

**CALLAWAY PLANT**  
**EXAMINATION COVER SHEET**  
**TRAINING DEPARTMENT**

COURSE NO.: \_\_\_\_\_ SESSION NO.: \_\_\_\_\_

COURSE TITLE: NRC Initial License Exam RO/SRO

NAME (Print): \_\_\_\_\_ PIN: \_\_\_\_\_ # QUESTIONS: 100

SIGNATURE: \_\_\_\_\_ DATE: 6/19/09 TEST #: \_\_\_\_\_ BOOKLET #: N/A

DIRECTIONS: BLACK OUT CORRECT ANSWERS

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EXAM PREPARER: \_\_\_\_\_  
 EXAM REVIEWER: \_\_\_\_\_ / \_\_\_\_\_  
Date

**SCORING**  
 POINTS POSSIBLE: 100  
 POINTS MISSED: \_\_\_\_\_  
 POINTS SCORED: \_\_\_\_\_  
 GRADE: \_\_\_\_\_

# EXAMINATION DIRECTIONS

*THIS IS NOT CONSIDERED A QA RECORD, DO NOT FILM*

## **GENERAL**

1. Ensure that you print your name, PIN and you sign the Examination Cover Sheet prior to starting the examination.
2. Make sure you read each question carefully before answering.
3. If you should have any questions during the examination, raise your hand and the Instructor will assist you.
4. All student responses will be graded. Point value will be determined by the type of question and the student response.
5. All exam questions should be answered from memory unless the Instructor provides specific instructions otherwise.

## **MULTIPLE CHOICE AND TRUE-FALSE QUESTIONS**

1. There is only one best answer for Multiple Choice and True-False questions.
2. Unless otherwise directed, mark the correct answer by filling in the appropriate box/letter for the question on the Examination Cover Sheet.
3. For True-False questions, True corresponds to A. False corresponds to B.
4. If you wish to change your answer, either erase or crossout the previous answer.
5. If Examination Booklets are used, DO NOT MARK IN THE BOOKLET. If you should need scratch paper, ask the instructor.

## **ESSAY QUESTIONS**

1. When answering essay questions, the detail and time spent on the answer should be proportional to the point value assigned.
2. State all assumptions in your answer unless they are stated in the exam question.
3. When questions on exam call for a list of items, all student responses will be graded and the number of responses will be divided by the point total.

## **EXAMINATION FAILURE**

1. If you should fail an examination, your supervisor will be notified.

## **CHEATING**

1. Any student observed cheating on an examination will be removed from the classroom, receive an immediate counseling session with the appropriate STS, and receive a 0% on the examination.

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Examination Outline Cross-reference:	Level	RO	SRO	
	Tier #	1		
	Group #	1		
	K/A #	007 EK1.02		
	Importance Rating	3.4		
Knowledge of the operational implications of the following concepts as they apply to the reactor trip: Shutdown margin				

**Question #1**

Given the following plant conditions:

- Reactor tripped from 100% power equilibrium conditions at 0100.
- Boron concentration remains constant.
- $T_{avg}$  is at the no-load value.
- Shutdown Margin (SDM) is -5600 pcm at 0700.
- Critical Rod Height for 0700 is 115 steps on Control Bank 'D'.

Which ONE of the following correctly describes the change in SDM and Critical Rod Height if the reactor startup is delayed until 0800?

- A. More SDM, Critical Rod Height is HIGHER.
- B. Less SDM, Critical Rod Height is LOWER.
- C. More SDM, Critical Rod Height is LOWER.
- D. Less SDM, Critical Rod Height is HIGHER.

*Justification:*

*More SDM due to Xenon building in - More rods withdrawn to make up for negative reactivity*

- A. Correct.
- B. Incorrect. Would be more SDM and higher rods. see above.
- C. Incorrect. Would be higher rods, see above.
- D. Incorrect. Would be more SDM, see above.

Technical Reference(s): OSP-SF-00001, Shutdown Margin Calculations

Proposed references to be provided to applicants during examination: None

Learning Objective: Reactor Theory – Fission Product Poisons and Reactor Operational Physics

Question Source: Bank #   R12180    
Modified Bank #             
New           

Question History: Last NRC Exam   N/A  

Question Cognitive Level:

Memory or Fundamental Knowledge         
Comprehension or Analysis   X

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10 CFR Part 55 Content:

55.41   5  

55.43       

Comments:

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Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	009 EK2.03	
	Importance Rating	3.0	
Knowledge of the interrelations between the small break LOCA and the following: S/Gs			

**Question #2**

Given the following plant conditions:

- A Small Break Loss of Coolant Accident (SBLOCA) has occurred and operator actions have not been initiated
- The reactor has tripped from 100% power after operating for 450 days
- ECCS is operating as designed and the Main Steam Isolation Valves are open
- Steam dumps are available
- The RCS is saturated with RCS pressure above steam generator pressure

Which ONE of the following components will be used to establish Long Term Cooling?

- A. Steam Generators
- B. Accumulators
- C. Reactor Coolant Pumps
- D. Safety Injection Pumps

*Justification*

- A. Correct.
- B. Incorrect, Used only in the injection phase not for long term cooling
- C. Incorrect, RCP's (forced flow) not required for long term cooling
- D. Incorrect, SI pumps not required for long term cooling

Technical Reference(s): ES-1.2, step 9

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # \_\_\_\_\_  
Modified Bank # \_\_\_\_\_  
New  \_\_\_\_\_

Question History: Last NRC Exam  N/A \_\_\_\_\_

Question Cognitive Level:

Memory or Fundamental Knowledge  \_\_\_\_\_  
Comprehension or Analysis \_\_\_\_\_

10 CFR Part 55 Content:

55.41  3, 4

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55.43 \_\_\_\_\_

Comments:

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Examination Outline Cross-reference:	Level	RO	SRO	
	Tier #	1		
	Group #	1		
	K/A #	015/017 AK3.02		
	Importance Rating	3.0		
Knowledge of the reasons for the following responses as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow) : CCW lineup and flow paths to RCP oil coolers				

### Question #3

Given the following plant conditions:

- The Callaway Plant is operating at 100% power.
- EG HV-59, CCW from Ctmt Outer Isolation Valve closed 2 minutes ago due to an electrical short and cannot be opened.
- EG HV-60, CCW from RCS Inside Ctmt Isolation Valve has remained open.
- Highest Upper Radial Bearing temperature is currently reading 196°F and rising on all RCPs.

In accordance with OTO-BB-00002, RCP Off-Normal, which ONE of the following actions, if any, is required and why?

- A. Pumps can remain in service since CCW flow to oil coolers will be maintained through return valve EG HV-61, CCW from RCP Thermal Barrier Outer Ctmt Isolation.
- B. Pumps can remain in service until CCW Heat Exchanger Disch Temp Hi annunciator is received.
- C. Reactor must be tripped and ALL RCPs stopped due to loss of CCW flow to ALL RCP motor bearing coolers.
- D. Reactor must be tripped and "A" and "B" RCPs stopped due to loss of CCW flow to their respective oil coolers.

*Justification*

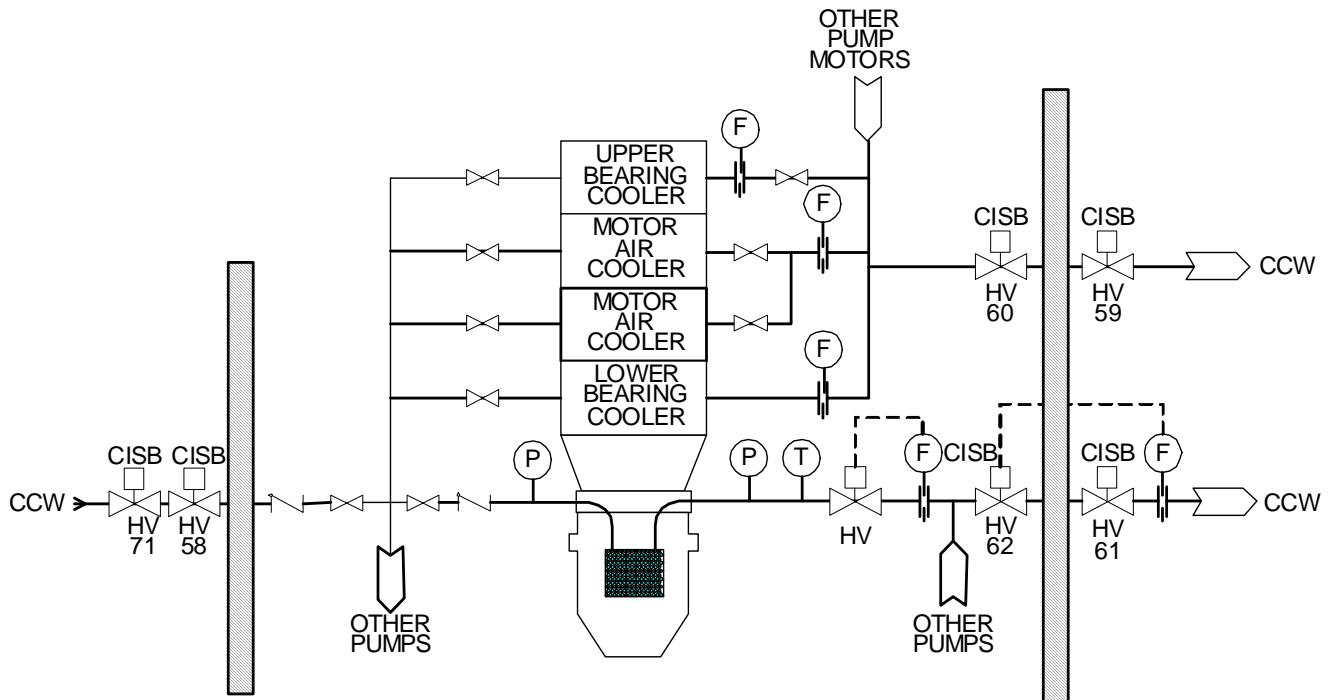
- A. *Incorrect. All CCW flow is lost*
- B. *Incorrect. Action is not conservative alarm and RCP trips are not based on CCW HX temperatures.*
- C. *Correct. Step C1 RNO is performed because the temp is >195°F.*
- D. *Incorrect. All RCPs must be tripped.*

*The cooling water passes through the thermal barrier heat exchanger and then through an orifice metering device (FT-17, 18, 19, 10). The flow device will shut a motor operated valve (BB-HV-13, 14, 15, 16) downstream of the thermal barrier on a sensed high CCW flow in excess of 50 gpm. This high flow would be indicative of a primary to CCW leak in the thermal barrier heat exchanger. Similarly, in the common return line for the CCW, from all the thermal barrier heat exchangers is another motor operated valve (EG-HV-62) which will automatically shut on a combined CCW return flow of greater than 206 gpm as sensed by flow device FT-62 in the common return line.*

*Component Cooling Water to the RCPs will also be automatically isolated on a Phase B Containment Isolation Signal (CISB). The CISB can be generated by either a containment pressure of 27 psig (High 3) on a 2 out of 4 coincidence, or by manual actuation of containment spray. The signal will cause the following six valves to shut: EG-HV-58, 71 (series CCW supply to RCP bearing coolers and thermal barrier heat exchangers), EG-HV61, 62*

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(series CCW combined return from the RCP thermal barrier heat exchangers), and EG HV-59, 60 (series CCW combined return from the RCP oil and air coolers). The above valves can also be operated from the Main Control Board (MCB). Valves BB HV-13, 14, 15, 16 and be operated from MCB panel RL021 and valves EG HV-58, 59, 60, 61, 62, 71 from panel RL019.



Technical Reference(s): OTO-BB-00002

Proposed 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  \_\_\_\_\_  
Comprehension or Analysis  X \_\_\_\_\_

10 CFR Part 55 Content:  
55.41  5, 10 \_\_\_\_\_  
55.43  \_\_\_\_\_

Comments:



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Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	022 AK1.03	
	Importance Rating	3.0	
Knowledge of the operational implications of the following concepts as they apply to Loss of Reactor Coolant Makeup: Relationship between charging flow and PZR level			

**Question #4**

Given the following plant conditions:

- PZR Level is 34%
- OTO-BB-00003, Reactor Coolant System Excessive Leakage, has been entered due to an Identified RCS leakage of 8 gpm
- $T_{ave}$  is constant
- Letdown is stable at 120 gpm
- Charging is in manual and stable at 132 gpm

With NO OPERATOR ACTION what is the longest amount of time until Letdown Isolates?

- A. 42.5 minutes
- B. 105.0 minutes
- C. 127.5 minutes
- D. 142.5 minutes

*Justification*

- A. Incorrect.  $34-17 = 17 \times 20 \text{ gal/\%} = 340 \text{ gals}$ , 42.5 minutes (20 gal/% is for the VCT)
- B. Incorrect.  $34-20 = 14 \times 60 \text{ gal/\%} = 840 \text{ gals}$ , 105 minutes
- C. Correct.  $34-17 = 17 \times 60 \text{ gal/\%} = 1020 \text{ gals}$ , 127.5 minutes
- D. Incorrect.  $34-15 = 19 \times 60 \text{ gal/\%} = 1140 \text{ gals}$ , 142.5 minutes

Technical Reference(s): OTO-BB-00003 , OTA-RK-00018 ADD 32B

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # \_\_\_\_\_  
Modified Bank # \_\_\_\_\_  
New  \_\_\_\_\_

Question History: Last NRC Exam  N/A \_\_\_\_\_

Question Cognitive Level:

Memory or Fundamental Knowledge \_\_\_\_\_  
Comprehension or Analysis  \_\_\_\_\_

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Reactor Operator

10 CFR Part 55 Content:

55.41 \_\_8, 10\_\_

55.43 \_\_\_\_\_

Comments:

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Examination Outline Cross-reference:	Level	RO	SRO	
	Tier #	1		
	Group #	1		
	K/A #	025 AK3.01		
	Importance Rating	3.1		
Knowledge of the reasons for the following responses as they apply to the Loss of Residual Heat Removal System: Shift to alternate flowpath				

**Question #5**

The Callaway Plant is in Mode 5, "COLD SHUTDOWN," with the following plant conditions:

- All CET's read 195°F and are stable.
- All S/G Narrow range levels are 44%.
- All S/G secondary water temperatures are 51°F higher than RCS cold leg temperatures.
- All RCP's are off.
- Train 'A' RHR is in service.
- Train 'B' RHR is inoperable for repairs.
- All systems aligned in their normal configuration for the present plant conditions.
- A loss of 'A' RHR pump has just occurred and cannot be restored.
- RCS temperature is rising.

