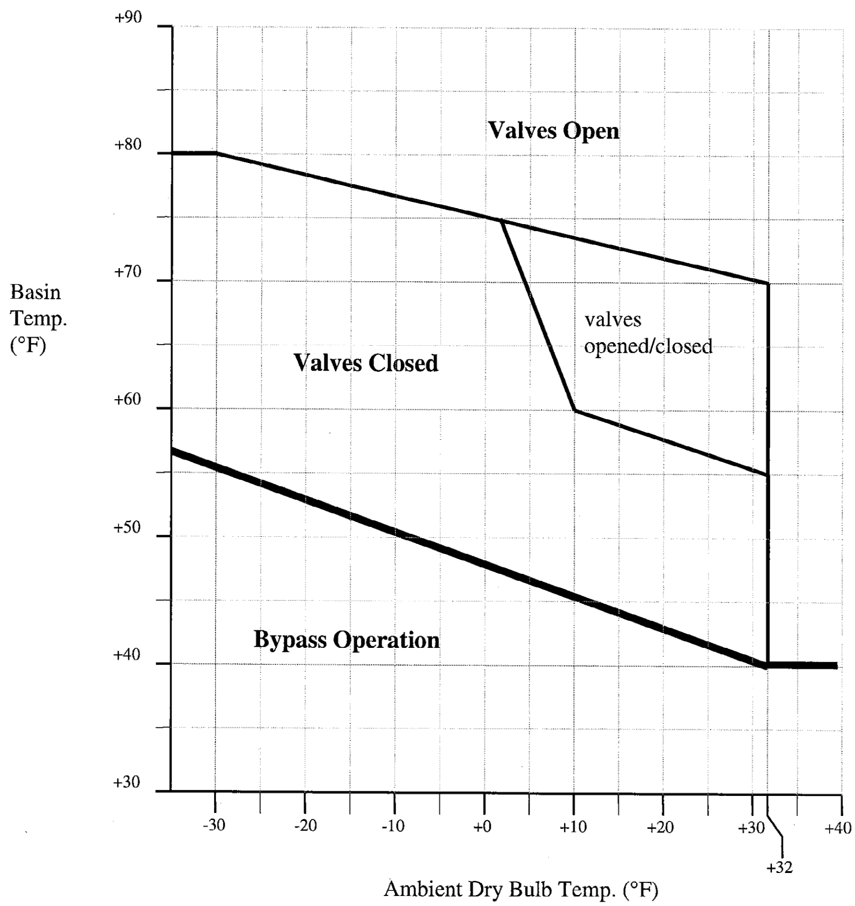
Explanation:

A. Correct
B. Incorrect. See Attachment below.
C. Incorrect. See Attachment below.
D. Incorrect. See Attachment below.

Attachment 1

Cooling Tower Freeze Protection Curve

Sheet 1 of 1



 CONTINUOUS USE

Technical Reference(s): OTN-DA-00001, Add 04

References to be provided to applicants during examination: None

Learning Objective: T61.0110 6, LP-4, Objs B and D

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

Question Source: Bank # __R12247____
Modified Bank # ______
New ________

Question History: Last NRC Exam ____N/A________

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X___

10 CFR Part 55 Content:
55.41.4

Comments:

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

Examination Outline Cross-reference:	**Level**	**RO**		
	Tier #	2		
	Group #	2		
	K/A #	086K5.03		
	Importance Rating	3.1		
Knowledge of the operational implication of the following concepts as they apply to the Fire Protection System: Effect of water spray on electrical components.				

Question #65

If the Main Transformer deluge is ____(1)______ actuated, the associated Main Transformer will _____(2)_________ with a potential hazard from ______(3)_________.

	(1)	(2)	(3)
A.	inadvertently	remain energized	exposed bushings
B.	automatically	de-energize	exposed surge capacitors
C.	automatically	remain energized	exposed bushings
D.	inadvertently	de-energize	exposed surge capacitors

Answer: A

Explanation:

A. Correct
B. Incorrect. Construction of surge capacitors do not present any hazard.
C. Incorrect. The deluge will only actuate for the main transformers if the transformer is de-energized.
D. Incorrect. The main transformer will not de-energize if the deluge is inadvertently actuated.

Technical Reference(s): FPP-ZZ-00007

References to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # ______
Modified Bank # ______
New ____X___

Question History: Last NRC Exam ____N/A________

Question Cognitive Level:
Memory or Fundamental Knowledge __X___
Comprehension or Analysis _____

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

10 CFR Part 55 Content: 55.41.4

Comments:

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

Examination Outline Cross-reference:	**Level**	**RO**		
	Tier #	3		
	Group #	1		
	K/A #	G2.1.13		
	Importance Rating	2.5		
Conduct of Operations: Knowledge of facility requirements for controlling vital/controlled access.				

Question #66

You are serving as an escort for a college tour group in the Protected Area when you are called to the Control Room. You were serving as an escort for 3 students.

Which One (1) of the following describes the action you must take before going to the Control Room?

A. Leave the students in the Field Office Conference Room and let them know that a Security Officer will come and get them.

B. Transfer your escort duties to the Shift Clerk, who is not currently escorting anyone.

C. Transfer your escort duties to the Field Supervisor, who is currently also escorting 3 students.

D. Give them directions to return to the Main Access Facility where security will let them out of the Protected Area.

Answer: B

Explanation:

A. Incorrect. They can not be left alone without an escort, even if they are staying together in the Conference Room.
B. Correct
C. Incorrect. This would have the FS escorting 6 visitors; the maximum allowed number is 5.
D. Incorrect. Visitors can not be allowed to move around the PA without being.

Technical Reference(s): APA-ZZ-01105

References to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # ______
Modified Bank # ______

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

New ___X____

Question History: Last NRC Exam ____N/A_________

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content:
55.41.10

Comments:

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

Examination Outline Cross-reference:	**Level**	**RO**		
	Tier #	3		
	Group #	1		
	K/A #	G2.1.20		
	Importance Rating	4.6		
Conduct of Operations: Ability to interpret and execute procedure steps.				

Question #67

Which ONE (1) of the following definitions best defines the word TRY, as it is used in the Emergency Operating Procedures (EOP)?

A. Make a one time effort to perform an evolution, and then continue to the next EOP step.

B. Do not continue in the EOP until the effort has been successful or the evolution is deemed not necessary.

C. To make a continued effort when success may not be immediately obtainable, while continuing in the EOP.

D. Effort only needs to be performed at the discretion of the Shift Manager.

Answer: C

Explanation:

A. Incorrect. You do continue to the next step, but the step bypassed will be performed when conditions allow.
B. Incorrect. You would only not continue if the procedure specifically said not to.
C. Correct
D. Incorrect. SM does not have this discretion in the EOPs.

Technical Reference(s): ODP-ZZ-00025

References to be provided to applicants during examination: None

Learning Objective: T61.003D 6, LP-1, Obj A12

Question Source: Bank # ___X___
Modified Bank # ______
New ________

Question History: Last NRC Exam ____N/A________

Question Cognitive Level:

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

Memory or Fundamental Knowledge	__X__
Comprehension or Analysis	_____

10 CFR Part 55 Content:

55.41.10

Comments:

Examination Outline Cross-reference:	Level	RO		
	Tier #	3		
	Group #	1		
	K/A #	G2.1.38		
	Importance Rating	3.7		
Conduct of Operations: Knowledge of the station's requirements for verbal communications.				

Question #68

While performing actions in E-3, Steam Generator Tube Rupture, the Control Room Supervisor asks the Balance of Plant Operator (BOP) to check intact Steam Generator narrow range levels greater than 7%.

Which ONE (1) of the following BOP responses would satisfy Callaway Plant Communication Guidelines?

A. "Control Room Supervisor, intact Steam Generator narrow range levels are 10%"

B. "Control Room Supervisor, intact Steam Generator narrow range levels are 50% and stable"

C. "Control Room Supervisor, intact Steam Generator narrow range levels are increasing"

D. "Control Room Supervisor, intact Steam Generator narrow range levels are greater than 7%"

Answer: B

Explanation:

A. Incorrect. No trend given.
B. Correct
C. Incorrect. No value given and word increase rather than rising used.
D. Incorrect. No trend given.

Technical Reference(s): ODP-ZZ-00001, Addendum 06

References to be provided to applicants during examination: None

Learning Objective: T61.0110 6, LP-66, Obj B-1

Question Source: Bank # __L7079____
Modified Bank # ______

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

New _______

Question History: Last NRC Exam ____N/A________

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content:
55.41.10

Comments:

Examination Outline Cross-reference:	**Level**	**RO**		
	Tier #	3		
	Group #	2		
	K/A #	G2.2.12		
	Importance Rating	3.7		
Equipment Control: Knowledge of surveillance procedures.				

Question #69

A Conditional Surveillance is a surveillance…

A. performed to satisfy a Technical Specification requirement.

B. that has a periodicity of one week or less.

C. that must be performed as a result of normal operational transients.

D. that is performed to satisfy a plant Mode Change requirement.

Answer: C

Explanation:

A. Incorrect. Would be considered a normal surveillance.
B. Incorrect. Would be considered a normal surveillance.
C. Correct
D. Incorrect. Would be considered a normal surveillance.

Technical Reference(s): APA-ZZ-00340

References to be provided to applicants during examination: None

Learning Objective: T61.003A 6, LP-22, Obj A

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ____N/A________

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content:
55.41.10

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

Comments:

Examination Outline Cross-reference:	**Level**	**RO**		
	Tier #	3		
	Group #	2		
	K/A #	G2.2.21		
	Importance Rating	2.9		
Equipment Control: Knowledge of pre- and post-maintenance operability requirements.				

Question #70

The Post-Maintenance Test (PMT) Program addresses testing after maintenance and modification work to ensure ALL of the following EXCEPT:

A. Equipment performs its design function when returned to service following maintenance activities.

B. All original deficiencies have been corrected following corrective maintenance activities.

C. No new deficiencies are created applicable to plant modifications or maintenance on plant systems.

D. Used to verify that a modification was installed per design and installed components meet minimum preliminary functional requirements.

Answer: D

Explanation:

A. Incorrect. Is a function of PMT.
B. Incorrect. Is a function of PMT.
C. Incorrect. Is a function of PMT.
D. Correct. Not given as a function of PMT in procedure.

Technical Reference(s): APA-ZZ-00322 and APA-ZZ-00322, Appendix E

References to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ____N/A________

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content:
55.41.10

Comments:

Examination Outline Cross-reference:	Level	RO		
	Tier #	3		
	Group #	3		
	K/A #	G2.3.5		
	Importance Rating	2.9		
Radiation Control: Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.				

Question #71

Which ONE (1) of the following would be a worker expectation when performing work inside the Radiological Controlled Area (RCA)?

A. Monitor your Secondary Monitoring Device every 20 minutes and exit the RCA prior to receiving a dose rate alarm.

B. Monitor your Secondary Monitoring Device every 15 minutes and exit the RCA prior to receiving a dose rate alarm.

C. Monitor your Secondary Monitoring Device every 15 minutes and exit the RCA prior to receiving a dose alarm.

D. Monitor your Secondary Monitoring Device every 20 minutes and exit the RCA prior to receiving a dose alarm.

Answer: C

Explanation:

A. Incorrect. Time of 20 minutes is wrong and action is wrong.
B. Incorrect. Leave before a dose alarm, not a dose rate alarm.
C. Correct
D. Incorrect. SMD should be monitored every 15 minutes.

Technical Reference(s): APA-ZZ-01004

References to be provided to applicants during examination: None

Learning Objective: T61.0110 6, LP-75, Obj C-7

Question Source: Bank # ______
Modified Bank # ______
New ____X___

Question History: Last NRC Exam ____N/A________

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content:
55.41.12

Comments:

Examination Outline Cross-reference:	Level	RO		
	Tier #	3		
	Group #	3		
	K/A #	G2.3.14		
	Importance Rating	3.4		
Radiation Control: Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.				

Question #72

In which ONE (1) of the following plant conditions could the Shift Manager waive the requirement to perform an Independent Verification when placing WPA tags on the Safety Injection (EM) System?

A. Area contamination level is 1500 dpm/100cm^2 beta-gamma.

B. The General Area Dose Rate is 30 mRem/Hr.

C. A radiation exposure of greater than 5 mRem is likely.

D. Surveys have NOT been taken in the area where the WPA will be placed.

Answer: B

Explanation:

A. Incorrect. No allowance given in procedure to waive IV based on contamination levels.
B. Correct. IVs can be waived if General Area Dose Rates are greater than 25.
C. Incorrect. Expected exposure must be more than 10.
D. Incorrect. If radiation levels are unknown, a survey would be required or constant HP coverage.

Technical Reference(s): ODP-ZZ-00310

References to be provided to applicants during examination: None

Learning Objective: T61.003A 6, LP-13, Obj B-3

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ____N/A________

Question Cognitive Level:

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content:
55.41.12
Comments:

Examination Outline Cross-reference:	**Level**	**RO**		
	Tier #	3		
	Group #	3		
	K/A #	G2.3.15		
	Importance Rating	2.9		
Radiation Control: Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.				

Question #73

Given the following conditions:

- Reactor Power is 20%
- A 0.5 gpm Steam Generator (SG) tube leak develops
- A HI HI alarm is received on BM RE-52, SG Blowdown Discharge Pumps Discharge Radiation Element
- The Crew enters OTO-BB-00001, Steam Generator Tube Leak

Which ONE (1) of the following will provide the quickest indication of the affected Steam Generator (SG)?

A. A local radiation survey of the steam lines.

B. The SG PORV Exhaust Monitors (AB RE-111/112/113/114).

C. An unexpected rise in SG feedwater flow with a stable SG level.

D. The SG Blowdown Discharge Pumps Discharge Radiation Element (BM RE-52).

Answer: A

Explanation:

A. Correct. Given as a means to identify the affected SG in OTO-BB-00001.
B. Incorrect. These monitors are SG specific but only monitor flow through the SG ASD. Nothing in the information given indicates any flow through the ASDs.
C. Incorrect. Reduced FW flow is an indication of a SG tube leak, not increased flow.
D. Incorrect. BM RE-52 is downstream of the blowdown system demineralizers and would not be useful in determining a SG tube leak.

Technical Reference(s): OTO-BB-00001

References to be provided to applicants during examination: None

Learning Objective:

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

Question Source: Bank # __WC 2009 Exam____
Modified Bank # ______
New _______

Question History: Last NRC Exam ____N/A________

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content:
55.41.11

Comments:

Examination Outline Cross-reference:	**Level**	**RO**		
	Tier #	3		
	Group #	4		
	K/A #	G2.4.37		
	Importance Rating	3.0		
Emergency Procedures/Plan: Knowledge of the lines of authority during implementation of the emergency plan.				

Question #74

An imminent airborne threat has been confirmed IAW OTO-SK-00002, Plant Security Event-Aircraft Threat.

Which ONE (1) of the following plant individuals will make the initial Radiological Emergency Response Plan (RERP) event declaration **AND** which plant individual will relieve this individual after the RERP organization becomes functional?

	Declares the Event	Relieves Individual Declaring Event
A.	Shift Manager (SM)	Emergency Duty Officer (EDO)
B.	Security Shift Supervisor (SSS)	Recovery Manager (RM)
C.	Shift Manager (SM)	Recovery Manager (RM)
D.	Security Shift Supervisor (SSS)	Emergency Duty Officer (EDO)

Answer: A

Explanation:

A. Correct
B. Incorrect. SM makes initial declaration.
C. Incorrect. SM will be relieved by the EDO.
D. Incorrect. SM makes initial declaration.

Technical Reference(s): EIP-ZZ-00101

References to be provided to applicants during examination: None

Learning Objective: T68.1020 6, LP-1, Obj A

Question Source: Bank # ______
Modified Bank # ______

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

New ____X___

Question History: Last NRC Exam ____N/A________

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content:
55.41.10

Comments:

Examination Outline Cross-reference:	Level	RO		
	Tier #	3		
	Group #	4		
	K/A #	G2.4.45		
	Importance Rating	4.1		
Emergency Procedures/Plan: Ability to prioritize and interpret the significance of each annunciator or alarm.				

Question #75

A Reactor trip and Safety Injection occurred from 100% power.

The following conditions now exist:

- RCS pressure — 1880 psig and stable
- RCS subcooling — 55° subcooled
- Pressurizer level — 25% and stable
- D Steam Generator (SG) Pressure — Depressurized
- A/B/C SG Pressures — 1000 psig and slowly lowering
- Total AFW flow to intact SGs — 400,000 lbm/hr
- Annun 88C, HI CTMT PRESS SI RX TRIP — LIT

Based on the above conditions, which ONE (1) of the following parameters will prevent a transition to ES-1.1, SI Termination, when the crew performs Step 6, Check IF ECCS Flow Should Be Reduced, in E-1, Loss of Reactor or Secondary Coolant ?

A. RCS pressure

B. RCS subcooling

C. SG pressures

D. Pressurizer level

Answer: D

Explanation:

A. Incorrect. Pressure is stable, which would allow a transition.
B. Incorrect. Subcooling is greater than the minimum required of 50.
C. Incorrect. No requirement for SG pressures to be stable.
D. Correct. Candidate must know the significance of Annun 88C, which is not a high level alarm (A or B). The setpoint for Annun 88C is 3.5 psig. This is the containment pressure at which adverse condition parameters must be used in the EOPs. Applying adverse ctmt parameters, the

crew cannot terminate SI based on PZR level (9% required for normal ctmt pressure / 29% required for adverse ctmt).