Which ONE of the following is the preferred method for heat removal under these conditions in accordance with OTO-EJ-00001, Loss of RHR Flow?

- A. One train of SI valves aligned for injection and a High-Head Safety Injection pump running, spill through the Pressurizer PORVs.
- B. Charging Pump injecting flow through the normal charging line, spill through the Pressurizer PORVs.
- C. Natural Circulation RCS flow with all available S/G steam dump to atmosphere valves open, Auxiliary Feedwater flow established.
- D. An RCP running with forced RCS flow with all available S/G steam dump to atmosphere valves open, Auxiliary Feedwater flow established.

*Justification*

*A - Incorrect; This is an alternate RCS feed and bleed cooling method if secondary heat sink can not be established (i.e. at least two S/G available) and temperature is INCREASING.*

*B - Incorrect; This charging lineup is established for increasing RCS inventory on a sustained loss of RHR during reduced inventory conditions. The bleed path is the correct RCS bleed path if secondary heat sink can not be established (i.e. at least two S/G available).*

*C. Correct.*

*D - Incorrect; An RCP would not be started until after natural circulation has been established and RCS cold leg temperatures are greater than 275°F and S/G temperatures are within 10°F of RCS Tcold.*

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*S/G's must be >= 86% WR to be used as a Heat Sink*

Technical Reference(s): OTO-EJ-00001

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # \_\_\_\_\_  
Modified Bank # \_\_\_\_\_  
New  \_\_\_\_\_

Question History: Last NRC Exam  N/A

Question Cognitive Level:  
Memory or Fundamental Knowledge   
Comprehension or Analysis

10 CFR Part 55 Content:  
55.41  5, 10   
55.43

Comments:

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Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	026 AA1.05	
	Importance Rating	3.1	
Ability to operate and / or monitor the following as they apply to the Loss of Component Cooling Water: The CCWS surge tank, including level control and level alarms, and radiation alarm			

**Question #6**

Given the following plant conditions:

- The annunciator 51D, CCW Srg Tk A Lev HiLo, came in a few minutes ago when a second CCW pump was started for a test.
- LI-1, Tank "A", indicated 87% and slowly RISING.

A Safety Injection has subsequently occurred.

- While checking the Component Cooling pumps "A" and "C" running, the operator notices annunciator 51D, CCW Srg Tk A Lev HiLo, is flashing.
- Radiation Monitor RE-9 indicates  $6 \times 10^{-6}$   $\mu\text{Ci/ml}$ .
- The Component Cooling Surge Tank "A" level indicates 43% and slowly LOWERING.

Which ONE of the following describes the appropriate CCW system/operator response?

- A. Demineralized water auto makeup starts at 63%.
- B. Demineralized water auto makeup starts at 43.75%.
- C. Essential Service Water manual makeup is initiated at 43.75%.
- D. Essential Service Water manual makeup is initiated at 63%.

*Justification*

- A. Incorrect, Demin m/u starts at 43.75%, 63 is the number in inches.
- B. Correct.
- C. Incorrect, ESW m/u is only initiated manually.
- D. Incorrect, ESW m/u is only initiated manually. .

*The makeup valves will automatically open on a low level of 63 inches (43.75%) and close on a high level of 87 inches (60.4%). They also close on a high radiation alarm in their associated loop. LV-1 & 2 can also be operated from main control board panel RL019.*

*An activity level of  $1 \times 10^{-5}$   $\mu\text{Ci/ml}$  will generate an alert alarm.*

Annun 51D, CCW Srg Tk A Lev HiLo - 85.4/45%.  
Annun 53D, CCW Srg Tk B Lev HiLo - 85.4/45%.

Technical Reference(s): OTA-RK-0020 ADD 51D

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Proposed 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 \_\_\_7\_\_\_  
55.43 \_\_\_\_\_

Comments:

NRC Site-Specific Written Examination  
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Reactor Operator

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	027 AK1.03		
	Importance Rating	2.6		
Knowledge of the operational implications of the following concepts as they apply to Pressurizer Pressure Control Malfunctions: Latent heat of vaporization/condensation				

**Question #7**

Given the following plant conditions:

- The Callaway Plant is at 72% Reactor Power.
- All systems and controls are in automatic and stable.
- The OUTPUT of the PZR Master Pressure Controller is failed AS IS.
- The BOP initiates a load reduction to 65% at 1% per minute due to rising condenser pressure.
- Pressurizer level rises to 52% as a result of the transient.

What is the INITIAL response of the Pressurizer Pressure Control System during this event?

- A. BACKUP Heaters turn OFF due to rising RCS pressure.
- B. BACKUP Heaters turn ON to heat incoming surge volume.
- C. BOTH PZR Spray valves THROTTLE OPEN to reduce pressure to normal.
- D. ONE PZR PORV OPENS to maintain pressure below the High reactor trip setpoint.

*Justification*

- A. *Incorrect. Controller failed as is.*
- B. *Correct. Htrs are on from PZR Level deviation 5% above program level (raises temp to Latent Heat of Vaporization.)*
- C. *Incorrect. Controller failed as is.*
- D. *Incorrect. The two PORVs open together, not separately as they had in the past.*

Technical Reference(s): OTA-RK-00018, Add 32D

Proposed 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:

NRC Site-Specific Written Examination  
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Reactor Operator

Memory or Fundamental Knowledge  X  
Comprehension or Analysis  \_\_\_\_\_

10 CFR Part 55 Content:

55.41  7, 8, 10   
55.43  \_\_\_\_\_

Comments:



NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	038 2.1.20	
	Importance Rating	4.6	
Steam Generator Tube Rupture / Ability to interpret and execute procedure steps.			

**Question #8**

Given the following plant conditions:

- The plant was operating at 100% power when a reactor trip occurred on low pressurizer pressure.
- A Steam Generator Tube Rupture was diagnosed, and E-3, Steam Generator Tube Rupture was entered.
- RCS Cooldown and Depressurization is complete.

Given the following control room indications:

- SG "C" Blowdown Sample indicates high radiation.
- SG "C" NR level is 32% and dropping.
- Feed flow has been isolated to SG "C".
- SG "A", "B", and "D" levels are slowly lowering.
- Pressurizer level is 63% and rising.

Which ONE of the following describes the appropriate operator action?

- A. Depressurize RCS.
- B. Lower Charging flow.
- C. Turn on Pressurizer heaters.
- D. Depressurize RCS and lower Charging flow.

*Justification*

- A. Incorrect. If ruptured SG level is rising with a lower pwr level than exists, would depressurize RCS
- B. Incorrect. If pwr level is greater than 71%, would lower charging
- C. Correct.
- D. Incorrect. If ruptured SG level was rising, would perform both

Technical Reference(s): E-3 Step 29

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # \_\_\_\_\_  
Modified Bank # \_\_\_\_\_  
New \_\_\_X\_\_\_

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Question History: Last NRC Exam \_\_\_\_\_

Question Cognitive Level:

Memory or Fundamental Knowledge \_\_\_\_\_

Comprehension or Analysis   X  

10 CFR Part 55 Content:

55.41   10  

55.43 \_\_\_\_\_

Comments:

NRC Site-Specific Written Examination  
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Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	0040 AK2.02	
	Importance Rating	2.6*	
Knowledge of the interrelations between the Steam Line Rupture and the following: Sensors and detectors			

**Question #9**

Given the following plant conditions:

- The plant was at a steady state power level of 90%.
- Pressurizer pressure and level have suddenly started lowering rapidly.
- Pressurizer pressure and level control systems are responding properly in AUTO.

Which ONE of the following parameters ALONE can be used, PRIOR to a plant trip to determine that the pressurizer changes are the result of a Faulted Steam Generator vs a LOCA?

- A. Charging Flow
- B. Loop Differential Temperature
- C. Containment Humidity
- D. Reactor Coolant System Pressure

*Justification*

- A. Charging flow will rise for both events.
- B. Correct
- C. Containment Humidity rise for both events.
- D. RCS pressure will lower for both events.

Technical Reference(s): OTO-ZZ-00008

Proposed references to be provided to applicants during examination: None

Learning Objective: Control Board Certification – Mod D, D-03 Obj B, C and 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:  
55.41   7    
55.43 \_\_\_\_\_

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Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	0056 AA1.31	
	Importance Rating	3.3	
Ability to operate and / or monitor the following as they apply to the Loss of Offsite Power: PZR heater group control switches			

**Question #10**

Given the following plant conditions:

- The Callaway Plant is responding to a loss of offsite power.
- Both Emergency Diesel Generators have started and loaded onto their respective buses.
- Safety Injection did NOT actuate.
- Pressurizer level is 25%.

The Reactor Operator is attempting to control Pressurizer pressure. What must be done to energize BB HIS-52A, Backup Group B Heaters?

- A. Turn the BB HIS-52A control switch to TRIP.  
Place BB PK-455K, PZR PRESS MASTER CTRL, in Manual and raise setting.  
Return the BB HIS-52A control switch in AUTO.
- B. Reset the NB03 lockout relays.  
Close Breaker NB0208, Fdr Bkr to PG22.  
Leave the BB HIS-52A control switch in AUTO.
- C. Reset the NB01 lockout relays.  
Restore power to NB01.  
Then turn the BB HIS-52A control switch to ON.
- D. Turn the BB HIS-52A control switch to TRIP.  
Close Breaker NB0208, Fdr Bkr to PG22.  
Then turn the BB HIS-52A control switch to ON.

*Justification*

- A. Incorrect, have to take control switch to TRIP then ON to reset heaters
- B. Incorrect, heaters are powered from NB01, have to reset the control switch
- C. Incorrect, EDG restored power, switch to TRIP to reset
- D. Correct

Technical Reference(s): EOP ADD 8

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # \_\_\_\_\_

NRC Site-Specific Written Examination  
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Modified Bank # \_\_\_\_\_  
New  \_\_\_\_\_

Question History: Last NRC Exam  N/A \_\_\_\_\_

Question Cognitive Level:

Memory or Fundamental Knowledge \_\_\_\_\_  
Comprehension or Analysis  \_\_\_\_\_

10 CFR Part 55 Content:

55.41  7 \_\_\_\_\_  
55.43 \_\_\_\_\_

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	0057 AK3.01	
	Importance Rating	4.1	
Knowledge of the reasons for the following responses as they apply to the Loss of Vital AC Instrument Bus: Actions contained in EOP for loss of vital ac electrical instrument bus			

**Question #11**

Given the following plant conditions:

- A reactor startup is in progress
- Source range channels N31 and N32 indicate  $10^4$  CPS
- Intermediate range channels N35 and N36 indicate  $5 \times 10^{-11}$  Amps
- The annunciator 25A, NN01 Inst bus UV, has just alarmed

Which ONE of the following describes the actions that are required for this condition?

- A. Verify reactor trip, AND  
Restore power to NN01 from alternate AC power source
- B. Commence a reactor shutdown to insert all control and shutdown banks, AND  
Restore power to NN01 from alternate AC power source
- C. Verify reactor trip, AND  
Isolate Instrument Inverter NN11
- D. Commence a reactor shutdown to insert all control and shutdown banks, AND  
Isolate Instrument Inverter NN11

*Justification*

- A. Correct
- B. Incorrect. Reactor will trip, 2nd part correct.
- C. Incorrect. Reactor will trip, Shift power to alternate source.
- D. Incorrect. Reactor will trip, Shift power to alternate source.

Technical Reference(s): OTN-NN-00001, OTO-NN-00001

Proposed 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 \_\_\_\_\_

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Reactor Operator

Comprehension or Analysis

10 CFR Part 55 Content:

55.41  5, 10

55.43

Comments:



NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	058 AA1.03	
	Importance Rating	3.1	
Ability to operate and / or monitor the following as they apply to the Loss of DC Power: Vital and battery bus components			

**Question #12**

Given the following plant conditions:

- The Callaway Plant has experienced a Loss of NK01.
- The crew has entered OTO-NK-00002, Loss of Vital 125 VDC Bus.
- Maintenance has determined that there is a fault on Battery NK11.

What is the proper sequence of actions required in accordance with OTO-NK-00002 to allow maintenance on "A" Battery?

- A. Disconnect Battery NK11 by removing control power fuses for its battery output breaker, Place DC Bus NK01 on Battery Charger NK25.
- B. Place DC Bus NK01 on its Battery Charger, Isolate Battery NK11 by opening the battery output breaker.
- C. Energize Charger NK25, Disconnect Battery NK11 by opening the battery output breaker, Place DC Bus NK01 on its Battery Charger.
- D. Disconnect Battery NK11 by opening the battery output breaker, Place DC Bus NK01 on its Battery Charger.

*Justification*

- A. *Incorrect. No fuses in the circuit*
- B. *Incorrect. Improper sequence*
- C. *Incorrect. Improper sequence*
- D. *Correct.*

*OTO-NK-00002 Step A14 RNO directs to disconnect the battery.*

Technical Reference(s): OTO-NK-00002

Proposed 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  
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Reactor Operator

Question Cognitive Level:

Memory or Fundamental Knowledge   X    
Comprehension or Analysis       

10 CFR Part 55 Content:

55.41   7    
55.43       

Comments:

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Reactor Operator

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	062 2.4.31		
	Importance Rating	4.2		
Knowledge of annunciator alarms, indications, or response procedures.				

**Question #13**

Given the following plant conditions:

- The plant is operating at 100%, steady state power.
- The Service Water system is aligned as follows:
  - SW Pump A                      Running
  - SW Pump B                      Running
  - SW Pump C                      Standby
- CSEA2102, Service Water Pump Auto Backup Selector Switch, is in AUTO.
- Annunciator 12A, Service Water Pump Lockout is Lit.

Which ONE of the following will result in an automatic start of Service Water Pump C?

- A. SW Pump "A" lube water pressure 6 psig for 20 seconds.
- B. Securing SW Pump "A" from the MCB.
- C. SW Pump "B" lube water flow 2.0 gpm for 20 seconds.
- D. Securing SW Pump "B" locally.

*Justification*

- A. Correct.
- B. Incorrect. Normal shutdown of a pump does not result in a lockout.
- C. Incorrect. It is a trip, but the setpoint is not low enough to result in a trip.
- D. Incorrect. Local shutdown of a pump does not result in a lockout

Technical Reference(s): OTN-EA-00001 and OTA-RK-00014, Add 12A

Proposed references to be provided to applicants during examination: None

Learning Objective: T61-011-006.6, H

Question Source:     Bank # \_R12305\_\_\_\_\_

                              Modified Bank # \_\_\_\_\_

                              New \_\_\_\_\_

Question History: Last NRC Exam \_\_\_\_ N/A \_\_\_\_\_

Question Cognitive Level:

Memory or Fundamental Knowledge     \_\_X\_\_

Comprehension or Analysis                \_\_\_\_\_

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Reactor Operator

10 CFR Part 55 Content:

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55.43 \_\_\_\_\_

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	0065 2.2.44	
	Importance Rating	4.2	
Loss of Instrument Air / Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.			