Technical Reference(s): E-1 and OTA-RK-00022, Add 88C

References to be provided to applicants during examination: None

Learning Objective: T61.003D 6, LP-8, Obj G

Question Source: Bank # ______
Modified Bank # ______
New ____X___

Question History: Last NRC Exam ____N/A________

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X___

10 CFR Part 55 Content:
55.41.10

Comments:

Examination Outline Cross-reference:	Level	SRO		
	Tier #	1		
	Group #	1		
	K/A #	011EA2.09		
	Importance Rating	4.3		
Ability to determine or interpret the following as they apply to a Large Break LOCA: Existence of adequate natural circulation.				

Question #76

Given the following conditions:

- A large break Loss of Coolant Accident (LOCA) has occurred.
- The Crew is attempting to start a Reactor Coolant Pump (RCP) IAW ES-1.2, Post LOCA Cooldown and Depressurization, Step 20, Check RCP Status.
- RCP seal cooling flow was lost following the Safety Injection.
- The Reactor Operator reports the following parameters to the CRS:
  - Containment Pressure 8.5 psig lowering slowly
  - RCS Subcooling 40 deg subcooled
  - SG Pressures 500 psig and stable
  - RCS Hot Leg Temperatures 499 deg F and stable
  - Core Exit Thermocouples 510 deg F and stable
  - RCS Cold Leg Temperatures 490 deg F and stable

Which ONE (1) of the following describes the correct action and procedure usage to be taken by the Control Room Supervisor?

A. Raise Steam Dump Flow IAW ES-1.2

B. Transition back to E-1, Loss of Reactor or Secondary Coolant

C. Transition to ES-1.1, SI Termination, and Terminate Safety Injection

D. Continue monitoring Natural Circulation Conditions using EOP Addendum 1, Natural Circulation Verification

Answer: A

Explanation:

A. Correct. Natural circulation does not exist. Procedure directs to dump more steam.
B. Incorrect. There is no procedure transition from ES-1.2 to E-1.
C. Incorrect. ES-1.1 is not a transition from ES-1.2. SI equipment is secured by steps in ES-1.2 without transitioning to ES-1.1. ES-1.1 could be used to terminate SI equipment prior to entering ES-1.2 if certain plant conditions exist.

D. Incorrect. Natural circulation does not exist. Action needs to be taken to address loss of natural circulation.

Technical Reference(s): OTA-KJ-00121, Addendum 4C; OTO-NB-00001

References to be provided to applicants during examination: None

Learning Objective: T61.0110 6, LP-3, Obj J

Question Source: Bank # ______
Modified Bank # ______
New ____X___

Question History: Last NRC Exam ____N/A________

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X___

10 CFR Part 55 Content:
55.43.5

Comments:

Examination Outline Cross-reference:	**Level**	**SRO**		
	Tier #	1		
	Group #	1		
	K/A #	040AA2.01		
	Importance Rating	4.7		
Ability to determine and interpret the following as they apply to the Steam Line Rupture: Occurrence and location of a steam line rupture from pressure and flow indications.				

Question #77

Given the following conditions:

- A Reactor Trip and Safety Injection occurred from 100%.
- A Containment Spray Signal has actuated.
- The Balance of Plant Operator (BOP) reports the following to the CRS:
    - Steam Generator A/B/C/D Pressure 425 psig and lowering
    - Steam Generator A/B/C/D Flow 4E6 lbm/hr and lowering
    - The MSIVs have failed open and cannot be shut

Which ONE (1) of the following procedures will be used by the Control Room Supervisor to stabilize the plant based on the conditions given above?

A. E-2, Faulted Steam Generator Isolation

B. ECA-2.1, Uncontrolled Depressurization of All Steam Generators

C. E-1, Loss of Reactor or Secondary Coolant

D. ES-1.1, SI Termination

Answer: B

Explanation:

A. Incorrect. E-2 would not be used unless a SG pressure starts to rise and SI termination is not in progress IAW ECA-2.1.
B. Correct.
C. Incorrect. E-1 would be the procedure to use if conditions were satisfied and a transition made to E-2.
D. Incorrect. ECA-2.1 contains SI termination criteria to be used; ES-1.1 would not be used to terminate SI for this condition.

Technical Reference(s): ECA-2.1

References to be provided to applicants during examination: None

NRC Site-Specific Written Examination
Callaway Plant
Senior Reactor Operator

Learning Objective: T61.003D 6, D-16, Obj B

Question Source: Bank # ______
Modified Bank # ______
New ____X___

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:
55.43.5

Comments:

Examination Outline Cross-reference:	Level	SRO		
	Tier #	1		
	Group #	1		
	K/A #	062 2.1.39		
	Importance Rating	4.3		
Loss of Nuclear Svc Water: Conduct of Operations: Knowledge of conservative decision making practices.				

Question #78

Given the following conditions:

- Reactor Power is 100%.
- Diesel Generator 'A', NE01, is in service due to a loss of XNB01.
- Train 'A' of Essential Service Water (ESW) is in service to support NE01 operation.
- PEF01A, ESW Pump 'A', trips
- When realigning ESW to Service Water, the breaker for EF HV-25, Service Wtr/ESW Train A Cross Connect, trips open.
- NE01 jacket water temperature is 185°F and rising at 1°/minute.
- The Diesel Room Operator reports Annunciator 4C, Jacket Water Temp High, is lit on the local alarm panel.

Which ONE (1) of the following directions should be given to the crew by the CRS?

A. Unload Diesel Generator 'A', NE01, and secure in 15 minutes, then go to OTO-NB-00004, LOOP To NB01/NB02 With EDG Paralleled.

B. Unload Diesel Generator 'A', NE01, and secure in 15 minutes, then go to OTO-NB-00001, Loss of Power To NB01.

C. Trip Diesel Generator 'A', NE01, and go to OTO-NB-00001, Loss of Power To NB01.

D. Trip Diesel Generator 'A', NE01, and go to OTO-NB-00004, LOOP To NB01/NB02 With EDG Paralleled.

Answer: C

Explanation:

A. Incorrect. This can be done, but requires manual valve operation in the field. Also, not allowed to cross connect ESW trains in Mode 1.
B. Incorrect. With temperature rise given, diesel will trip within 12 minutes.

C. Correct. CRS should direct to trip the DG as it will trip within 12 minutes based on the jacket water temperature rise given. The prudent and conservative action to take if an automatic trip signal is being approached is to take manual action and not rely on the automatic signal.
D. Incorrect. The plant is designed to operate with only one train of ESW.

Technical Reference(s): OTA-KJ-00121, Addendum 4C

References to be provided to applicants during examination: None

Learning Objective: T61.0110 6, LP-3, Obj J

Question Source: Bank # ______
Modified Bank # ______
New ____X___

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:
55.43.5

Comments:

Examination Outline Cross-reference:	Level	SRO		
	Tier #	1		
	Group #	1		
	K/A #	065AA2.04		
	Importance Rating	2.7		
Ability to determine and interpret the following as they apply to the Loss of Instrument Air: Typical conditions which could cause a compressor trip (e.g., high temperature).				

Question #79

Given the following conditions:

- Reactor Power is 100%.
- CKA01A, 'A' Air Compressor, is out of service for overhaul.
- CKA01B, 'B' Air Compressor, trips due to low oil pressure.
- The in service Central Chiller, SGB01A, trips and the Standby Central Chiller, SGB01B, can not be started.

Which of the following directions will the Control Room Supervisor (CRS) give to the Crew **AND** what is the status of CKA01C, 'C' Air Compressor?

	CRS DIRECTION	CKA01C STATUS
A.	Trip the Reactor and enter E-0, Reactor Trip or Safety Injection	Will continue to run with ESW providing cooling water flow
B.	Commence a Turbine Load Reduction to 90% IAW OTO-MA-00008, Rapid Load Reduction	Will trip on high air discharge temperature
C.	Commence a Turbine Load Reduction to 90% IAW OTO-MA-00008, Rapid Load Reduction	Will continue to run with ESW providing cooling water flow
D.	Trip the Reactor and enter E-0, Reactor Trip or Safety Injection	Will trip on high air discharge temperature

Answer: D

Explanation:

A. Incorrect. C air compressor is cooled by central chilled water. A and B air compressors are cooled by ESW.
B. Incorrect. If a load reduction is performed, it would be to 80%, not 90%.

C. Incorrect. C air compressor is cooled by central chilled water. A and B air compressors are cooled by ESW. Also, if a turbine load reduction can be done, OTO-KA-00001 directs a load reduction to 80%, not 90%.
D. Correct

Technical Reference(s): OTO-KA-00001

References to be provided to applicants during examination: None

Learning Objective: T61.0110 6, LP-14, Obj B and H

Question Source: Bank # ______
Modified Bank # ______
New ____X___

Question History: Last NRC Exam ____N/A________

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X___

10 CFR Part 55 Content:
55.43.5

Comments:

Examination Outline Cross-reference:	Level	SRO		
	Tier #	1		
	Group #	1		
	K/A #	077 G2.4.5		
	Importance Rating	4.3		
Generator Voltage and Electric Grid Disturbances: Knowledge of the organization of the operating procedures network for normal, abnormal, and emergency evolutions.				

Question #80

Given the following:

- Annunciator 134D, Switchyard Voltage High Low, is in alarm.
- The Power Supervisor has reported to the CRS that a grid disturbance is occurring.
- The BOP reports Main Generator output to be 1296 MWe and Main Generator reactive load to be 175 MVARS IN.

Which ONE (1) of the following describes the condition of the Main Generator and any direction to be given by the CRS?

(REFERENCE PROVIDED)

The Main Generator is operating…

A. Within the limits of the Main Generator Capability Curve; no further action is required.

B. Outside the limits of the Main Generator Capability Curve; raise hydrogen gas pressure in accordance with OTN-CC-00001, Main Generator Gas System and Gas System Auxiliaries.

C. Outside the limits of the Main Generator Capability Curve; raise Main Generator Excitation in accordance with OTA-RK-00026 Add134D, Switchyard Voltage High Low.

D. Outside the limits of the Main Generator Capability Curve; lower Main Generator Excitation in accordance with OTA-RK-00026 Add134D, Switchyard Voltage High Low.

Answer: C

Explanation:

A Incorrect. Per capability curve Callaway exceeds MVAR IN limit for 25 KV.
B Incorrect. OTA does not give guidance to raise or lower H2 pressure; if gas press was less than 60 psig, raising pressure would allow more MVARS IN, just not enough.
C Correct.
D Incorrect. Lowering voltage would cause more MVARS IN, making the situation worse.

Technical Reference(s): OTA-RK-00026 Add 134 D

References to be provided to applicants during examination: None

Learning Objective: T61.GFES, Motors and Generators, OBJ 20

Question Source: Bank # ___STP 2009 SRO Exam_
Modified Bank # __ ____
New _______

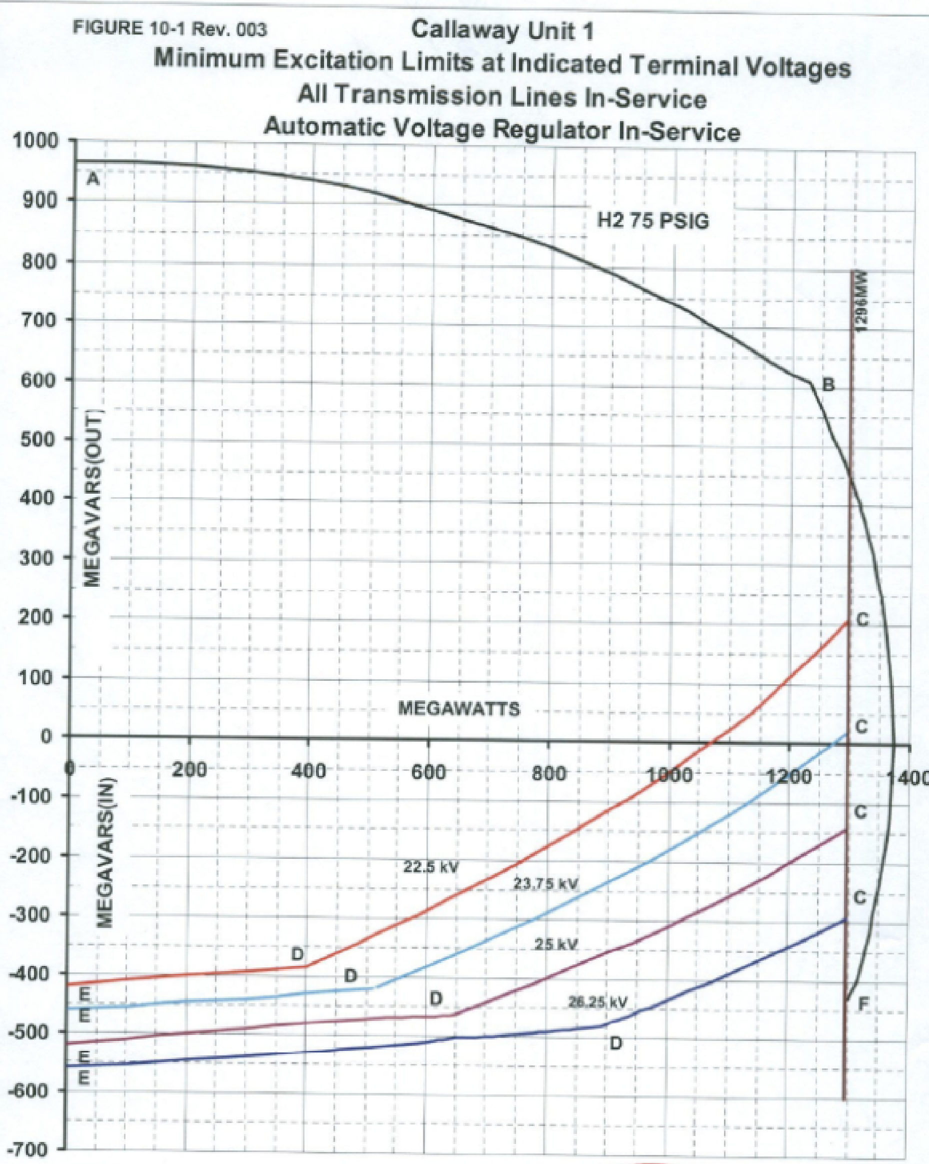
Question History: Last NRC Exam ___________

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X___

10 CFR Part 55 Content: 55.43.5

Comments:
:

FIGURE 10-1 Rev. 003



Minimum Excitation Limits to provide adequate stability margin should the most critical outlet line be outaged

Curve AB Limited by Field Heating
Curve BF Armature Heating
Curve CD Limited by Steady State Stability
Curve DE Limited by Regulator Limiter

JAN 2008 RECEIVED

Transmissions & Interconnections
1/9/2008

Examination Outline Cross-reference:	Level	SRO		
	Tier #	1		
	Group #	1		
	K/A #	W/E04 2.4.9		
	Importance Rating	4.2		
Emergency Procedures/Plan: Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.				

Question #81

Given the following conditions:

- A Reactor Trip and Safety Injection have occurred from 100%.
- Containment radiation is normal.
- Auxiliary Building radiation is slowing rising.
- The Crew has completed the actions of ECA-1.2, LOCA Outside Containment.
- RCS pressure is 1300 psig and lowering.

The Control Room Supervisor will direct the Crew to perform which ONE (1) of the following actions **AND** what is the basis for this direction?

	ACTION	BASIS
A.	Transition to ECA-1.1, Loss of Emergency Coolant Recirculation	Provide recovery actions since there will be no inventory in the containment sump
B.	Return to Step 1 of ECA-1.2 while monitoring Critical Safety Functions	Provide recovery actions since there will be no inventory in the containment sump
C.	Return to Step 1 of ECA-1.2 while monitoring Critical Safety Functions	Re-attempt to isolate the leak to preserve RWST inventory
D.	Transition to ECA-1.1, Loss of Emergency Coolant Recirculation	Re-attempt to isolate the leak to preserve RWST inventory

Answer: A

Explanation:

A. Correct

B. Incorrect. ECA-1.2 is not reperformed as action needs to be taken to preserve RWST inventory and there is no inventory in the sump for ECCS recirculation. These actions are provided in ECA-1.1.
C. Incorrect. This action is only directed if throttling of AFW is not successful and temperature continues to lower.
D. Incorrect. There is no action to perform ECA-1.2 a second time. The proper action is to transition to ECA 1.1 for actions to preserve RWST inventory.

Technical Reference(s): ECA-1.2, BD-ECA-1.2

References to be provided to applicants during examination: None

Learning Objective: T61.003D 6, LP-14, Obj. E

Question Source: Bank # ______
Modified Bank # ______
New ____X___

Question History: Last NRC Exam ____N/A________

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X___

10 CFR Part 55 Content:
55.43.5

Comments:

NRC Site-Specific Written Examination
Callaway Plant
Senior Reactor Operator

Examination Outline Cross-reference:	**Level**	**SRO**		
	Tier #	1		
	Group #	2		
	K/A #	024 2.1.25		
	Importance Rating	4.2		
Emergency Boration: Conduct of Operations: Ability to interpret reference materials, such as graphs, curves, tables, etc.				

Question #82

Given the following conditions:

- Callaway Plant is in a refueling.
- RCS temperature is 190°F.
- NB02 has been taken out of service for refueling periodic maintenance.
- ‘A’ Boric Acid Tank Temperature 67°F
- ‘A’ Boric Acid Tank Volume 3,000 gal
- ‘A’ Boric Acid Tank Boron Concentration 7,400 ppm
- ‘B’ Boric Acid Tank Temperature 64°F
- ‘B’ Boric Acid Tank Volume 9,800 gal
- ‘B’ Boric Acid Tank Boron Concentration 7,435 ppm

As the Control Room Supervisor, which ONE (1) of the following flow paths would you accept as meeting the required Boration Flow Path – Suction Source, in accordance with OSP-BG-0001A, Boron Injection Flow Paths Mode 4 Through 6?