**Question #14**

Given the following plant conditions:

- The Callaway Plant is in MODE 6 at reduced inventory to support SG nozzle dam installation prior to core offload.
- RHR Train "B" is in service for cooldown when a loss of instrument air occurs.

Which ONE of the following describes the effects on RHR Train "B" operation and RCS temperature?

- A. CCW flow to the RHR heat exchanger lowers and RCS temperature lowers.
- B. All RHR flow is bypassed around the heat exchanger and RCS temperature rises.
- C. All RHR flow is directed through the heat exchanger and RCS temperature lowers.
- D. CCW flow to the RHR heat exchanger rises and RCS temperature lowers.

*Justification*

- A. *Incorrect, CCW Temp control Valves fail closed*
- B. *Incorrect, bypass valves fail closed*
- C. *Correct.*
- D. *Incorrect, CCW Temp control Valves fail closed*

*EJ FCV-618 (619) fails closed on loss of control air or control power. These valves are also seatless butterfly valves, which will allow 245-gpm flow in the closed position.  
Outlet flow control valves 606/607 fail open, CCW Temp control Valves fail closed*

Technical Reference(s): OTO-KA-00001, Att. 4

Proposed references to be provided to applicants during examination: None

Learning Objective: Residual Heat Removal – EJ, System Description

Question Source: Bank #   R12082    
Modified Bank #             
New           

Question History: Last NRC Exam   N/A  

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

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10 CFR Part 55 Content:

55.41   5    
55.43       

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	077 AA2.09	
	Importance Rating	4.3	
Ability to determine and interpret the following as they apply to Generator Voltage and Electric Grid Disturbances: Operational status of Emergency Diesel Generators			

**Question #15**

Given the following plant conditions:

- The Callaway plant is at 100% power.
- DG “A” has been paralleled with 4160VAC bus NB01 and is carrying 5.8 MWe of load in accordance with OSP-NE-0001A, Standby Diesel Generator “A” Periodic Tests.
- A Category 8 alarm has come in on the switchyard and low voltage is indicated on the Electrical Grid.
- The Transmission Operations Supervisor is contacted and informs the crew that a massive power outage has occurred in the Northeast causing voltage swings on the Electric Grid.
- Shortly after this a Grid disturbance causes a Loss of Offsite Power to the Callaway Plant.

Which ONE of the following describes the status of the “A” Train Safeguards Power system?

- A. NB01 Normal Feeder Breaker will remain CLOSED, NE01 will remain running, “A” train shutdown sequencer will not actuate.
- B. NB01 Normal Feeder Breaker will remain closed, NE01 will stop and then restart, “A” Train LOCA sequencer will start.
- C. NB01 Emergency Supply Breaker will OPEN, NE01 will stop and then restart, “A” Train shutdown sequencer will not actuate.
- D. NB01 Emergency Supply Breaker will remain closed, NE01 will remain running, “A” Train LOCA Sequencer will actuate.

*Justification*

- A. *Correct.*
- B. *Incorrect, D/G will not stop, LOCA sequencer not correct*
- C. *Incorrect, wrong breaker, D/G doesn't stop*
- D. *Incorrect, wrong breaker, wrong sequencer*

Technical Reference(s):

OTO-NB-00004, LOOP to NB01/NB02 with EDG Paralleled

Proposed references to be provided to applicants during examination: None

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Learning Objective:

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  
55.43 \_\_\_\_\_

Comments:



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Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	E04 EK2.2	
	Importance Rating	3.8	
Knowledge of the interrelations between the (LOCA Outside Containment) 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 #16**

A Loss of Coolant Accident (LOCA) outside containment has resulted in RCS subcooling dropping to 0°F. Attempts are being made to determine if the leak has been isolated in accordance with ECA-1.2, LOCA Outside Containment.

Which ONE of the following is the primary indication that the completed actions have been successful?

- A. ECCS flow lowering
- B. Containment Sump level rising
- C. RCS Pressure rising
- D. Pressurizer level rising

*Justification:*

- a. *Incorrect. ECCS flow would not necessarily lower, may stay the same.*
- b. *Incorrect. Not necessarily a LOCA inside containment*
- c. *Correct.*
- d. *Incorrect. Pressurizer may be below indicated level*

Technical Reference(s): ECA-1.2

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # 003D140B02A  
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 \_8, 10\_  
55.43 \_\_\_\_\_

Comments:

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<b>Examination Outline Cross-reference:</b>	<b>Level</b>	<b>RO</b>		<b>SRO</b>
	<b>Tier #</b>	1		
	<b>Group #</b>	1		
	<b>K/A #</b>	E05 EA2.1		
	<b>Importance Rating</b>	3.4		4.4
Ability to determine and interpret the following as they apply to the (Loss of Secondary Heat Sink) Facility conditions and selection of appropriate procedures during abnormal and emergency operations.				

**Question #17**

Which ONE of the following sets of plant parameters will result in a red path on the Heat Sink Status Tree?

A. Containment Pressure is 2 psig

	<u>S/G A</u>	<u>S/G B</u>	<u>S/G C</u>	<u>S/G D</u>
NR Level	0%	6%	6%	12%
FW Flow (lbm/hr)	100K	90K	100K	90K

B. Containment Pressure is 2 psig

	<u>S/G A</u>	<u>S/G B</u>	<u>S/G C</u>	<u>S/G D</u>
NR Level	0%	6%	5%	0%
FW Flow (lbm/hr)	88K	86K	95K	96K

C. Containment Pressure is 4 psig

	<u>S/G A</u>	<u>S/G B</u>	<u>S/G C</u>	<u>S/G D</u>
NR Level	15%	30%	10%	10%
FW Flow (lbm/hr)	83K	90K	90K	90K

D. Containment Pressure is 4 psig

	<u>S/G A</u>	<u>S/G B</u>	<u>S/G C</u>	<u>S/G D</u>
NR Level	5%	20%	15%	15%
FW Flow (lbm/hr)	80K	88K	90K	90K

*Justification:*

- A. *Incorrect. Containment conditions are not adverse. Narrow range level in the S/G #4 is greater than 7%, so the heat sink safety function cannot be worse than yellow. Plausible if applicant applies total FW criteria before evaluating SG levels.*
- B. *Incorrect. Containment conditions are not adverse. Although no S/G narrow range levels are greater than 7%, total FW flow is greater than 355K, so the heat sink safety function cannot be worse than yellow. Plausible if applicant evaluates SG levels and then fails to apply the additional criteria of total FW flow.*

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- C. *Incorrect. Containment conditions are adverse. Although no S/G NR level is greater than 25%, total FW flow is greater than 355K (value for FW flow does not change with adverse containment), so the heat sink safety function cannot be worse than yellow. Plausible if applicant evaluates SG levels and then fails to apply the additional criteria of total FW flow or believes that the required FW flow is greater for adverse containment conditions.*
- D. *Correct. Containment conditions are adverse. No S/G level is greater than 25%NR AND total FW flow is less than 355K.*

Technical Reference(s): CSF-1

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source:      Bank # \_\_\_\_\_  
                                 Modified Bank # \_\_\_\_\_  
                                 New  \_\_\_\_\_

Question History: Last NRC Exam  N/A

Question Cognitive Level:  
    Memory or Fundamental Knowledge       \_\_\_\_\_  
    Comprehension or Analysis                      \_\_\_\_\_

10 CFR Part 55 Content:  
   55.41  \_\_\_\_\_  
   55.43 \_\_\_\_\_

Comments:

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<b>Examination Outline Cross-reference:</b>	<b>Level</b>	<b>RO</b>		<b>SRO</b>
	<b>Tier #</b>	1		
	<b>Group #</b>	1		
	<b>K/A #</b>	E11 EA2.1		
	<b>Importance Rating</b>	3.4		
Ability to determine and interpret the following as they apply to the (Loss of Emergency Coolant Recirculation) Facility conditions and selection of appropriate procedures during abnormal and emergency operations.				

**Question #18**

Given the following plant conditions:

- During a LOCA, emergency coolant recirculation capability was lost and ECA-1.1, Loss of Emergency Coolant Recirculation, is currently in progress.
- A RED path is identified on the CONTAINMENT status tree, and transition to FR-Z.1, Response to High Containment Pressure, is performed.

Which ONE of the following describes the procedure that should be used to operate the containment spray pumps and why?

- A. ECA-1.1, because it provides for REDUCED containment spray.
- B. FR-Z.1, because it provides for GREATER containment spray.
- C. FR-Z.1, because it takes precedence over ECA-1.1.
- D. ECA-1.1, because an ECA should be completed prior to transferring to an FR.

*Justification*

- A. Correct.
- B. Incorrect. FR-Z.1 Step 1 RNO
- C. Incorrect. FR-Z.1 Step 1 RNO
- D. Incorrect. FR-Z.1 Step 1 RNO

Technical Reference(s): ECA-1.1, FR-Z.1

Proposed 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:

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55.41 \_10\_  
55.43 \_\_\_\_\_

Comments:

NRC Site-Specific Written Examination  
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Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	028 AK2.03	
	Importance Rating	2.6	
Knowledge of the interrelations between the Pressurizer Level Control Malfunctions and the following: Controllers and positioners			

**Question #19**

Given the following plant conditions:

- The Callaway Plant is at 75% power, steady state conditions.
- The Pressurizer Backup heaters have automatically energized.

Which ONE of the following describes a potential cause for this action?

- A. Pressurizer Level Transmitter BB LT-0459 fails to 48%.
- B. Pressurizer Pressure Master Controller output fails to 100%.
- C. Pressurizer Level deviation lowering to 5% less than program.
- D. Pressurizer Pressure Transmitter BB PT-0456 fails high.

*Justification*

- A. *Incorrect, This happens to be the program level for 75% power. The candidate will have to calculate program level at 75% and then determine if 48% is > 5% deviation to energize the heaters.*
- B. *Correct, If the pressurizer master controller output fails to 100%, the system would react as if pressure was low, this would energize the B/U heaters.*
- C. *Incorrect, Pressurizer level deviation low does not energize the B/U heaters.*
- D. *Incorrect, PT-456 has no input for controlling the pressurizer B/U heaters.*

Technical Reference(s): OTN-BB-00005

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # \_\_\_\_\_  
Modified Bank # \_\_\_\_\_  
New  \_\_\_\_\_

Question History: Last NRC Exam \_\_\_\_\_ N/A \_\_\_\_\_

Question Cognitive Level:

Memory or Fundamental Knowledge  \_\_\_\_\_  
Comprehension or Analysis \_\_\_\_\_

10 CFR Part 55 Content:

55.41  \_\_\_\_\_  
55.43 \_\_\_\_\_

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	033 AK3.01	
	Importance Rating	3.2	
Knowledge of the reasons for the following responses as they apply to the Loss of Intermediate Range Nuclear Instrumentation: Termination of startup following loss of intermediate range instrumentation			

**Question #20**

Given the following plant conditions:

- A Reactor Startup is in progress following an extended outage.
- During the course of the startup, the RO notes that neither channel of Intermediate Range Nuclear Instrumentation is responding.

Which ONE of the following choices indicates the reason that a power reduction is required?

- A. Protection against a cold water accident is reduced.
- B. Protection against a rod ejection accident is reduced.
- C. Protection against a steam line break accident is reduced.
- D. Protection against an uncontrolled RCCA bank rod withdrawal is reduced.

*Justification*

- A. Incorrect, PRNI basis
- B. Incorrect, PRNI basis
- C. Incorrect, OP Delta T basis
- D. Correct.

Technical Reference(s): TS 3.3.1 Bases

Proposed references to be provided to applicants during examination: None

Learning Objective: Systems SB, Reactor Protection – Reactor Trips

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 \_5, 10\_  
55.43 \_\_\_\_\_

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Comments:



NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	068 AA2.08	
	Importance Rating	3.9	
Ability to determine and interpret the following as they apply to the Control Room Evacuation: S/G pressure			

**Question #21**

Given the following plant conditions:

- The Callaway plant was at 100% power
- The control room was evacuated due to a fire
- OTO-ZZ-00001, Control Room Inaccessibility, has been entered
- The crew has been directed to maintain temperature at 557°F using Steam Dumps

Which ONE of the following Steam Generator pressures would be indicative of maintaining RCS temperature at the desired value?

- A. 1030 psig
- B. 1090 psig
- C. 1125 psig
- D. 1185 psig

*Justification*

- A. Incorrect. Pressure for 550 degrees is the P-12 interlock
- B. Correct. Pressure for 557 degrees – Condenser Steam Dumps available
- C. Incorrect. Pressure if relying on the Atmos Steam Dumps
- D. Incorrect. Pressure if using SG safety's to control temperature

Technical Reference(s): OTO-ZZ-00001

Proposed references to be provided to applicants during examination: Steam Tables

Learning Objective:

Question Source: Bank # \_\_\_\_\_  
Modified Bank # \_\_\_\_\_  
New  \_\_\_\_\_

Question History: Last NRC Exam  N/A

Question Cognitive Level:  
Memory or Fundamental Knowledge   
Comprehension or Analysis

10 CFR Part 55 Content:  
55.41  5  
55.43

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO	
	Tier #	1		
	Group #	2		
	K/A #	074 EA2.07		
	Importance Rating	4.1		
Ability to determine or interpret the following as they apply to a Inadequate Core Cooling: The difference between a LOCA and inadequate core cooling, from trends and indicators				

**Question #22**

Given the following plant conditions:

- A LOCA has occurred.
- ALL RCPs are STOPPED.
- RVLIS indication is NOT available.

Which ONE of the following parameters would indicate Inadequate Core Cooling conditions?

- A. CETC Temperature 712°F  
RCS pressure 700 psig  
No ECCS injection is available
- B. Cold Leg Temperature 340°F  
RCS pressure 100 psig  
ECCS injection is available
- C. CETC Temperature 550°F  
RCS Pressure 1000 psig  
ECCS injection is available
- D. Cold Leg Temperature 547°F  
RCS Pressure 1500 psig  
No ECCS injection is available

*Justification*

- A. *Correct.*
- B. *Incorrect.* -12°F subcooling
- C. *Incorrect.* -3°F subcooling
- D. *Incorrect.* Subcooled

Technical Reference(s): CSF-1

Proposed references to be provided to applicants during examination: None

Learning Objective:

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 \_\_\_X\_\_\_  
Comprehension or Analysis \_\_\_\_\_

10 CFR Part 55 Content:

55.41 \_5, 14\_  
55.43 \_\_\_\_\_

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO	
	Tier #	1		
	Group #	2		
	K/A #	E03 EA1.1		
	Importance Rating	4.0		
Ability to operate and / or monitor the following as they apply to the (LOCA Cooldown and Depressurization) Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.				