(REFERENCES PROVIDED)

A. Attachment 3

B. Attachment 5

C. Attachment 7

D. Attachment 8

Answer: C

Explanation:

A. Incorrect. No power available to emergency borate valve BG HV-8104.
B. Incorrect. No power available to emergency borate valve BG HV-8104 and B BAT temperature is to low.
C. Correct
D. Incorrect. B BAT temperature is too low.

NRC Site-Specific Written Examination
Callaway Plant
Senior Reactor Operator

Technical Reference(s): OSP-BG-0001A, FSAR 16.1.2.1

References to be provided to applicants during examination: OSP-BG-0001A, Attachments 3, 5, 7, 8

Learning Objective: T61.0110 6, LP-11, Obj. T and V

Question Source: Bank # ______
Modified Bank # ______
New ____X___

Question History: Last NRC Exam ____N/A________

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X___

10 CFR Part 55 Content:
55.43.2

Comments:

Examination Outline Cross-reference:	Level	SRO		
	Tier #	1		
	Group #	2		
	K/A #	W/E02EA2.1		
	Importance Rating	4.2		
Ability to determine and interpret the following as they apply to the (SI Termination): Facility conditions and selection of appropriate procedures during abnormal and emergency operations.				

Question #83

The following conditions exist at Callaway:

- A small break LOCA has occurred while operating at 100%.
- The crew has transitioned to ES-1.1, SI Termination, and is in the process of establishing letdown IAW Step 14.

The Reactor Operator (RO) reports the following to the Control Room Supervisor (CRS):

- The MSIVs have just closed due to a slow rise in containment pressure.
- RCS Subcooling is 45°F and stable.
- Pressurizer Level is 20% and stable.
- RCS Pressure is 1800 psig and slowly lowering.

Which ONE (1) of the following directions will the CRS give to the crew based on the information provided by the RO?

A. Re-initiate Safety Injection from SB HS-27 & 28, SI Actuation Switches, and go to E-1, Loss of Reactor or Secondary Coolant.

B. Re-establish Boron Injection Header flow as necessary and go to E-1, Loss of Reactor or Secondary Coolant.

C. Continue in ES-1.1 while monitoring RCS subcooling. Go to E-1, Loss of Reactor or Secondary Coolant if RCS subcooling lowers to 30°F.

D. Restart the Safety Injection Pumps and go to E-1, Loss of Reactor or Secondary Coolant.

Answer: B

Explanation:

A Incorrect. SI is not reinitiated as this would just reestablish the conditions to terminate SI. Correct action is to restart SI equipment as necessary to address plant conditions and stabilize the plant.
B Correct. Proper action as per the foldout page of ES-1.1.
C Incorrect. The subcooling requirement per the foldout page is already meant due to being in adverse containment (MSIVs just closed). The 30 deg subcooling setpoint is for normal containment.
D Incorrect. RCS pressure is given as above the shutoff head of the SI pumps, thus they would not address the plant condition at this time.

Technical Reference(s): ES-1.1

References to be provided to applicants during examination: None

Learning Objective: T61.003D, LP-9, Obj H

Question Source: Bank # ____
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ___________

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X___

10 CFR Part 55 Content: 55.43.5

Comments:

Examination Outline Cross-reference:	**Level**	**SRO**		
	Tier #	1		
	Group #	2		
	K/A #	W/E09EA2.2		
	Importance Rating	3.8		
Ability to determine and interpret the following as they apply to the (Natural Circulation Operations): Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.				

Question #84

The following conditions exist at Callaway:

- Loss of Offsite AC power.
- NB01 and NB02 are energized by their diesels.
- 3 CRDM fans are operating.
- The crew has commenced a RCS depressurization IAW Step 13 of ES-0.2, Natural Circulation Cooldown.

The Reactor Operator reports the following to the Control Room Supervisor (CRS):

- RCS subcooling indicates 100°F.
- RVLIS (Pumps Off) indicates 110%.
- Pressurizer level is 12% and stable.

In Accordance With ES-0.2 which ONE (1) of the following is the correct action to be taken by the CRS?

A. Stop depressurization and transition to ES-0.3, Natural Circulation Cooldown With Steam Void in Vessel (With RVLIS)

B. Stop depressurization and transition to ES-0.4, Natural Circulation Cooldown With Steam Void in Vessel (Without RVLIS)

C. Stop depressurization and reestablish subcooling IAW ES-0.2

D. Actuate Safety Injection and go to E-0, Reactor Trip Or Safety Injection

Answer: C

Explanation:

A Incorrect. Procedure would only be considered for a transition if RVLIS is less than 100% or there were large variations in pressurizer level.
B Incorrect. Procedure would only be considered for a transition if RVLIS is less than 100% or there were large variations in pressurizer level.
C Correct.
D Incorrect. Would be correct if subcooling decreases to 30 deg or PZR level decreases to 6%.

Technical Reference(s): ES-0.2

References to be provided to applicants during examination: None

Learning Objective: T61.003D, D-7, Obj E and G

Question Source: Bank # ____
Modified Bank # ______
New ____X___

Question History: Last NRC Exam ___________

Question Cognitive Level:
Memory or Fundamental Knowledge ____
Comprehension or Analysis __X___

10 CFR Part 55 Content: 55.43.5

Comments:

Examination Outline Cross-reference:	Level	SRO		
	Tier #	1		
	Group #	2		
	K/A #	W/E15 2.2.37		
	Importance Rating	4.6		
Containment Flooding: Ability to determine operability and/or availability of safety related equipment.				

Question #85

A major LOCA has occurred while operating at 100% reactor power. The crew is currently in E-1, Loss of Reactor Or Secondary Coolant, responding to the LOCA.

The Shift Technical Advisor (STA) reports the following containment parameters associated with the Containment Critical Safety Function:

- Containment Pressure 25 psig and stable
- Containment Normal Sump Level 108 inches and stable
- Containment Radiation 3.2 rad/hour and slowly rising

Which ONE (1) of the following actions should be taken by the Control Room Supervisor to respond to the containment conditions reported by the STA?

A. Continue in E-1 as there are currently no challenges to the Containment Critical Safety Function

B. Go To FR-Z.1, Response to High Containment Pressure, to ensure that the containment design pressure is not exceeded

C. Go To FR-Z.2, Response to Containment Flooding, to ensure critical plant components necessary for plant recovery are not rendered inoperable

D. Go To FR-Z.3, Response to High Containment Radiation Level, to ensure isolation of non-essential ventilation penetrations to prevent potential radioactive releases from containment

Answer: C

Explanation:

A. Incorrect. Sump level is an orange path containment CSF that requires the CRS to make a transition to FR-Z.2.
B. Incorrect. Do not meet the entry requirement for FR-Z.1. Containment pressure would have to be at least 27 psig.

C. Correct. A severe challenge exists to the Containment CSF with a sump level of 108 inches. If action is not taken, critical components necessary for plant recovery could be rendered inoperable, making them unavailable to respond to the plant casualty and maintain the plant shutdown and stable.
D. Incorrect. The entry level for FR-Z.3 is met, but this is a yellow path procedure. A transition would not be made by the CRS to this procedure when an orange path procedure is required.

Technical Reference(s): CSF-1, BD-FR-Z.2

References to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # ____
Modified Bank # __ ____
New ___X____

Question History: Last NRC Exam ___________

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X___

10 CFR Part 55 Content: 55.43.5

Comments:

Examination Outline Cross-reference:	**Level**	**SRO**		
	Tier #	2		
	Group #	1		
	K/A #	004A2.15		
	Importance Rating	3.7		
Ability to (a) predict the impacts of the following malfunctions or operations on the CVCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: High or low PZR level.				

Question #86

Given the following conditions:

- The plant is at 100% power.
- All controls are in the normal power operation lineup.
- Pressurizer level is lowering.
- Volume Control Tank level is rising.
- SEAL INJ TO RCP FLOW LO alarm is lit.
- LTDN REGEN HX TEMP HI alarm is lit.
- LTDN HX DISCH TEMP HI alarm is lit.
- CHG LINE FLOW HILO alarm is lit.

Which ONE (1) of the following procedures will be implemented by the Control Room Supervisor to respond to the above plant conditions?

A. OTO-BB-00006, Pressurizer Pressure Control Malfunction

B. OTO-BG-00004, VCT Level Channel Failure

C. OTO-BB-00003, Reactor Coolant System Excessive Leakage

D. OTO-BG-00001, Pressurizer Level Control Malfunction

Answer: D

Explanation:
Considering and analyzing all the given indications together indicate that a PZR level channel has failed high. If given indications are analyzed independently, other failures could be explained, but would not be correct when using all given indications.

A Incorrect. Pressure would change if a level malfunction occurred, but other conditions given would not be consistent with a pressure control malfunction.
B Incorrect. VCT level is rising, which would be an entry to OTO-BG-00004, but other conditions would not be consistent with a VCT level channel failure.

C Incorrect. PZR level would be lowering if an RCS leak existed, but other conditions are not consistent with excessive RCS leakage.
D Correct. Correct procedure for conditions given.

Technical Reference(s): OTO-BG-00001

References to be provided to applicants during examination: None

Learning Objective: T61.003B, B-43, Obj D

Question Source: Bank # _L7121___
Modified Bank # __ ____
New __ _____

Question History: Last NRC Exam ___________

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X___

10 CFR Part 55 Content: 55.43.5

Comments:

Examination Outline Cross-reference:	Level	SRO		
	Tier #	2		
	Group #	1		
	K/A #	005 2.4.9		
	Importance Rating	4.2		
Residual Heat Removal: Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.				

Question #87

The following conditions exist at Callaway:

- ‘A’ RHR train is in shutdown cooling mode of operation.
- ‘B’ RHR train is in standby.
- ‘A’ CCP is in service.
- The reactor vessel head is removed.
- RCS temperature is 140°F.
- RCS level is 54 inches.

A loss of both NB01 and NB02 has occurred; neither diesel can be started.

(1) Which procedure will the CRS enter and (2) what is the FIRST method the CRS will direct to add inventory to the RCS?

A. (1) OTO-EJ-00003, Loss of RHR While Operating at Reduced Inventory or Mid-Loop Conditions.
(2) Initiate RWST Gravity Feed.

B. (1) OTO-EJ-00001, Loss of RHR Flow.
(2) Start the NCP and align to the charging header.

C. (1) OTO-EJ-00003, Loss of RHR While Operating at Reduced Inventory or Mid-Loop Conditions.
(2) Start the NCP and align to the charging header.

D. (1) OTO-EJ-00001, Loss of RHR Flow.
(2) Initiate RWST Gravity Feed.

Answer: A

Explanation:

A Correct.

B Incorrect. The NCP is available and would be started later in OTO-EJ-00003. <64” RCS level is reduced inventory, the OOA directs to OTO-EJ-00003.
C Incorrect. The NCP is available and would be started later in OTO-EJ-00003.
D Incorrect. <64” RCS level is reduced inventory, the OOA directs to OTO-EJ-00003.

Technical Reference(s): OOA-ZZ-SSM01, OTO-EJ-00003

References to be provided to applicants during examination: None

Learning Objective: T61.003E, E-3, Obj H

Question Source: Bank # ____
Modified Bank # ______
New ____X___

Question History: Last NRC Exam ___________

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X___

10 CFR Part 55 Content: 55.43.5

Comments:

Examination Outline Cross-reference:	Level	SRO		
	Tier #	2		
	Group #	1		
	K/A #	006A2.12		
	Importance Rating	4.8		
Ability to (a) predict the impacts of the following malfunctions or operations on the ECCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Conditions requiring actuation of ECCS.				

Question #88

The following conditions exist at Callaway:

- A reactor trip has occurred.
- The crew has transitioned to ES-0.1, Reactor Trip Response.

After entering ES-0.1 the following conditions were reported to the CRS:

- RCS pressure is 1930 psig and lowering slowly.
- RCS temperature is 554°F and lowering slowly.
- Letdown is isolated.
- Pressurizer level is 5% and lowering slowly.

Which ONE (1) of the following actions will the CRS direct?

A. Initiate SI and continue in ES-0.1.

B. Initiate ECCS flow as necessary and continue in ES-0.1.

C. Initiate SI and return to E-0, Reactor Trip or Safety Injection.

D. Initiate ECCS flow as necessary and return to E-0.

Answer: C

Explanation:

A Incorrect. Per ES-0.1 if SI is initiated transition to E-0.
B Incorrect. If in E-1 and PZR level lowers to< 9% then establish ECCS flow and continue in E-1.
C Correct.
D Incorrect. Per the foldout page the operator is directed to actuate SI, not just initiate ECCS flow.

Technical Reference(s): ES-0.1

References to be provided to applicants during examination: None

NRC Site-Specific Written Examination
Callaway Plant
Senior Reactor Operator

Learning Objective: T61.003D, D-6, Obj E

Question Source: Bank # ____
Modified Bank # ______
New ____X___

Question History: Last NRC Exam ___________

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

10 CFR Part 55 Content: 55.43.5

Comments:

Examination Outline Cross-reference:	Level	SRO		
	Tier #	2		
	Group #	1		
	K/A #	026 2.2.22		
	Importance Rating	4.7		
Containment Spray: Knowledge of limiting conditions for operations and safety limits.				

Question #89

Given the following plant conditions:

Callaway is in Mode 1

1200 CTMT Spray Pump 'A' declared INOPERABLE due to a failed surveillance.
1227 CTMT Spray Pump 'B' also declared INOPERABLE due to the results of a common cause failure analysis.
1254 Plant shutdown to Mode 3 commenced.
1319 CTMT Spray Pump 'A' returned to OPERABLE status.
1338 CTMT Spray Pump 'B' returned to OPERABLE status.

Plant conditions allow the plant shutdown to be terminated at (1) due to the Containment Spray System now being able to ensure that containment pressure will not exceed (2) .

	(1)	(2)
A.	1319	47.8 psig
B.	1338	60 psig
C.	1319	60 psig
D.	1338	47.8 psig

Answer: A

Explanation:

A Correct.
B Incorrect. This pressure is containment design pressure, not the TS basis pressure for containment spray system operability.
C Incorrect. This pressure is containment design pressure, not the TS basis pressure for containment spray system operability.

D Incorrect. Plausible to think that both pumps need to be returned to service before stopping the SD once it has started.

Technical Reference(s): TS 3.6.6, TS Basis B 3.6.6

References to be provided to applicants during examination: None

Learning Objective: T61.0110 6, LP-18, Obj K

Question Source: Bank # __ Wolf Creek 2009 SRO exam__
Modified Bank # ______
New ______

Question History: Last NRC Exam ___________

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X___

10 CFR Part 55 Content: 55.43.2

Comments:

Examination Outline Cross-reference:	Level	SRO		
	Tier #	2		
	Group #	1		
	K/A #	061A2.05		
	Importance Rating	3.4		
Ability to (a) predict the impacts of the following malfunctions or operations on the AFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Automatic control malfunction.				

Question #90

The following initial conditions exist at Callaway:

- Mode 1, 100% power.
- 'A' MDAFP is tagged OOS.

After a reactor trip due to a trip of the running Main Feed Pump, the following conditions were reported to the CRS when checking Feedwater Status:

The Reactor Operator (RO) reports the following to the CRS:
- RCS Tave is 562°F and lowering.

The Balance of Plant Operator (BOP) reports the following to the CRS:
- Both Main Feed Pumps are tripped.
- All Feedwater reg and reg bypass valves are closed.
- 'A' Steam Generator Aux Feed flow is 125,000 lbm/hr.
- 'B' Steam Generator Aux Feed flow is 0 lbm/hr.
- 'C' Steam Generator Aux Feed flow is 0 lbm/hr.
- 'D' Steam Generator Aux feed flow is 150,000 lbm/hr.

Which ONE (1) of the following describes the actions the CRS would perform?

A. Direct the BOP to start 'B' MDAFP using AL HIS-22A while continuing in current procedure in effect.

B. Direct the BOP to start 'B' MDAFP using AL HIS-22A and transition to FR-H.1, Response to Loss of Secondary Heat Sink.

C. Direct the RO to initiate a TDAFAS using SA HS-17 and transition to FR-H.1, Response to Loss of Secondary Heat Sink.

D. Direct the RO to initiate a TDAFAS using SA HS-17 while continuing in current procedure in effect.

Answer: D

Explanation:

A Incorrect. Based on the indications given "B" MDAFP is already in service.
B Incorrect. Based on the indications given "B" MDAFP is already in service.
C Incorrect. The TDAFW pump should be started. After this pump is started there will be sufficient AFW flow and no transition will be required to FR-H.1.
D Correct.

Technical Reference(s): ES-0.1

References to be provided to applicants during examination: None

Learning Objective: T61.003D, D-6, Obj H

Question Source: Bank # ____
Modified Bank # ______
New ___X___

Question History: Last NRC Exam ___________

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X___

10 CFR Part 55 Content: 55.43.5

Comments:

Examination Outline Cross-reference:	Level	SRO		
	Tier #	2		
	Group #	2		
	K/A #	014A2.05		
	Importance Rating	4.1		
Ability to (a) predict the impacts of the following malfunctions or operations on the RPIS; and (b) based on those on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Reactor trip.				

Question #91

The following conditions exist:

- Mode 1, 100% reactor power.
- Annunciator 81A, TWO/MORE RODS AT BOTTOM, annunciates.

The Reactor Operator reports that rod bottom lights for rods D-4 and M-12 are lit on SB-074, DRPI Rod Position Indication.

Which ONE (1) of the following procedures will the Control Room Supervisor utilize for the transient in progress to stabilize the plant AND what is the basis for this action?