**Question #23**

Given the following plant conditions:

- A Small Break LOCA has occurred.
- Due to a failure of Voltage Restoration for Buses PA01 and PA02, these buses are deenergized.
- The actions of ES-1.2, Post LOCA Cooldown and Depressurization, are in progress.
- Charging Pumps "A" and "B" are running with suction aligned to the RWST.
- Both RHR Pumps are stopped in AUTO.
- Both SI Pumps are running.
- The crew is ready to depressurize the RCS to refill the Pressurizer.

Which ONE of the following describes how this depressurization will be achieved?

- A. Utilize Pressurizer Auxiliary Spray Valve, BG HV-8145, to spray down the Pressurizer steam space.
- B. Utilize BOTH Pressurizer Spray Control valves, BB PCV-455B AND BB PCV-455C, to spray down the Pressurizer steam space.
- C. Open BOTH Pressurizer PORVs, BB PCV-455A and BB PCV-456A to vent the Pressurizer.
- D. Open ONE Pressurizer PORV, BB PCV-455A or BB PCV-456A to vent the Pressurizer.

*Justification*

- A. *Incorrect. BGHV-8145 is a method for depressurization and for a SGTR is utilized as the third method. However, it is not used in this case since the requirements place a limit of spray dT, and letdown is required to be in service if Aux Spray is to be used.*
- B. *Incorrect. This is the "normal" method used to depressurize the RCS. However, with Buses PA01 and PA02 deenergized, the RCPs are NOT running and are therefore unable to provide the driving head for normal sprays.*
- C. *Incorrect. Opening TWO PORVS is not an appropriate action. This action has a less stable depressurization rate and raises the probability of a PORV failing to close.*
- D. *Correct.*

Technical Reference(s): ES-1.2

NRC Site-Specific Written Examination  
Callaway Plant  
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Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # \_\_\_\_\_  
Modified Bank # \_\_\_\_\_  
New  \_\_\_\_\_

Question History: Last NRC Exam  N/A \_\_\_\_\_

Question Cognitive Level:  
Memory or Fundamental Knowledge \_\_\_\_\_  
Comprehension or Analysis  \_\_\_\_\_

10 CFR Part 55 Content:  
55.41  7 \_\_\_\_\_  
55.43 \_\_\_\_\_

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO	
	Tier #	1		
	Group #	2		
	K/A #	E09 EK1.3		
	Importance Rating	3.3		

Knowledge of the operational implications of the following concepts as they apply to the (Natural Circulation Operations) Annunciators and conditions indicating signals, and remedial actions associated with the (Natural Circulation Operations).

**Question #24**

Given the following plant conditions:

The plant was operating at 98% power when a loss of off-site power caused a reactor trip.

Twenty minutes after the trip the following plant conditions exist.

- |                            |           |          |
|----------------------------|-----------|----------|
| ▪ RCS Pressure             | 2235 psig | STABLE   |
| ▪ RCS Hot Leg Temperature  | 564°F     | LOWERING |
| ▪ RCS Cold Leg Temperature | 560°F     | LOWERING |
| ▪ Core Exit Temperature    | 580°F     | LOWERING |
| ▪ Steam Generator Pressure | 1128 psig | LOWERING |

Which ONE of the following describes plant conditions?

- A. Heat removal IS BEING maintained by Condenser Steam Dumps. Natural Circulation EXISTS.
- B. Heat removal MAY BE established by opening the Atmospheric Steam Dumps. Natural Circulation DOES NOT exist.
- C. Heat removal MAY BE established by opening the Condenser Steam Dumps. Natural Circulation DOES NOT exist.
- D. Heat removal IS BEING maintained by Atmospheric Steam Dumps. Natural Circulation EXISTS.

*Justification*

- A. Incorrect. Condenser not available.
- B. Incorrect. NC does exist.
- C. Incorrect. NC does exist. Condenser not available
- D. Correct.

Technical Reference(s): ES-0.2, EOP ADD 1

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank #   R8678    
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 \_\_\_\_\_

Comprehension or Analysis   X  

10 CFR Part 55 Content:

55.41 \_8, 10\_

55.43 \_\_\_\_\_

Comments:



NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO	
	Tier #	1		
	Group #	2		
	K/A #	E13 2.1.7		
	Importance Rating	4.4		
Steam Generator Overpressure - Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.				

**Question #25**

Given the following plant conditions:

- Reactor has been manually tripped due to a secondary system malfunction
- E-0 has been performed and a transition made to ES-0.1, Reactor Trip Response
- The crew has entered FR-H.2, Response to Steam Generator Overpressure
- The crew is preparing to dump steam from the affected steam generator

Which ONE of the following describes the effect of dumping steam if the affected SG NR level is >94%?

- A. Will be ineffective in lowering SG pressure since the SG water is likely subcooled.
- B. Will cause a rapid pressure drop in the RCS, potentially resulting in a safety injection.
- C. May result in two phase flow and water hammer, potentially damaging pipes and valves.
- D. May cause an uncontrolled radiation release since it is likely that the steam generator is ruptured.

*Justification*

- A. *Incorrect. Water is saturated not subcooled.*
- B. *Incorrect. Not a rapid drop.*
- C. *Correct.*
- D. *Incorrect. Plausible since some tube leakage is assumed in analysis*

Technical Reference(s): BD-FR-H.3, BD-FR-H.2

Proposed 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   5  

55.43       

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO	
	Tier #	1		
	Group #	2		
	K/A #	E15 EA1.2		
	Importance Rating	2.7		
Ability to operate and / or monitor the following as they apply to the (Containment Flooding) Operating behavior characteristics of the facility				

**Question #26**

Given the following plant conditions:

- A LOCA has occurred.
- An ORANGE Path has developed on Containment Critical Safety Function due to Sump level.
- All Auto Actions have occurred and have not been overridden.
- Annunciator 51D, CCW Srg Tk A Lev HiLo, is lit along with other expected alarms.
- Containment Pressure peaked at 25 psig.

In accordance with FR-Z.2, Response To Containment Flooding, which ONE of the following would cause this condition?

- A. Service Water Leak inside Containment
- B. Fire Protection System Leak inside Containment
- C. Component Cooling Water Leak inside Containment
- D. Containment Spray Line Rupture inside Containment

*Justification*

- A. *Incorrect. ESW is the supply for Ctmt loads. Service water is isolated.*
- B. *Incorrect. FP Does supply components in Ctmt. FP alarms are not expected for a LOCA.*
- C. *Correct.*
- D. *Incorrect. Containment Spray actuates at 27 psig. Setpoint not reached*

*Ctmt Sump level of 106" = Orange Path*

Technical Reference(s): FR-Z.2

Proposed 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 \_\_7\_\_

55.43 \_\_\_\_\_

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO	
	Tier #	1		
	Group #	2		
	K/A #	E16 EK2.1		
	Importance Rating	3.0		
Knowledge of the interrelations between the (High Containment Radiation) and the following: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.				

**Question #27**

The Callaway Plant has experienced a large Loss of Coolant Accident (LOCA).

Containment Pressure, Temperature, Humidity, and Radiation are all reading abnormally high due to the LOCA conditions. The Reactor Operator has made the announcement the plant is now in "Adverse Containment"

Which ONE of the following describes the proper use of Adverse Containment?

Once in Adverse Containment . . . .

- A. Due to pressure, adverse values must be used for the duration of the event.
- B. Due to temperature, adverse values can be used when temperature lowers to a normal value.
- C. Due to humidity, adverse values can be used when humidity lowers to normal to a normal value.
- D. Due to radiation, adverse values must be used for the duration of the event.

*Justification*

- A. *Incorrect, can be exited once pressure lowers.*
- B. *Incorrect, does not determine adverse containment*
- C. *Incorrect, does not determine adverse containment*
- D. *Correct, due unreliability of the instrumentation*

Technical Reference(s): ODP-ZZ-00025

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # \_\_\_003D040R01C\_\_\_  
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\_\_  
55.43 \_\_\_\_\_

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO	
	Tier #	2		
	Group #	1		
	K/A #	003 K4.02		
	Importance Rating	2.5		
Knowledge of RCPS design feature(s) and/or interlock(s) which provide for the following: Prevention of cold water accidents or transients				

**Question #28**

OTG-ZZ-00001, Plant Heatup Cold Shutdown to Hot Standby, requires RCS cold leg temperatures to be greater than 275°F to start a Reactor Coolant Pump unless the Steam Generator temperature is within 50°F of RCS temperature.

This criteria will prevent . . .

- A. rapid depressurization of the RCS and subsequent injection of non-condensable gases upon RCP start.
- B. a subsequent reactivity excursion on RCP start.
- C. pressurized thermal shock of the Reactor Vessel and/or Steam Generators.
- D. a low temperature overpressure event due to a thermal transient when an RCP is started.

*Justification:*

- A. *Incorrect per reference. See below*
- B. *Incorrect per reference. See below*
- C. *Incorrect per reference. See below*
- D. *Correct per reference. See below*

*RCP starting limitations include the following:*

*A reactor coolant pump should NOT be started with any RCS Cold Leg temperature less than or equal to 275°F, UNLESS the secondary side water temperature of each steam generator 50°F above each of the RCS cold leg temperatures.*

*3.4.6 Basis - Note 2 requires that the secondary side water temperature of each SG be = 50°F above each of the RCS cold leg temperatures before the start of a reactor coolant pump (RCP) with any RCS cold leg temperature = 275°F. This restriction is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started.*

Technical Reference(s): OTG-ZZ-00001, Plant Heatup Cold Shutdown to Hot Standby, TS 3.4.6 and TS bases 3/4.4.6.

Proposed 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  \_\_\_\_\_

Question History: Last NRC Exam  N/A \_\_\_\_\_

Question Cognitive Level:

Memory or Fundamental Knowledge  \_\_\_\_\_

Comprehension or Analysis  \_\_\_\_\_

10 CFR Part 55 Content:

55.41  7 \_\_\_\_\_

55.43  \_\_\_\_\_

Comments:



NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	004 K2.01	
	Importance Rating	2.9	
Knowledge of bus power supplies to the following: Boric acid makeup pumps			

**Question #29**

Which ONE of the following describes the power supply for 'A' Boric Acid Transfer Pump?

- A. NG01A
- B. NG02A
- C. PG19N
- D. PG20N

*Justification:*

- A. Correct.
- B. Incorrect, this is the supply for B pump.
- C. Incorrect, this is the supply for RMW Pump A, 480V Non-Safety Related.
- D. Incorrect, this is the supply for RMW Pump B, 480V Non-Safety Related.

*PBG02A is powered off NG01A and PBG02B is powered off NG02A. Note that the pumps are load shed upon receipt of an SI signal and the breakers must be manually closed.*

Technical Reference(s): EOP ADD 8

Proposed references to be provided to applicants during examination: None

Learning Objective: CVCS System Description - Objective F

Question Source: Bank # \_\_\_\_\_  
Modified Bank # \_\_\_\_\_  
New  \_\_\_\_\_

Question History: Last NRC Exam \_\_\_\_\_

Question Cognitive Level:  
Memory or Fundamental Knowledge   
Comprehension or Analysis \_\_\_\_\_

10 CFR Part 55 Content:  
55.41  7  
55.43 \_\_\_\_\_

Comments:

NRC Site-Specific Written Examination  
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Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	005 K2.03	
	Importance Rating	2.7*	
Knowledge of bus power supplies to the following: RCS pressure boundary motor-operated valves			

**Question #30**

ES-1.4, Transfer to Hot Leg Recirculation, Step 1 has the operator check if NG02 is energized.

If NG02 cannot be energized, Residual Heat Removal (RHR) cannot be placed into Hot Leg Recirculation.

Which ONE of the following is the reason that RHR cannot be placed into Hot Leg Recirculation with NG02 de-energized?

- A. EJ HV-8840, RHR Combined Recirculation Isolation Valve, cannot be opened.
- B. EJ HV-8716B, "B" Train RHR Recirculation Isolation Valve, cannot be opened.
- C. EG HV-102, Component Cooling Water to "B" RHR Heat Exchanger cannot be opened.
- D. EM HV-8802B, Safety Injection Pump Discharge Isolation Valve cannot be opened.

*Justification:*

- A. *Correct. Common disch to establish Hot Leg Recirc*
- B. *Incorrect. Need the valve, but "A" train could be used*
- C. *Incorrect. Water may heat up, but could still supply Hot Leg Recirc*
- D. *Incorrect. Need the valve, but "A" train could be used*

Technical Reference(s): ES-1.4

Proposed references to be provided to applicants during examination: None

Learning Objective: RHR System Description, Obj C.

Question Source: Bank # \_\_\_\_\_  
Modified Bank # \_\_\_\_\_  
New  \_\_\_\_\_

Question History: Last NRC Exam  N/A

Question Cognitive Level:  
Memory or Fundamental Knowledge   
Comprehension or Analysis

10 CFR Part 55 Content:  
55.41  7   
55.43

NRC Site-Specific Written Examination  
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Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

<b>Examination Outline Cross-reference:</b>	<b>Level</b>	<b>RO</b>		<b>SRO</b>
	<b>Tier #</b>	2		
	<b>Group #</b>	1		
	<b>K/A #</b>	006 K6.01		
	<b>Importance Rating</b>	3.4		
Knowledge of the effect of a loss or malfunction on the following will have on the ECCS: BIT/borated water sources				

**Question #31**

Given the following plant conditions:

- The Callaway Plant is at NOP/NOT
- The crew is preparing to withdraw control rods for a plant startup
- Chemistry Lab reports that the RWST C<sub>B</sub> is 2325 ppm

Which ONE of the following identifies the **MINIMUM** volume and boron concentration required in the Boric Acid Storage Tank?

- | <u>Volume</u>     | <u>Concentration</u> |
|-------------------|----------------------|
| A. 17,900 gallons | 7800 ppm boron       |
| B. 16,900 gallons | 7100 ppm boron       |
| C. 17,900 gallons | 7100 ppm boron       |
| D. 16,900 gallons | 7600 ppm boron       |

*Justification:*

- A. *Incorrect. Acceptable volume, Concentration too high.*  
 B. *Incorrect. Volume too low, Acceptable concentration.*  
 C. *Correct. Min volume 17,658 gals, Concentration between 7000-7700 ppm.*  
 D. *Incorrect. Volume too low, Concentration acceptable.*

*Requires candidate to determine mode and know the FSAR limits*

Technical Reference(s): FSAR 16.1

Proposed references to be provided to applicants during examination: None

Learning Objective: CVCS System Description

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  \_\_\_\_\_

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

10 CFR Part 55 Content:

55.41   5  

55.43       

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	006 A3.05	
	Importance Rating	4.2	
Ability to monitor automatic operation of the ECCS, including: Safety Injection Pumps			

**Question #32**

Given the following plant conditions:

- A Large Break LOCA has occurred.
- RWST level is 11%.
- All actions of E-0, Reactor Trip or Safety Injection, and E-1, Loss of Reactor or Secondary Coolant, have been performed by the crew.

Which ONE of the following describes the current status of ECCS pumps?

- A. RHR Pumps running, taking suction from the Containment Recirc Sump; Charging/SI Pumps running taking suction from the RWST.
- B. RHR Pumps stopped with Containment Recirc Sump suction valves open; Charging/SI Pumps running taking suction from the RWST.
- C. RHR Pumps running, taking suction from the Containment Recirc Sump; Charging/SI Pumps running taking suction from the RHR Pump discharge.
- D. RHR Pumps stopped with Containment Recirc Sump suction valves open; Charging/SI Pumps running taking suction from the RHR Pump discharge.

*At 36% RWST level, RHR pumps are manually tripped and Containment Recirc sump isolation valves automatically open. RWST suction valves to RHR will auto close when Containment Recirc Sump valves are open. Alignment for Charging/SI pumps remains as is until ES-1.3 is performed, and piggyback operations are initiated.*

*Justification*

- A. *Incorrect. Pumps not aligned to RWST*
- B. *Incorrect. Pumps running. Not aligns to RWST*
- C. *Correct.*
- D. *Incorrect. Pumps running.*

Technical Reference(s): ES-1.3

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # \_\_\_\_\_  
Modified Bank # \_\_\_\_\_  
New  \_\_\_\_\_

Question History: Last NRC Exam \_\_\_\_\_ N/A \_\_\_\_\_

NRC Site-Specific Written Examination  
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Question Cognitive Level:

Memory or Fundamental Knowledge \_\_\_\_\_  
Comprehension or Analysis   X  

10 CFR Part 55 Content:

55.41   7    
55.43 \_\_\_\_\_

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO	
	Tier #	2		
	Group #	1		
	K/A #	007 K5.02		
	Importance Rating	3.1		
Knowledge of the operational implications of the following concepts as they apply to PRTS: Method of forming a steam bubble in the PZR				

**Question #33**

The Callaway Plant is preparing to heat up after a refueling outage.

- Preparations have begun to draw a bubble in the Pressurizer and Pressurizer Heaters have now been energized in accordance with OTG-ZZ-00001, Plant Heatup Cold Shutdown to Hot Standby
- Indicated Pressurizer Level starts lowering as the bubble starts to form

By which ONE of the following methods is Pressurizer level lowering?

- A. An Open PORV is venting fluid to the Pressurizer Relief Tank.
- B. Pressurizer outsurge is filling the Steam Generator U-Tubes.
- C. Increasing Auxiliary Spray flow which lowers Pressurizer temperature.
- D. Cold Calibrated Level Instruments indicate lower as the Pressurizer heats up.

*Justification:*

- A. *Incorrect, Level would rise*
- B. *Correct.*
- C. *Incorrect, Aux Spray is not inservice at this time*
- D. *Incorrect, this has no effect Cold cal is just slightly lower than Hot cal*

Technical Reference(s): OTG-ZZ-00001

Proposed 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 \_\_\_5\_\_\_  
55.43 \_\_\_\_\_



NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	007 2.1.28	
	Importance Rating	4.1	
Pressurizer Relief Tank / Quench Tank System - Knowledge of the purpose and function of major system components and controls.			

**Question #34**

Which ONE of the following is the reason for maintaining a nitrogen blanket on the Pressurizer Relief Tank (PRT)?

- A. Limits the peak pressure of the PRT to 50 psig following a design basis discharge to the tank.
- B. Minimizes the possibility of forming an explosive mixture of hydrogen and oxygen in the PRT.
- C. Ensures NPSH when circulating water from the PRT through the Reactor Coolant Drain Tank HX.
- D. Reduces the amount of hydrogen released to containment if overpressure causes rupture of the rupture disks.

*Justification:*

- A. *Incorrect, basis for volume of nitrogen*
- B. *Correct*
- C. *Incorrect, basis for RCDT Pumps*
- D. *Incorrect, hydrogen released to containment from the RCS is considered in design analysis*

*3 psig to prevent air in-leakage. The nitrogen blanket will minimize the possibility of hydrogen, coming out of solution, combining with oxygen to form an explosive mixture.*

Technical Reference(s): OTN-BB-00004

Proposed references to be provided to applicants during examination: None

Learning Objective: RCS-B.9, 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  7 \_\_\_\_\_  
55.43 \_\_\_\_\_

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

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

**Question #35**

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

- A. Low flow, High flow, SIS
- B. High flow, Low-Low level in CCW train "B" surge tank, SIS
- C. High flow, Low-Low level in CCW train "A" or "B" surge tank
- D. Low-Low Level in CCW train "A" surge tank, Low flow, SIS

*Justification:*

- A. *Incorrect, low flow will not close the valves*
- B. *Correct*
- C. *Incorrect, low level in "A" Surge tank closes EG HV 69A and B*
- D. *Incorrect, low level in "A" Surge tank closes EG HV 69A and B, Low flow does not close valves*

Technical Reference(s): M-22EG01, M-22EG03

Proposed references to be provided to applicants during examination: None

Learning Objective: System description System EG, obj. B

Question Source: Bank # \_\_\_ Wolf Creek #Q15714\_\_\_  
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 \_\_\_5\_\_\_  
55.43 \_\_\_\_\_

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO	
	Tier #	2		
	Group #	1		
	K/A #	010 K4.01		
	Importance Rating	2.7		
Knowledge of PZR PCS design feature(s) and/or interlock(s) which provide for the following: Spray valve warm-up .				

**Question #36**

Which ONE of the following provides the correct reasons for maintaining a minimum spray bypass flow to the Pressurizer?

- A. Reduce thermal shock to the spray nozzle.  
Equalize boron between Pressurizer and the RCS.
- B. Prevent excessive cooling to the surge line.  
Reduce the  $\Delta P$  across the spray valves.
- C. Prevent excessive cooling to the spray line.  
Ensure that the backup heaters cycles on.
- D. Minimize stress to the surge line thermal sleeve.  
Remove gases from the RCS.

*Justification*

- A. Correct.
- B. Incorrect, see below description
- C. Incorrect, see below description
- D. Incorrect, see below description

*Each spray valve is paralleled with a manual throttle valve which allows a small continuous flow of 1/2 gpm for each leg through the spray lines. This flow aids in reducing the thermal stresses and thermal shock when the spray valves open, and helps maintain uniform water chemistry and temperature in the pressurizer. The spray nozzle is further protected from thermal shock by low alarm temperature sensors that alert the operator to an insufficient bypass flow condition. The piping layout to the nozzle forms a water seal, preventing steam buildup back to the spray valve.*

*Requires candidate to know the different purposes for spray bypass flow.*

Technical Reference(s): SD

Proposed 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:

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    
55.43       

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	012 2.1.17		
	Importance Rating	3.9		
Reactor Protection System - Ability to make accurate, clear, and concise verbal reports.				

**Question #37**

The unit is in MODE 3 preparing to withdraw rods to enter MODE 2.

I&C is performing Source Range Surveillance testing.

Which ONE of the following describes the response of the Reactor Protection system and the reports made to the SRO?

If the control power fuses blow on a source range channel, the source range high flux trip will:

- A. Not actuate; the trip will NOT be able to be bypassed at the source range drawer.
- B. Actuate; the trip will be able to be bypassed at the source range drawer.
- C. Actuate; the trip will NOT be able to be bypassed at the source range drawer.
- D. Not actuate; the trip will be able to be bypassed at the source range drawer.

*Justification*

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

*On a loss of the control power the bistable will trip. This is a widely misunderstood concept. There is much confusion over whether a loss of control power or instrument power trip the bistable. It doesn't make any difference whether the channel is bypassed or not, loss of control power trips the bistable.*

Technical Reference(s): OTO-SE-00001

Proposed references to be provided to applicants during examination: None

Learning Objective: System Description – SE, obj. B

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\_\_

55.43 \_\_\_\_\_

Comments:



NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	012 K3.04	
	Importance Rating	3.8*	
Knowledge of the effect that a loss or malfunction of the RPS will have on the following: ESFAS			

**Question #38**

Pressurizer Pressure Protection Channel 455 fails and is properly removed from service.

Which ONE of the following identifies the RPS and ESF actuation logic required, from the remaining in-service channels, to initiate a reactor trip and safety injection on low pressurizer pressure?

- A. Reactor Trip - 1/3; Safety Injection -1/3
- B. Reactor Trip - 1/2; Safety Injection -1/2
- C. Reactor Trip - 2/3; Safety Injection -2/3
- D. Reactor Trip - 1/3; Safety Injection -1/2

*Trip and SI is normally 2/4 for Pzr pressure. Channel 455 feeds both circuits. When a protection channel is removed from service, bistables are tripped in all cases except for the AUTO RB Spray actuation. Thus, AUTO SI will occur if either of the two remaining bistables trip and Reactor trip will occur if either of the 3 remaining bistables trip. 1/2 and 2/3 are credible distractors because the applicant must know what state bistables will be in after action is taken.*

*Justification*

- A. Correct.
- B. Incorrect, wrong initial logic
- C. Incorrect, assumes no effect
- D. Incorrect, wrong SI logic

Technical Reference(s): 7250D64-S006

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # \_\_\_\_\_  
Modified Bank # \_\_\_\_\_  
New  \_\_\_\_\_

Question History: Last NRC Exam  N/A

Question Cognitive Level:

Memory or Fundamental Knowledge  \_\_\_\_\_  
Comprehension or Analysis  \_\_\_\_\_

10 CFR Part 55 Content:

55.41  7   
55.43

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	013 K6.01	
	Importance Rating	2.7*	
Knowledge of the effect of a loss or malfunction on the following will have on the ESFAS: Sensors and detectors			

**Question #39**

Given the following plant conditions:

- Reactor power is 100%.
- The RO notices that RWST level instrument BN LT-930 failed off-scale high at 1135.
- All other RWST level indicators (BN LT-931, 932, 933) are at 99%.

Which ONE of the following describes the initial impact of this failure?

- A. Train A RHR suction swapover is disabled, Train B RHR suction swapover is operable.
- B. Train B RHR suction swapover is disabled, Train A RHR suction swapover is operable.
- C. Both trains of RHR suction swapover are inoperable.
- D. Both trains of RHR suction swapover are operable.

*Justification:*

- A. *Incorrect. 2/4 logic required for each swapover valve. Both trains are operable.*
- B. *Incorrect. 2/4 logic required for each swapover valve. Both trains are operable.*
- C. *Incorrect. 2/4 logic required for each swapover valve. Both trains are operable.*
- D. *Correct. 2/4 logic required for each swapover valve. This is satisfied with the remaining 3 level instruments.*

*The RWST is supplied with four level indication channels (LT-930, 931, 932 and 933). All four channels are displayed on the MCB, channels LT-930 and LT-931 also feed a level recorder on the MCB.*

Technical Reference(s): 8756D37 S038, TS 3.3.2 Condition K bypassed for 12 hours (testing), restore in 72 hours

Proposed references to be provided to applicants during examination: None

Learning Objective: RWST Objective C

Question Source: Bank # \_\_\_\_\_  
Modified Bank # \_\_\_\_\_  
New  \_\_\_\_\_

Question History: Last NRC Exam  N/A

Question Cognitive Level:  
Memory or Fundamental Knowledge \_\_\_\_\_  
Comprehension or Analysis

10 CFR Part 55 Content:

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

55.41 \_\_7\_\_  
55.43 \_\_\_\_\_

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	022 K1.01	
	Importance Rating	3.5	
Knowledge of the physical connections and/or cause-effect relationships between the CCS and the following systems: SWS/cooling system			

**Question #40**

The Callaway plant was initially operating at 100% power

- A Small Break Loss of Coolant Accident (SBLOCA) occurs.
- Containment pressure increases to 4.5 psig.
- RCS pressure then equalizes at 2020 psig.

Which ONE of the following describes the status of the containment cooling system?

- A. CRDM fans A and C are running in fast speed.
- B. CRDM fans B and D are running in slow speed.
- C. ESW flow to Containment coolers increases to approximately 3500 gpm.
- D. ESW flow to Containment coolers remains at approximately 1000 gpm.

*Justification:*

- A. *Incorrect. CRDM fans are single speed, but H2 mixing and Containment Coolers do have fast and slow speeds, this is a misconception by the Operators*
- B. *Incorrect. CRDM fans B and D are load shed*
- C. *Correct. An SI will be actuated on 4.5 psig sending 3500 gpm of water for containment cooling.*
- D. *Incorrect. ESW not running until the SI is actuated*

Technical Reference(s): EF-1, ESW

Proposed references to be provided to applicants during examination: None

Learning Objective: Cont. Vent , Objective D

Question Source: Bank # \_0110400D03A\_  
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 \_3, 5, 7\_  
55.43 \_\_\_\_\_

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	026 A2.03		
	Importance Rating	4.1		
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 ESF				

**Question #41**

Given the following plant conditions:

- The plant is at 100% power with the "B" and "D" Containment Air Cooling Fans out of service.
- An undervoltage occurs on NB01 and "A" Diesel fails to start.
- A Large Break Loss of Coolant Accident occurs and Containment Spray fails to automatically actuate.

Which ONE of the following predicts the effects on the Containment due to these malfunctions? What action will the operators take to mitigate these effects on the containment?

- A. Containment pressure will exceed 40 psig with no operator action.  
The operator will start Containment spray manually per E-0, Reactor Trip or Safety Injection.
- B. Containment pressure will remain less than 40 psig with no operation action.  
The operator will start Containment spray per E-0, Reactor Trip or Safety Injection.
- C. Containment pressure will exceed 40 psig with no operation action.  
The operator will start Containment spray per FR-Z.1, Response to High Containment pressure on a Red Path.
- D. Containment pressure will remain less than 40 psig with no operation action.  
The operator will start Containment spray per FR-Z.1, Response to High Containment Pressure on an Orange Path.

*Justification*

- A. Correct.
- B. Incorrect, pressure will go above 40
- C. Incorrect, E-0 not FR-Z.1
- D. Incorrect, pressure will go above 40 and E-0 not FR-Z.1

Technical Reference(s): BD-E-0, Attachment A, TS Bases B 3.6 and T61.0110.6, Containment Spray System

Proposed references to be provided to applicants during examination: None

Learning Objective: Mitigating Core Damage, C-10

NRC Site-Specific Written Examination  
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Question Source: Bank # \_\_\_\_\_  
Modified Bank # \_\_\_\_\_  
New  \_\_\_\_\_

Question History: Last NRC Exam  N/A

Question Cognitive Level:  
Memory or Fundamental Knowledge   
Comprehension or Analysis

10 CFR Part 55 Content:  
55.41  5   
55.43

Comments:



NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	039 A1.09	
	Importance Rating	2.5*	
Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MRSS controls including: Main steam line radiation monitors			

**Question #42**

Given the following plant conditions:

- The Callaway Plant is at 100% RTP
- GT-RE-31, CTMT Atmosphere indicates  $3.35 \times 10^{-13}$  uCi/ml
- GE-RE-92, Condenser Air Discharge Monitor indicates  $5.5 \times 10^1$  uCi/ml and showing a rising trend
- Main Steam Line Monitors indicates the following:
  - AB-RE-16A, 0.1 gal/day
  - AB-RE-16B, 10 gal/day
  - AB-RE-16C, 0.1 gal/day
  - AB-RE-16D, 0.1 gal/day
- RCS Iodine-131 last sample results indicate 23 uCi/ml

Which ONE of the following best describes the event and mitigating strategy?

**EVENT**

**STRATEGY**

- |                            |   |
|----------------------------|---|
| A. SG tube leak on loop 4. | Implement OTO-BB-00001, Steam Generator Tube Leak.                |
| B. SG tube leak on loop 2. | Implement OTO-BB-00001, Steam Generator Tube Leak.                |
| C. High RCS Activity.      | Implement OTO-BB-00005, RCS High Activity.                        |
| D. RCS leak.               | Implement OTO-BB-00003, Reactor Coolant System Excessive Leakage. |

**JUSTIFICATION:**

- A. *Incorrect, No indications to support - rad mon reading, normal on Loop 1.*
- B. *Correct.*
- C. *Incorrect; No indications to support - RCS activity normal for current condition.*
- D. *Incorrect, No indications to support - CTMT Atmosphere indicates normal.*

Technical Reference(s): OTO-BB-00001

Proposed 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  \_\_\_\_\_

Question History: Last NRC Exam  N/A

Question Cognitive Level:  
Memory or Fundamental Knowledge   
Comprehension or Analysis

10 CFR Part 55 Content:  
55.41  5   
55.43

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO	
	Tier #	2		
	Group #	1		
	K/A #	059 K3.02		
	Importance Rating	3.6		
Knowledge of the effect that a loss or malfunction of the MFW will have on the following: AFW system				

**Question #43**

Given the following plant conditions:

- The Callaway Plant is at 75% reactor power
- S/G water level control is in AUTOMATIC for all S/Gs
- The reactor trips due to high pressurizer pressure

Which ONE of the following describes the expected response on S/G levels? (Assume NO operator action).

- A. S/G levels initially rise due to swell. TDAFW FCVs and MDAFW FCVs will modulate to maintain >7% narrow range steam generator level.
- B. S/G levels initially lower due to shrink. MDAFW FCVs will modulate to maintain 52% narrow range steam generator level. TDAFW will feed S/Gs until manual action is taken.
- C. S/G levels initially rise due to swell. MDAFW FCVs will modulate to maintain 52% narrow range steam generator level. TDAFW will feed S/Gs until manual action is taken.
- D. S/G levels initially lower due to shrink. MDAFW and TDAFW FCVs will feed S/Gs until manual action is taken.

*Justification*

- A. *Incorrect. No AMSAC start.*
- B. *Incorrect. Wrong Setpoint*
- C. *Incorrect. No automatic level control*
- D. *Correct.*

*S/G Level decrease due to FRV's going closed and Shrink. S/G levels decrease to LO-LO level SP. Low tagv and SGWL combine for FWI, AFW Starts and restores level.*

*0% NR = 73.9 WR  
50% NR = 86.9 WR*

*AMSAC not armed below 40% power*

Technical Reference(s): ODP-ZZ-00030

Proposed 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  \_\_\_\_\_

Question History: Last NRC Exam  N/A

Question Cognitive Level:  
Memory or Fundamental Knowledge   
Comprehension or Analysis

10 CFR Part 55 Content:  
55.41  7   
55.43

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO	
	Tier #	2		
	Group #	1		
	K/A #	059 A4.12		
	Importance Rating	3.4		
Ability to manually operate and monitor in the control room: Initiation of automatic feedwater isolation				

**Question #44**

Given the following plant conditions:

- Reactor power is 8%
- Turbine is rolling at 1800 rpm
- Generator output breakers are OPEN
- "A" SG narrow range level (all indicators) is 80%
- "B" SG narrow range level (all indicators) is 88%
- "C" SG narrow range level (all indicators) is 76%
- "D" SG narrow range level (all indicators) is 93%

Which ONE of the following describes the plant response to the above conditions?

- A. Turbine trip, Reactor trip and Feedwater pumps trip.
- B. FRV's close & bypass valves open and Feedwater pumps trip.
- C. Turbine trip, Reactor trip and FRV & bypass valves close.
- D. Turbine trip, Feedwater pumps trip, AFW pumps start and FRV & bypass valves close.

*Justification*

- A. Incorrect, < P9 no reactor trip
- B. Incorrect, bypass valves do not open
- C. Incorrect, < P9 no reactor trip
- D. Correct

*Drawings show logic required*

P-14 permissive is a steam generator high level override. If two-of-four narrow range level instruments in any steam generator indicate a level of greater than 91.0 percent, the following occurs:

- The main and bypass feedwater regulating valves for all steam generators are shut,
- Both main feed pumps are tripped,
- The main turbine is tripped, and
- Feedwater isolation occurs.

- Technical Reference(s): System Notes SB-2 and SB -3

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Proposed references to be provided to applicants during examination: None

Learning Objective: Main Steam & Feedwater Isolation Valves & MSFIS – SA, obj. C

Question Source: Bank # \_\_\_\_\_

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Modified Bank # \_\_\_\_\_  
New  \_\_\_\_\_

Question History: Last NRC Exam  N/A

Question Cognitive Level:

Memory or Fundamental Knowledge   
Comprehension or Analysis

10 CFR Part 55 Content:

55.41  7   
55.43

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	061 K5.02	
	Importance Rating	3.2	
Knowledge of the operational implications of the following concepts as they apply to the AFW: Decay heat sources and magnitude			

**Question #45**

The Auxiliary Feed System is designed so that a minimum of \_\_\_\_\_ AFW pump(s) can sufficiently remove decay heat and cooldown the RCS at \_\_\_\_\_ °F/hr following a Reactor trip from 100% power.

- A. 1; 50
- B. 2; 50
- C. 1; 100
- D. 2; 100

*UFSAR Section 10.4.9.2.1, Each motor-driven auxiliary feedwater pump will supply 100 percent of the feedwater flow required for removal of decay heat from the reactor. The turbine-driven pump is sized to supply up to twice the capacity of a motor-driven pump. This capacity is sufficient to remove decay heat and to provide adequate feedwater for cooldown of the reactor coolant system at 50°F/hr within 1 hour of a reactor trip from full power.*

*Justification*

- A. Correct.
- B. Incorrect, wrong number of pumps
- C. Incorrect, wrong cooldown rate
- D. Incorrect, wrong number of pumps

Technical Reference(s): FSAR 10.4.9.2

Proposed references to be provided to applicants during examination: None

Learning Objective: Aux Feedwater System, obj. A

Question Source: Bank # \_\_\_\_\_  
Modified Bank # \_\_\_\_\_  
New  \_\_\_\_\_

Question History: Last NRC Exam  N/A

Question Cognitive Level:  
Memory or Fundamental Knowledge   
Comprehension or Analysis

10 CFR Part 55 Content:  
55.41  5  
55.43

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	062 A3.01	
	Importance Rating	3.0	
Ability to monitor automatic operation of the ac distribution system, including: Vital ac bus amperage			

**Question #46**

Given the following plant conditions:

- Loss of Coolant Accident (LOCA) in progress resulting in Safety Injection on Containment High Pressure
- Safety Injection signal has been RESET
- All systems have responded per design
- The crew is currently in Step 1 of E-1, Loss of Reactor or Secondary Coolant

Which ONE of the following describes the response of LSELS (Load Shed Emergency Load Sequencer) if the Startup Transformer is DE-ENERGIZED?

- A. NE01 will START and the Shutdown Sequencer will actuate 'A' train components
- B. NE01 will continue to run unloaded, NE02 will energize NB02, and the shutdown Sequencer will actuate 'B' train components.
- C. NE02 will continue to run unloaded, NE01 will energize NB01, and the shutdown Sequencer will actuate 'A' train components.
- D. An SI load shed will occur on NB02. NE02 will START and the Shutdown Sequencer will actuate 'B' train components.

*Justification*

- A. *Incorrect, Startup transformer feed NB02, no effect on "A" train components*
- B. *Correct. Loss of power to NB02*
- C. *Incorrect, NE02 will pick up load to supply NB02*
- D. *Incorrect, NE02 already running from the SI previously received.*

Basically looking for which bus will be supplied by which component as read by the load on the bus and the Diesel.

Technical Reference(s): E-21001

Proposed references to be provided to applicants during examination: None

Learning Objective: System Description LSELS – NF, obj A, E

Question Source: Bank # \_\_R12266\_\_  
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   7    
55.43 \_\_\_\_\_

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO	
	Tier #	2		
	Group #	1		
	K/A #	062 A1.03		
	Importance Rating	2.5		
Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ac distribution system controls including: Effect on instrumentation and controls of switching power supplies				

**Question #47**

Given the following plant conditions:

The core has been off-loaded to the Spent Fuel Pool.  
Steam Generators A and C are drained for sludge lancing.

Maintenance activities on NN14 Inverter are complete. The Shift Manager has authorized NN04 to be de-energized in order to shift back to the inverter (NN14) from the SOLA transformer (XNN06).

At about the same time, the Control Room Supervisor authorizes I&C to calibrate CST to AFP Suction Transmitter, AL PT-38.

Which ONE of the following describes the consequences of performing these activities simultaneously?

- A. With the core off-loaded Technical Specifications for BOP ESFAS do not apply, therefore there will be no consequences.
- B. Deenergizing instrument bus NN04 results in reducing the protective instrumentation to a 2 out of 3 trip logic.
- C. Deenergizing NN04 with a low suction pressure signal from AL PT-38 will result in an AFW suction swapover signal.
- D. With the core off-loaded, ESFAS is placed in bypass and actuations will not occur, there will be no consequences.

**JUSTIFICATION:**

- A. *Incorrect. Some Tech Specs still apply*
- B. *Incorrect. This will make up the 2 of 3 logic.*
- C. *Correct. With NN04 down and the lowering of AL-PT-38 will cause an actuation signal*
- D. *Incorrect. ESFAS is not placed in Bypass as to affect this signal.*

Technical Reference(s): System Notes AL-1 & NB/NG/NK/NN-1

Proposed references to be provided to applicants during examination: None

Learning Objective: System Description –AL, Obj. F

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Question Source: Bank # \_R12004\_  
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\_  
55.43 \_\_\_\_\_

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO	
	Tier #	2		
	Group #	1		
	K/A #	063 K1.03		
	Importance Rating	2.9		
Knowledge of the physical connections and/or cause-effect relationships between the DC electrical system and the following systems: Battery charger and battery				

**Question #48**

Given the following plant conditions:

- The Callaway Plant is operating at 100% power.
- The 125V DC Power System is normally aligned.
- Offsite power is lost.
- "A" diesel generator starts and loads.
- "B" diesel generator did **NOT** start.
- **NO** operator action has yet been taken.

Which ONE of the following statements describes the effect of this failure on the 125V DC system?

- A. **NO** vital 125V DC buses are energized from a battery charger powered from an operating diesel. **ALL** Vital buses are energized by their battery.
- B. Vital 125V DC buses NK02 and NK04 are energized from a battery charger powered from an operating diesel. Vital buses NK01 and NK03 are energized by their battery.
- C. Vital 125V DC buses NK01 and NK03 are energized from a battery charger powered from an operating diesel. Vital buses NK02 and NK04 are energized by their battery.
- D. All vital 125V DC buses are energized from a battery charger powered from an operating diesel.

*Justification*

- A. *Incorrect. EDG "A" will supply power to the battery chargers*
- B. *Incorrect. EDG "A" will supply NG01 and NG03, not NG02 and NG04*
- C. *Correct. EDG "A" will supply NG01 and NG03*
- D. *Incorrect. EDG "A" will supply NG01 and NG03, not NG02 and NG04*

*Non-Safety Battery Chargers are shed on the initial loss and must be manually restarted*

Technical Reference(s): E-21NG01 & E-21NG02

Proposed references to be provided to applicants during examination: None

Learning Objective: System Description -Safeguards Power, Obj.A

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 \_\_\_7\_\_\_

55.43 \_\_\_\_\_

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	064 K1.05	
	Importance Rating	3.4	
Knowledge of the physical connections and/or cause-effect relationships between the ED/G system and the following systems: Starting air system			

**Question #49**

The Diesel Generators are designed to start and be ready to close in on the respective bus within twelve (12) seconds. To assist or ensure this capability exists (select all that apply):

1. The lube oil is circulated and heated to keep the engine warm.
2. There are two (2) separate air starting systems.
3. The D/G room temperature is kept below 85°F.
4. The fuel oil day tank keeps the fuel warm to promote rapid combustion when injected.
5. The jacket cooling water is heated and circulated to keep the engine warm.

- A. 2, 3, 4
- B. 1, 4, 5
- C. 2, 4, 5
- D. 1, 2, 5

*Justification*

- A. Incorrect, no fuel oil pre-heat
- B. Incorrect, no fuel oil pre-heat
- C. Incorrect, no fuel oil pre-heat
- D. Correct.

Technical Reference(s): OTN-NE-0001A

Proposed references to be provided to applicants during examination: None

Learning Objective: Standby Generation – KJ/NE, Obj. C

Question Source: Bank # \_\_\_0110030C03A\_\_\_  
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\_\_\_  
55.43 \_\_\_\_\_

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	073 A2.02		
	Importance Rating	2.7		
Ability to (a) predict the impacts of the following malfunctions or operations on the PRM system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Detector failure				

**Question #50**

Given the following plant conditions:

- I&C is performing a functional test on Fuel Building Radiation Detector GG RE-27.
- Due to failure to self-check, the technician causes GG-RE-27 gas channel to exceed the HiHi alarm setpoint without having the key switch on the ESFAS panel in BYPASS.

Which ONE of the following describes the resulting plant configuration?

- A. Only Fuel Building Supply Air Unit "A" starts.
- B. Both Emergency Exhaust Fans start.
- C. Only Emergency Exhaust Fan "A" starts.
- D. Both Fuel Building Supply Air Units start.

*Justification:*

- A. *Incorrect. Both Emer Exh Fans Start, not Supply fans*
- B. *Correct. Both Emer Exh Fans Start*
- C. *Incorrect. Both Emer Exh Fans Start, not just one*
- D. *Incorrect. Both Emer Exh Fans Start, not Supplyfans*

Technical Reference(s): OTA-SP-RM011

Proposed references to be provided to applicants during examination: None

Learning Objective: Ventilation Systems –Primary – GG/GK/GL, Obj. C & D

Question Source: Bank # \_R12336\_\_\_\_\_  
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 \_\_5\_\_



NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

55.43 \_\_\_\_\_

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	076 A2.01	
	Importance Rating	3.5*	
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: Loss of SWS			

**Question #51**

Given the following plant conditions:

- The Callaway Plant is at 100% power.
- Annunciator 12A, Service Water Pump Lockout, alarms.
- Investigation reveals that all Service Water pumps have tripped.

Which ONE of the following describes the required crew response?

- A. Start both ESW Trains, with one train supplying Service Water. Trip the Turbine if any trip setpoint is reached.
- B. Start both ESW Trains, with one train supplying Service Water. Trip the Reactor if any Turbine trip setpoint is reached.
- C. Place both ESW Trains in manual operation. Trip the Reactor if any Turbine trip setpoint is reached.
- D. Place both ESW Trains in manual operation. Trip the Turbine if any trip setpoint is reached.

*JUSTIFICATION:*

- A. *Incorrect, Trip the reactor, not the turbine*
- B. *Incorrect; Not supplying service water*
- C. *Correct.*
- D. *Incorrect; Trip the reactor, not the turbine*

Technical Reference(s): OTA-RK-00014, Addendum 12A

Proposed references to be provided to applicants during examination: None

Learning Objective: System Lesson- DA, Obj. F

Question Source: Bank #   R12306    
Modified Bank #             
New           

Question History: Last NRC Exam   N/A  

Question Cognitive Level:

Memory or Fundamental Knowledge         
Comprehension or Analysis   X

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

10 CFR Part 55 Content:

55.41 \_5, 10\_  
55.43 \_\_\_\_\_

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

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

**Question #52**

Given the following plant conditions:

- An Instrument Air line break has occurred at the Condensate Polishers
- KA-PI-40, Instrument Air Header Pressure indicator, is reading 102 psig and dropping

Which ONE of the following describes the sequence of events that occurs due to this failure?

- A. The First Backup air compressor loads at 119 psig; and all compressors will be running at 110 psig.
- B. Service Air header isolation valve KA-PV-11 will close at 117 psig; the Second Backup air compressor loads at 115 psig.
- C. The First Backup air compressor loads at 117 psig; the Service Air Header Isolation valve KA-PV-11 closes at 115 psig.
- D. The First Backup air compressor loads at 117 psig; and all air compressors should be running at 115 psig.

*Justification:*

- A. *Incorrect, Loads at 117.*
- B. *Incorrect, Valve closes at 110.*
- C. *Incorrect, Valve closes at 110.*
- D. *Correct*

Technical Reference(s): OTO-KA-00001

Proposed references to be provided to applicants during examination: None

Learning Objective: System Lesson –KA, Obj. D

Question Source: Bank # \_\_\_\_\_  
Modified Bank # **0110140D02A** \_\_\_\_\_  
New \_\_\_\_\_

Question History: Last NRC Exam \_\_\_ N/A \_\_\_\_\_

Question Cognitive Level:

Memory or Fundamental Knowledge \_\_\_\_\_  
Comprehension or Analysis   X

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

10 CFR Part 55 Content:

55.41 \_\_7\_\_

55.43 \_\_\_\_\_

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

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

**Question #53**

Given the following plant conditions:

- The operating crew has responded to a loss of coolant accident.
- A Safety Injection (SIS) and Containment Isolation - Phase A (CISA) have actuated.
- It is now required to Purge Hydrogen from the containment and dilution air is required.

Which ONE of the following states how air will be supplied to containment?

- A. Reset CISA, then OPEN Instrument Air Supply Containment Isolation, KA FV-29, and Instrument Air Supply to H2 Control System KA HV-30.
- B. Reset CISA, then OPEN Service Air Containment Isolation, KA V-118, and Instrument Air Supply Containment Isolation, KA FV-29.
- C. OPEN Instrument Air Supply Containment Isolation, KA FV-29, and Instrument Air Supply to H2 Control System KA HV-30.
- D. OPEN Service Air Containment Isolation, KA V-118, and Instrument Air Supply to H2 Control System KA HV-30.

*Justification*

- A. Correct. Correct CISA must be reset to get KA FV-29 open to supply KA HV-30
- B. Incorrect. Still need KA HV-30 to be opened
- C. Incorrect. Need to reset CISA to get KA FV-29 open
- D. Incorrect. Need CISA and KA FV-29 open.

Technical Reference(s): OTN-GS-00001

Proposed 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:

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Memory or Fundamental Knowledge \_\_\_\_\_  
Comprehension or Analysis   X  

10 CFR Part 55 Content:

55.41   7    
55.43 \_\_\_\_\_

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	103 2.4.14		
	Importance Rating	3.8		
Knowledge of general guidelines for EOP usage.				

**Question #54**

Given the following plant conditions:

- The crew is responding to a large break LOCA.
- They begin a transfer to ECCS cold leg recirculation due to low RWST level.
- Current plant conditions:
  - SI Signal Reset.
  - Containment Pressure 49 psig stable.
  - Containment Recirc Sump Level 125" rising.
  - Containment Spray Pumps Off.

Which ONE of the following describes the correct crew action?

- A. Complete ES-1.3 through step 4, then transition to FR-Z.2, Response to Containment Flooding.
- B. Complete ES-1.3 through step 4, then transition to FR-Z.1, Response to High Containment Pressure.
- C. Complete ES-1.3, then transition to FR-Z.2, Response to Containment Flooding.
- D. Complete ES-1.3, then transition to FR-Z.1, Response to High Containment Pressure.

*Justification*

- A. *Incorrect. Level is not at Entry conditions for Z.2*
- B. *Correct.*
- C. *Incorrect. Level is not at Entry conditions for Z.2*
- D. *Incorrect. Transition to Z.1 is correct after step 4, do not wait until 1.3 is completed.*

Technical Reference(s): CSF-1, ES-1.3, note prior to step 1

Proposed references to be provided to applicants during examination: None

Learning Objective: Procedure ES-1.3 Lesson, obj. G

Question Source: Bank # **R11792** \_\_\_\_\_  
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   10    
55.43 \_\_\_\_\_

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	103 K4.06	
	Importance Rating	3.1	
Knowledge of containment system design feature(s) and/or interlock(s) which provide for the following: Containment Isolation System			

**Question #55**

A main steam line break inside containment has occurred causing the containment pressure to rise to 20 psig.

Which ONE of the following containment isolation systems will actuate to mitigate the pressure increase?

- A. SIS
- B. CISA
- C. CISB
- D. SLIS

*JUSTIFICATION:*

- A. *Incorrect; Does not stop the pressure increase*
- B. *Incorrect; Isolates Containment but does not stop pressure rise*
- C. *Incorrect; CSAS does, but not CISB*
- D. *Correct;*

Technical Reference(s): Tech Spec Bases - B 3.3.2

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # 003B460C08A  
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    
55.43     

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	002 A1.11	
	Importance Rating	2.7	

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RCS controls including: Relative level indications in the RWST, the refueling cavity, the PZR and the reactor vessel during preparation for refueling

**Question #56**

Given the following plant conditions:

- The Callaway Plant is in mode 5 and shut down for 350 hours.
- The RCS is being drained down to a Mid-Loop condition.
- RHR is in service maintaining RCS temperature at approximately 130°F.
- PZR Level indicates 53%.

If level is lowered by 4.5 feet, using OOA-BB-00003 (attached), what will the Tygon Hose level indicate?

- A. 2053.06
- B. 2056.27
- C. 2057.56
- D. 2062.06

*Justification:*

- A. Correct.
- B. Incorrect. Uses 50% level without interpolating or using Note.
- C. Incorrect. Interpolates but does not use Note.
- D. Incorrect. Applies Note incorrectly (adding instead of subtracting).

*Provide students with the drawing. Students have to use the formula in Note 4 on the drawing to calculate the 53% level then subtract 4.5 from that value.*

Technical Reference(s): OOA-BB-00003

Proposed references to be provided to applicants during examination: **OOA-BB-00003**

Learning Objective:

Question Source: Bank # \_\_\_\_\_  
Modified Bank # \_\_\_\_\_  
New  X \_\_\_\_\_

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   5  

55.43       

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO	
	Tier #	2		
	Group #	2		
	K/A #	015 K1.08		
	Importance Rating	2.6*		
Knowledge of the physical connections and/or cause-effect relationships between the NIS and the following systems: RCS (pump start)				

**Question #57**

Given the following plant conditions:

- "A" reactor coolant pump (RCP) is circulating reactor coolant at 100°F.
- After several hours the reactor coolant temperature has risen to 150°F.

Assuming coolant flow rate (gpm) is constant, RCP motor amps, as read on BB II-1 thru 4, will have \_\_\_\_\_ and NI response, as read on SE NI-41B thru 44B, will \_\_\_\_\_.

- A. lowered; rise due to a lower density
- B. lowered; lower due to a rise in head loss
- C. risen; lower due to rise in density
- D. risen; rise due to a lower density

*Justification*

- A. Correct. Amps will decrease due to lower densities, NI's will show an increase due to more leakage from the core.
- B. Incorrect. See "A"
- C. Incorrect. See "A"
- D. Incorrect. See "A"

Technical Reference(s): GFE Lesson – Reactor Theory/ Reactivity Coefficients

Proposed 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 \_\_\_\_\_  
Comprehension or Analysis \_\_\_X\_\_\_

10 CFR Part 55 Content: 55.41 \_\_\_5\_\_\_

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

55.43 \_\_\_\_\_

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO	
	Tier #	2		
	Group #	2		
	K/A #	027 A2.01		
	Importance Rating	3.0*		
Ability to (a) predict the impacts of the following malfunctions or operations on the CIRS; and (b) based on those predictions, use Procedures to correct, control, or mitigate the consequences of those malfunctions or operations: High temperature in the filter system				

**Question #58**

Given the following plant conditions:

- The plant is at 100% power.
- The temperature indicated on CTMT PURGE FLTR ADS UNIT HI TEMP SW, GT TSH-0019, reads 209°F.

Which ONE of the following describes the Containment Purge Filter Absorber Unit response and procedural requirements?

**Filter Absorber**

**Procedural Requirements**

- |  |   |
|--|---|
| A. Operating Fan Will Stop             | Neither exhaust fan will be able to start and the opened fan filter damper will need to be verified closed unless an SI occurs. |
| B. Operating Fan Will Stop             | Neither exhaust fan will be able to start and the opened fan filter damper will close.  |
| C. Operating Fan Will Continue Running | Neither exhaust fan will stop until the high temperature signal is deactivated.   |
| D. Operating Fan Will Continue Running | Neither exhaust fan will stop and the opened fan filter damper will need to be verified closed.                                 |

*Justification*

- A. Incorrect, because an SI has no effect on the fan or damper.  
 B. Correct.  
 C. Incorrect, Fan will stop.  
 D. Incorrect, Fan will stop.

Technical Reference(s): E-23GT05

Proposed references to be provided to applicants during examination: None

Learning Objective: Containment Purge System – GN/GS/GT

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 \_\_5\_\_

55.43 \_\_\_\_\_

Comments:



NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

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

**Question #59**

Given the following plant conditions:

- The Callaway Plant is at 100% power
- "A" EDG is out of service for maintenance
- A Large break design basis Loss Of Coolant Accident has occurred
- Offsite Power is still available

Which ONE of the following describes the effect on Containment if "B" train Hydrogen Recombiner is lost after SI initiates?

Containment hydrogen concentration will . . . .

- A. not go above 4%.
- B. rise to > 8%.
- C. rise and stabilize between 4 and 8%.
- D. rise to > 8% and then lower to < 4% by containment cooler operation.

*Justification:*

- A. Correct. A single recombiner is designed to maintain H2 < 4% during a design basis LOCA.
- B. Incorrect. See A above
- C. Incorrect. See A above
- D. Incorrect. See A above

Technical Reference(s): Tech Spec Bases B 3.6.8

Proposed 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 \_\_7\_\_

55.43 \_\_\_\_\_

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO	
	Tier #	2		
	Group #	2		
	K/A #	029 A3.01		
	Importance Rating	3.8		
Ability to monitor automatic operation of the Containment Purge System including: CPS isolation				

**Question #60**

Given the following plant conditions:

- The Callaway Plant was at 100% power
- A Containment Purge was in progress
- A reactor trip was initiated due to a leak in Containment
- Containment Pressure is now 3.8 psig

Which ONE of the following describes the effect on the Containment purge supply and exhaust fans and the DIRECT actuating signal?

<u>Supply/Exhaust Fans</u>	<u>Actuating Signal</u>
A. Continue Running	Safety Injection Signal
B. Trip	Safety Injection Signal
C. Continue Running	Containment Isolation Phase A
D. Trip	Containment Isolation Phase A

*Justification*

*The Safety injection actuates the CISA which actuates CPIS. Applicant has to know the SI setpoint is 3.5 psig.*

- A. *Incorrect, see above*
- B. *Incorrect, see above*
- C. *Incorrect, see above*
- D. *Correct, see above*

Technical Reference(s): 7250D64 SH8

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # \_\_\_\_\_  
Modified Bank # \_\_\_\_\_  
New  \_\_\_\_\_

Question History: Last NRC Exam  N/A \_\_\_\_\_

Question Cognitive Level:

Memory or Fundamental Knowledge  \_\_\_\_\_  
Comprehension or Analysis \_\_\_\_\_

NRC Site-Specific Written Examination  
Callaway Plant  
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10 CFR Part 55 Content:

55.41 \_\_7\_\_

55.43 \_\_\_\_\_

Comments:

NRC Site-Specific Written Examination  
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Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	033 2.4.11	
	Importance Rating	4.0	
Knowledge of abnormal condition procedures. Spent Fuel Pool Cooling System			

**Question #61**

Given the following plant conditions:

- Core load is in progress.
- ANN 76D, SFP LEV HI/LO is lit.
- Cavity level is currently at el. 2044.7' and dropping slowly due to a seal failure.

Which ONE of the following will be the FIRST action taken by the Control Room in accordance with OTO-EC-00001, Loss of SFP/Refuel Pool Level?

- A. Manually actuate a Fuel Building Isolation Signal (FBIS).
- B. Initiate emergency makeup from Essential Service Water (ESW)
- C. Close the Fuel Building Roll-up Door.
- D. Close EC-V995, Fuel Transfer Tube Isolation Valve.

*JUSTIFICATION:*

- A. *Incorrect. FBIS not Actuated until later and only if required.*
- B. *Incorrect. Later in the procedure and only in emergency*
- C. *Incorrect. Later in procedure*
- D. *Correct. To prevent draining of the Refuel Pool*

Technical Reference(s): OTO-EC-00001

Proposed 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 \_\_\_\_\_  
Comprehension or Analysis \_\_\_X\_\_\_

10 CFR Part 55 Content:

55.41 \_\_\_10\_\_\_  
55.43 \_\_\_\_\_

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO	
	Tier #	2		
	Group #	2		
	K/A #	035 A4.02		
	Importance Rating	2.7		
Ability to manually operate and/or monitor in the control room: Fill of dry S/G.				

**Question #62**

Given the following plant conditions:

- One safety valve on "A" SG failed open with the plant at 100% power.
- The reactor was tripped and "A" SG isolated per E-2, Faulted Steam Generator Isolation.
- The failed safety valve has been gagged shut and SI has been terminated.
- Containment Pressure is 0.7 psig.
- AFW issues have resulted in the following S/G levels.
- "A" SG level is 0% NR, 7% WR.
- "B" SG level is 0% NR, 3% WR.
- "C" SG level is 0% NR, 3% WR.
- "D" SG level is 0% NR, 3% WR.
- The crew has initiated feed flow to restore "A" S/G level based on Engineering staff recommendations.

Which ONE of the following describes the indications for a "dry" S/G and when unlimited AFW can be used?

	<u>Dry S/G</u>	<u>Maximum Flow Allowed</u>	
A.	< WR 10%	WR > 10%	
B.	< NR 10%	NR > 25%	
C.	< WR 25%	WR > 25%	
D.	< NR 25%	NR > 25%	

*Justification*

- A. Correct.
- B. Incorrect, NR not used.
- C. Incorrect, Not in adverse conditions
- D. Incorrect, NR not used, not in adverse conditions

*FR-H.5 Bkgd, In FR-H.1, a rapid restoration of feedwater may be necessary for the reestablishment of an adequate secondary heat sink. A rapid restoration of AFW flow is not necessary in FR-H.5 to establish level indication. Unless directed by the Plant Engineering Staff, it is prohibited to feed a dry steam generator. A dry steam generator is defined as a steam generator with a water level below the wide range level indication.*

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*Following an evaluation by the Plant Engineering Staff as part of the long term recovery actions, the affected steam generator may be refilled. This evaluation should consider steam generator materials and properties, Technical Specification considerations, etc.*

Technical Reference(s): FR-H.1 FOP

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # \_\_\_\_\_  
Modified Bank # \_\_\_\_\_  
New  \_\_\_\_\_

Question History: Last NRC Exam  N/A \_\_\_\_\_

Question Cognitive Level:  
Memory or Fundamental Knowledge \_\_\_\_\_  
Comprehension or Analysis  \_\_\_\_\_

10 CFR Part 55 Content:  
55.41  7 \_\_\_\_\_  
55.43 \_\_\_\_\_

Comments:



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Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	068 K5.03	
	Importance Rating	2.6	
Knowledge of the operational implication of the following concepts as they apply to the Liquid Radwaste System: Units of radiation, dose, and dose rate			

**Question #63**

An Operations Technician spent 30 minutes in a field of 150 mr/hour lining up to transfer the contents of one discharge monitor tank to another. He said later that if he had 'preplanned' his work he could have been finished in 20 minutes.

Which ONE of the following describes how much dose could have been avoided if he had preplanned the job?

- A. 12.5 mrem
- B. 25 mrem
- C. 50 mrem
- D. 75 mrem

*Justification*

- A. Incorrect, 1/2 of saved dose
- B. Correct
- C. Incorrect, 20 min dose
- D. Incorrect, 30 min dose

30 mins = 75 mr, 20 mins = 50 mr

Technical Reference(s): ALARA

Proposed 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 \_\_\_\_\_  
Comprehension or Analysis \_\_\_X\_\_\_

10 CFR Part 55 Content:

55.41 \_12\_  
55.43 \_\_\_\_\_

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Comments:

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Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	075 K2.03	
	Importance Rating	2.6*	
Knowledge of bus power supplies to the following: Emergency/essential SWS pumps			

**Question #64**

Given the following plant conditions:

- No equipment is out-of-service and the "B" train is protected.
- A loss of offsite power and reactor trip have occurred.
- "A" EDG is powering bus NB01 and is loaded.
- "B" EDG is powering bus NB02 but the Sequencer failed at Step 4 during sequencing of loads onto bus NB02.
- All other systems have functioned normally.

In order to complete the load sequencing on the proper order, what will be the next load that the Reactor Operator must start?

- A. Component Cooling Water pump.
- B. MDAFW pump.
- C. Essential Service water pump.
- D. Containment cooler fans.

*Justification*

- A. *Incorrect, next pump to start*
- B. *Incorrect, started after Containment Cooler Fans*
- C. *Correct.*
- D. *Incorrect, started after CCW Pump*

Technical Reference(s): E-22NF01

Proposed references to be provided to applicants during examination: None

Learning Objective: Systems Lesson LSELS –NF, Obj 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:

NRC Site-Specific Written Examination  
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55.41 \_\_7\_\_  
55.43 \_\_\_\_\_

Comments:

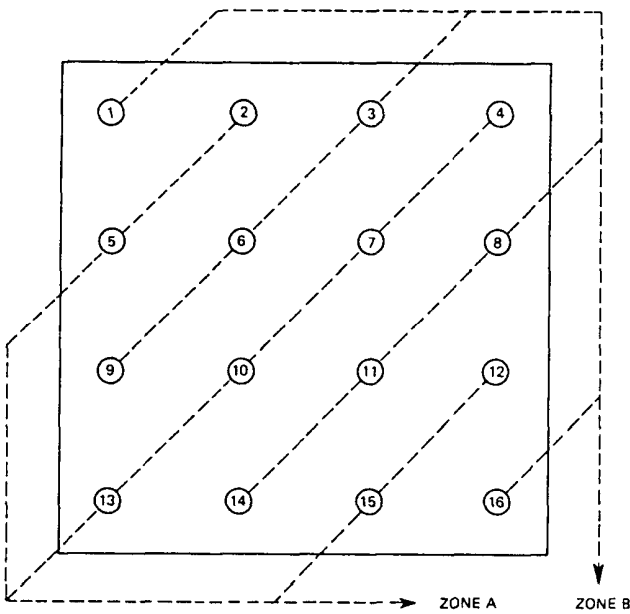
NRC Site-Specific Written Examination  
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Examination Outline Cross-reference:	Level	RO	SRO	
	Tier #	2		
	Group #	2		
	K/A #	086 K4.03		
	Importance Rating	3.1		
Knowledge of design feature(s) and/or interlock(s) which provide for the following: Detection and location of fires				

**Question #65**

Given the following plant conditions:

- The plant is in Mode 4 going to Mode 6
- A plant cooldown is in progress
- Grinding work is in progress in the Electrical Penetration Room A



Which ONE of the following describes the required signals to actuate the Halon 1301 system?

- A. Detector 9 and 3 in alarm
- B. Detector 3 in alarm, detector 9 has a trouble alarm
- C. Detector 3 and 13 in alarm
- D. Detector 9 in alarm, detector 1 has a trouble alarm

*Justification*

- A. Incorrect, both are in the same zone
- B. Incorrect, both are in the same zone

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- C. Correct. See below
- D. Incorrect, both are in the same zone

*In order for the Halon 1301 system to automatically actuate, detectors in both loops must sense a fire or a detector in one loop senses a fire while a trouble signal is present on the other loop. Detection of a fire by one loop without a detection or trouble signal in the other loop will give an alarm only*

Technical Reference(s): T61.0110 6 RO Systems, LP 35

Proposed references to be provided to applicants during examination: None

Learning Objective: LP 35 RO/SRO Objective B3

Question Source: Bank # \_\_\_\_\_  
Modified Bank # \_\_\_\_\_  
New  \_\_\_\_\_

Question History: Last NRC Exam  N/A \_\_\_\_\_

Question Cognitive Level:  
Memory or Fundamental Knowledge \_\_\_\_\_  
Comprehension or Analysis  \_\_\_\_\_

10 CFR Part 55 Content:  
55.41  7 \_\_\_\_\_  
55.43 \_\_\_\_\_

Comments:  
We don't expect operators to memorize which detectors are in which zone. They are required to know the logic required for actuation.

NRC Site-Specific Written Examination  
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Examination Outline Cross-reference:	Level	RO	SRO	
	Tier #	3		
	Group #			
	K/A #	2.1.25		
	Importance Rating	3.9		
Ability to interpret reference materials, such as graphs, curves, tables, etc.				

**Question #66**

Given the following plant conditions:

- All reactor coolant pumps are secured.
- RCS WR Pressure (BB PI-405) 400 psig.
- RCS WR Pressure (BB PI-406) 350 psig.
- Charging Header Pressure (BG PI-120A) 575 psig.
- VCT Pressure (BG PI-115) 50 psig.

What is the MAXIMUM #1 seal leak-off flow rate that would allow a reactor coolant pump to be started, using the attached figure?

- A. 1.0 gpm
- B. 1.5 gpm
- C. 2.0 gpm
- D. 2.5 gpm

*Justification*

- A. Incorrect. 200# D/p
- B. Incorrect. 300# D/P
- C. Correct. 500# D/P = 2.0 gpm
- D. Incorrect. 650# D/P

Technical Reference(s): OTN-BB-00003, Attachment 4

Proposed references to be provided to applicants during examination: OTN-BB-00003, Attachment 4

Learning Objective:

Question Source: Bank # **003A20C104A** \_\_\_\_\_  
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\_\_

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Comments:



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Examination Outline Cross-reference:	Level	RO	SRO	
	Tier #	3		
	Group #			
	K/A #	2.1.29		
	Importance Rating	4.1		
Knowledge of how to conduct system lineups, such as valves, breakers, switches, etc.				

**Question #67**

During an independent verification a valve is found out of position. Which ONE of the following describes how the verifier is to handle the component out of position in accordance with APA-ZZ-00100, Written Instructions Use and Adherence?

- A. Do NOT change valve position. Notify the Shift Manager of the discrepancy.
- B. Do NOT change valve position. Notify the initial valve positioner of the discrepancy.
- C. Correct the valve position. Have Shift Manager obtain new verifier for independent verification for that valve only.
- D. Place the component in a safe position. Have the initial valve positioner perform the independent verification for that valve only.

*Justification*

- A. Correct.
- B. Incorrect, notify SM
- C. Incorrect, do not reposition component
- D. Incorrect, do not reposition component, notify SM

Technical Reference(s): APA-ZZ-00100, step 4.4.1

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # \_\_\_\_\_  
Modified Bank # \_\_\_\_\_  
New  \_\_\_\_\_

Question History: Last NRC Exam  N/A \_\_\_\_\_

Question Cognitive Level:

Memory or Fundamental Knowledge  \_\_\_\_\_  
Comprehension or Analysis \_\_\_\_\_

10 CFR Part 55 Content:

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55.43 \_\_\_\_\_

Comments:

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Examination Outline Cross-reference:	Level	RO	SRO	
	Tier #	3		
	Group #			
	K/A #	2.1.32		
	Importance Rating	3.8		
Ability to explain and apply system limits and precautions.				

**Question #68**

Given the following plant conditions:

- The Callaway Plant is at 100% power.
- Reactor Engineering has requested Turbine and Reactor power be reduced to 70% for a special test procedure to be performed.
- As a result power is currently 73% and lowering in accordance with OTG-ZZ-00004, Power Operation.

Which ONE of the following describes the turbine backpressure limit?

- A. 4.0 in Hga
- B. 5.0 in Hga
- C. 6.5 in Hga
- D. No limit currently in effect

*Justification*

- A. Incorrect. See table below
- B. Correct
- C. Incorrect. See table below
- D. Incorrect. See table below

**TURBINE LOAD**

< 30%  
> 30% to < 75%  
> 75%

**BACK PRESSURE**

< 4.0 in Hga  
< 5.0 in Hga  
< 6.5 in Hga

Technical Reference(s): OTG-ZZ-00004, step 3.4.1

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # \_003A10D101B\_\_\_\_\_  
Modified Bank # \_\_\_\_\_  
New \_\_\_\_\_

Question History: Last NRC Exam \_\_\_\_ N/A \_\_\_\_\_

Question Cognitive Level:

Memory or Fundamental Knowledge \_\_\_\_\_

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Comprehension or Analysis

X

10 CFR Part 55 Content:

55.41  10

55.43

Comments:

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Examination Outline Cross-reference:	Level	RO	SRO	
	Tier #	3		
	Group #			
	K/A #	2.2.14		
	Importance Rating	3.9		
Knowledge of the process for controlling equipment configuration or status.				

**Question #69**

You as the RO have directed an OT to verify a valve lineup per the applicable OTN and flow diagram. The OT reports later that an existing valve was listed in the OTN but was not on the drawing.

Which ONE of the following describes the required actions for this plant configuration situation?

Notify the CRS and initiate . . .

- A. a Work Request to update the flow diagram.
- B. a Request For Resolution (RFR) to update the flow diagram.
- C. a Callaway Action Request (CAR) to update the flow diagram.
- D. an Operator Workaround and annotate on the OTN that the valve is not shown on the flow diagram.

*Justification*

- A. *Incorrect, Work request process not the correct process.*
- B. *Incorrect, RFRs used to seek engineering questions and design changes.*
- C. *Correct*
- D. *Incorrect, Workaround plausible if OTN does not work. OTN is correct. Flow