A. OTO-SF-00001, Rod Control Malfunctions; prevent unacceptable power peaking factors

B. E-0, Reactor Trip or Safety Injection; prevent unacceptable power peaking factors

C. E-0, Reactor Trip or Safety Injection; minimize consequences of reduced shutdown margin

D. OTO-SF-00001, Rod Control Malfunctions; minimize consequences of reduced shutdown margin

Answer: B

Explanation:

A Incorrect. OTO-SF-00001 could be entered but a transition would be immediately made to E-0 and that procedure would be used to stabilize the plant.
B Correct.
C Incorrect. SDM assumes all rods are inserted, therefore a dropped rod does not reduce SDM.
D Incorrect. OTO-SF-00001 could be entered but a transition would be immediately made to E-0 and that

procedure would be used to stabilize the plant.

Technical Reference(s): OTA-RK-00022, Addendum 81A, Tech Spec B 3.1.4

References to be provided to applicants during examination: None

Learning Objective: T61.003B, LP-45, Obj G

Question Source: Bank # ____
Modified Bank # ______
New ___X___

Question History: Last NRC Exam ___________

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

10 CFR Part 55 Content: 55.43.5

Comments:

Examination Outline Cross-reference:	Level	SRO		
	Tier #	2		
	Group #	2		
	K/A #	034K6.01		
	Importance Rating	3.0		
Knowledge of the effect of a loss or malfunction on the following will have on the Fuel Handling System: Fuel handling equipment.				

Question #92

Callaway Plant is performing a refueling with the core offload in progress.

The Refueling SRO calls the Control Room Supervisor (CRS) and informs him that a fuel assembly has been dropped into the lower section of the Refuel Pool due to a failure of the Refueling Machine Gripper.

Containment radiation levels are currently normal and stable.

Which ONE (1) of the following describes the CRS directions in response to the dropped fuel assembly?

A. Initiate Control Room Ventilation Isolation Signal (CRVIS)
Initiate Fuel Building Isolation Signal (FBIS)
Evacuate Containment

B. Initiate Control Room Ventilation Isolation Signal (CRVIS)
Evacuate Containment
Ensure Only 1 Fuel/Aux Bldg Norm Exhaust Fan Running In Slow

C. Initiate Control Room Ventilation Isolation Signal (CRVIS)
Evacuate Containment
Close Fuel Transfer Tube Isolation Valve

D. Initiate Fuel Building Isolation Signal (FBIS)
Ensure Only 1 Fuel/Aux Bldg Norm Exhaust Fan Running In Slow
Close Fuel Transfer Tube Isolation Valve

Answer: C

Explanation:

A. Incorrect. FBIS only done if assembly dropped in FB.
B. Incorrect. Exhaust Fan only checked if assembly dropped in FB.
C. Correct
D. Incorrect. FBIS only done and Exhaust Fan only checked if assembly dropped in FB.

NRC Site-Specific Written Examination
Callaway Plant
Senior Reactor Operator

Technical Reference(s): OTO-KE-00001

References to be provided to applicants during examination: None

Learning Objective: T61.003E 6, LP-5, Obj I

Question Source: Bank # ______
Modified Bank # ______
New ____X___

Question History: Last NRC Exam ____N/A________

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content:
55.43.7

Comments:

Examination Outline Cross-reference:	Level	SRO		
	Tier #	2		
	Group #	2		
	K/A #	045 G2.4.4		
	Importance Rating	4.7		
Main Turbine Generator: Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.				

Question #93

Given the following conditions:

- Reactor power is 99%.
- All plant systems are in the correct configuration for this power level.
- The following sequence of events occur:
  - 0605: Main Generator Gas Temperature is 57°C and trending up.
  - 0608: EATV0007, Generator H2 Cooler TCV is fully open.
  - 0610: EAV0037, Generator H2 Cooler Service Water Outlet Isolation valve is fully open.
  - 0611: Main Generator Gas Temperature is 59°C and still trending up.

Which ONE (1) of the following directions should the Control Room Supervisor give to the Reactor Operators for the conditions given above?

A. Reduce Main Generator MVARs to reduce gas temperature. If gas temperature cannot be lowered to <56°C in the next 9 minutes, trip the reactor and go to E-0, Reactor Trip or Safety Injection.

B. Reduce Main Generator load at 1%/min to reduce gas temperature. If gas temperature cannot be lowered to <56°C in the next 9 minutes, trip the main turbine and go to OTO-AC-00001, Turbine Trip Below P-9.

C. Reduce Main Generator MVARs and continue Main Generator operation if gas temperature stabilizes at current value. If gas temperature does not stabilize, go to OTO-MA-00004, Generator Gas System Malfunction (Temperature).

D. Reduce Main Generator load in accordance with OTO-MA-00008, Rapid Load Reduction, until gas temperature is reduced to <58°C.

Answer: A

Explanation:

A. Correct.
B. Incorrect. Correct action would be to trip the reactor, not the turbine. With indications given, power would still be above 50%, thus requiring the reactor to be tripped before the turbine.
C. Incorrect. Even if temperature stabilizes, it is still above the limit and requires a unit trip within 15 minutes. Also, OTO-MA-00004 would not be entered at this time as it is already implemented based on the given conditions.
D. Incorrect. Temperature must be lowered to ≤56°C within 15 minutes.

Technical Reference(s): OTO-MA-00004

References to be provided to applicants during examination: None

Learning Objective: T61.003B 6, LP-22, Obj C, D, E

Question Source: Bank # __R14239____
Modified Bank # ______
New _______

Question History: Last NRC Exam ____N/A________

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X___

10 CFR Part 55 Content:
55.43.5

Comments:

Examination Outline Cross-reference:	Level	SRO		
	Tier #	3		
	Group #	1		
	K/A #	G2.1.25		
	Importance Rating	4.2		
Ability to interpret reference materials, such as graphs, curves, tables, etc.				

Question #94

Given the following conditions:

- The Safety Monitor is Out of Service.
- The Turbine Driven AFW Pump is out of service for maintenance.

Which ONE (1) of the following can the Control Room Supervisor authorize to be removed from service with the least amount of safety risk?

(REFERENCE PROVIDED)

A. Residual Heat Removal Pump A

B. Essential Service Water Pump B

C. Centrifugal Charging Pump A

D. Containment Cooler B

Attachment 1

Equipment OOS PRA Matrix

Sheet 1 of 1

Train Being Taken OOS

	AL-T (4)	AL-A (5)	AL-B (5)	BG-A	BG-B	EM-A	EM-B	EJ-A	EJ-B	EG-A	EG-PA	EG-PC	EG-B	EG-PB	EG-PD	EF-A	EF-B	EA (1)	NE-A	NE-B	SWYD (2)	PZR (3)	GN-A	GN-B	EN-A	EN-B
AL-T (4)	X																									
AL-A (5)		X																								
AL-B (5)			X																							
BG-A				X																						
BG-B					X																					
EM-A						X																				
EM-B							X																			
EJ-A								X																		
EJ-B									X																	
EG-A										X	N/A	N/A														
EG-PA											X															
EG-PC												X														
EG-B													X	N/A	N/A											
EG-PB														X												
EG-PD															X											
EF-A																X							***			
EF-B																	X							***		
EA (1)																		X								
NE-A																			X							
NE-B																				X						
SWYD (2)																					X					
PZR (3)																						X				
GN-A																***							X			
GN-B																	***							X		
EN-A																									X	
EN-B																										X

Key:

(grey)	Increased risk undesirable	X	Same train	***	Train is NOT functional. Can also be removed from service
(black)	NOT allowed by Tech Specs	N/A	Not Applicable	(white)	Increased risk acceptable

See Notes and Cautions for Matrix usage on Sheets 2 and 3 of this attachment.
The Notes associated to this Matrix are system specific, the Cautions are generic.

Answer: C

Explanation:

A. Incorrect. Increased risk not acceptable.
B. Incorrect. Increased risk not acceptable.
C. Correct
D. Incorrect. Increased risk not acceptable.

Technical Reference(s): EDP-ZZ-01129

References to be provided to applicants during examination: EDP-ZZ-01129, Attachment 1, Equipment OOS PRA Matrix

Learning Objective: T61.003A 6, LP-24, Obj J

Question Source: Bank # __R8610____
Modified Bank # ______
New _______

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:
55.43.5

Comments:

Examination Outline Cross-reference:	**Level**	**SRO**		
	Tier #	3		
	Group #	1		
	K/A #	G2.1.35		
	Importance Rating	3.9		
Knowledge of the fuel-handling responsibilities of SROs.				

Question #95

Given the following conditions:

- Callaway Plant is in a refueling outage.
- Core offload is presently in progress.
- An adjustment needs to be made to the Refuel Machine bridge upender zone interlock that requires the Refuel Machine to be placed into Bypass Operation.

Which ONE (1) of the following correctly identifies the individuals who may approve placing the Refuel Machine into Bypass Operation?

A. Refueling Reactor Engineer AND Shift Manager

B. Refueling SRO AND Emergency Duty Officer

C. Refueling Reactor Engineer AND Emergency Duty Officer

D. Refueling SRO AND Shift Manager

Answer: D

Explanation: Per OTS-KE-00013 to place the refuel machine in bypass requires approval of the Refueling SRO and a second SRO.

A. Incorrect. Plausible as the Refueling Reactor Engineer and Refueling SRO oversee fuel movement during offload activities.
B. Incorrect. EDO not authorized as he is not currently licensed.
C. Incorrect. Plausible as the Refueling Reactor Engineer and Refueling SRO oversee fuel movement during offload activities and EDO can authorize most activities that the SM has responsible for.
D. Correct

Technical Reference(s): OTS-KE-00013

References to be provided to applicants during examination: None

Learning Objective: T61.003E 6, LP-5, Obj E

NRC Site-Specific Written Examination
Callaway Plant
Senior Reactor Operator

Question Source: Bank # ______
Modified Bank # ______
New ____X___

Question History: Last NRC Exam ____N/A________

Question Cognitive Level:
Memory or Fundamental Knowledge __X___
Comprehension or Analysis _____

10 CFR Part 55 Content:
55.43.7

Comments:

Examination Outline Cross-reference:	**Level**	**SRO**		
	Tier #	3		
	Group #	2		
	K/A #	G2.2.7		
	Importance Rating	3.6		
Knowledge of the process for conducting special or infrequent tests.				

Question #96

An Infrequently Performed Test or Evolution (IPTE) is being performed in accordance with APA-ZZ-0100A, Infrequently Performed Test or Evolution Guidance.

Which ONE (1) of the following would NOT require you, as the Control Room Supervisor, to terminate the evolution in progress?

A. Margin of safety IS significantly reduced

B. Prerequisite conditions are NOT as expected

C. Test termination criteria IS approached

D. Equipment does NOT respond as expected

Answer: C

Explanation:

A. Incorrect. Would require the evolution to be terminated.
B. Incorrect. Would require the evolution to be terminated.
C. Correct. Would only require evolution to be terminated if test termination criteria is reached or exceeded.
D. Incorrect. Would require the evolution to be terminated.

Technical Reference(s): APA-ZZ-0100A

References to be provided to applicants during examination: None

Learning Objective: T61.003A 6, LP-14, Obj E

Question Source: Bank # ______
Modified Bank # ______
New ____X___

Question History: Last NRC Exam ____N/A________

Question Cognitive Level:

NRC Site-Specific Written Examination
Callaway Plant
Senior Reactor Operator

Memory or Fundamental Knowledge __X___
Comprehension or Analysis _____

10 CFR Part 55 Content:
55.43.3

Comments:

Examination Outline Cross-reference:	Level	SRO		
	Tier #	3		
	Group #	3		
	K/A #	G2.3.4		
	Importance Rating	3.7		
Knowledge of radiation exposure limits under normal or emergency conditions.				

Question #97

The plant is in MODE 6 with core off load in progress.

A fuel handling accident has occurred that resulted in damage to a fuel assembly and the refueling machine gripper.

An Alert has been declared due to the damaged fuel assembly.

In order to repair the gripper and place the damaged fuel assembly in a safe location a diver must be used and the Occupational Dose Limits of 10CFR20 will have to be exceeded.

Which ONE (1) of the following individuals can authorize this dose exposure AND what will be the new dose limit be for this job.

A. Emergency Coordinator 10 rem DDE

B. Plant Director 10 rem DDE

C. Plant Director 100 rem DDE

D. Emergency Coordinator 100 rem DDE

Answer: A

Explanation: Limit is 2 Rem at Callaway not to exceed 4 Rem including prior site.

A. Correct.
B. Incorrect. Authorization to exceed 10CFR20 limits must come from at least the EC. The Plant Director does not have this authority.
C. Incorrect. Authorization to exceed 10CFR20 limits must come from at least the EC. The Plant Director does not have this authority. Also, 100 rem is the limit to save a life, not mitigate an accident.
D. Incorrect. 100 rem is the limit to save a life, not mitigate an accident.

Technical Reference(s): HDP-ZZ-01450

NRC Site-Specific Written Examination
Callaway Plant
Senior Reactor Operator

References to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # ______
Modified Bank # ______
New ____X___

Question History: Last NRC Exam ____N/A________

Question Cognitive Level:
Memory or Fundamental Knowledge __X___
Comprehension or Analysis _____

10 CFR Part 55 Content:
55.43.4

Comments:

Examination Outline Cross-reference:	**Level**	**SRO**		
	Tier #	3		
	Group #	3		
	K/A #	G2.3.11		
	Importance Rating	4.3		
Ability to control radiation releases.				

Question #98

Callaway is in Mode 6, with fuel being moved from the Reactor Vessel to the spent fuel pool.

During Core Alterations the most severe radiological consequences would result from a ____(1)______.

Reactor Vessel water level shall be maintained ≥ 23 ft above the top of the _____(2)________ to ensure that offsite doses are maintained within 25% of 10 CFR limits.

	(1)	**(2)**
A.	Fuel Handling Accident	Fuel seated in the Reactor Vessel
B.	Fuel Handling Accident	Top of the Reactor Vessel flange
C.	Loss of RHR Cooling	Top of the Reactor Vessel flange
D.	Loss of RHR Cooling	Fuel seated in the Reactor Vessel

Answer: B

Explanation:

A. Incorrect. Plausible for Item 2 as this is a FSAR 16.9.4 requirement, but only for the movement of control rods in Mode 6.
B. Correct.
C. Incorrect. Incorrect accident.
D. Incorrect. Incorrect accident and level is for movement of control rods in Mode 6.

Technical Reference(s): TS 3.9.7 and TS Basis B 3.9.4

References to be provided to applicants during examination: None

Learning Objective: T61.0110 6, LP-40, Obj O
T61.003E 6, LP-4, Obj A

NRC Site-Specific Written Examination
Callaway Plant
Senior Reactor Operator

Question Source: Bank # ____
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ____N/A________

Question Cognitive Level:
Memory or Fundamental Knowledge __X___
Comprehension or Analysis _____

10 CFR Part 55 Content:
55.43.2 and 4

Comments:

Examination Outline Cross-reference:	Level	SRO		
	Tier #	3		
	Group #	4		
	K/A #	G2.4.4		
	Importance Rating	4.7		
Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.				

Question #99

Given the following conditions:

- A large break Loss of Coolant Accident (LOCA) has occurred from 100%.
- The crew has just transitioned from E-0, Reactor Trip or Safety Injection, to E-1, Loss of Reactor or Secondary Coolant.

The Shift Technical Advisor (STA) starts monitoring the Critical Safety Functions and immediately reports that an ORANGE PATH exists under SUBCRITICALITY.

Which ONE (1) of the following describes correct procedure usage by the Control Room Supervisor based on the information provided by the STA?

A. Continue the current pass through the status trees and transition to FR-S.1, Response to Nuclear Power Generation if no RED path is encountered.

B. Complete the actions of E-1, then transition to FR-S.1, if the actions of E-1 do not address the ORANGE path.

C. Immediately transition to FR-S.1, then continue current pass through the status trees.

D. Immediately transition to FR-S.2, Response to Loss of Core Shutdown, while continuing the actions of E-1 to address the LOCA.

Answer: A

Explanation:

A. Correct
B. Incorrect. An orange path or red path requires a transition to the appropriate FRG procedure. This action takes precedence over the ORG procedure (E-1).
C. Incorrect. Before transitioning to FR-S.1 it must be verified that a higher priority FRG is not applicable.
D. Incorrect. The orange path and red path for subcriticality both use FR-S.1 as their response. Also, FRGs are not performed simultaneously with ORGs.

NRC Site-Specific Written Examination
Callaway Plant
Senior Reactor Operator

Technical Reference(s): CSF-1

References to be provided to applicants during examination: None

Learning Objective: T61.003D 6, LP-1, Obj L

Question Source: Bank # __L2255____
Modified Bank # ______
New _______

Question History: Last NRC Exam ____N/A________

Question Cognitive Level:
Memory or Fundamental Knowledge __X___
Comprehension or Analysis _____

10 CFR Part 55 Content:
55.43.10

Comments:

NRC Site-Specific Written Examination
Callaway Plant
Senior Reactor Operator

Examination Outline Cross-reference:	**Level**	**SRO**		
	Tier #	3		
	Group #	4		
	K/A #	G2.4.29		
	Importance Rating	4.4		
Knowledge of the emergency plan.				

Question #100

What is the LOWEST Emergency Plan declaration at which the Emergency Coordinator must implement EIP-ZZ-00230, Accountability?

A. Unusual Event

B. Alert

C. Site Area Emergency

D. General Emergency

Answer: C

Explanation:

A. Incorrect.
B. Incorrect.
C. Correct.
D. Incorrect.

Technical Reference(s): EIP-ZZ-00230

References to be provided to applicants during examination: None

Learning Objective: T68.1020 6, Obj L

Question Source: Bank # ______
Modified Bank # ______
New ____X___

Question History: Last NRC Exam ____N/A________

Question Cognitive Level:
Memory or Fundamental Knowledge __X___
Comprehension or Analysis _____

10 CFR Part 55 Content:
55.43.1

NRC Site-Specific Written Examination
Callaway Plant
Senior Reactor Operator

Comments: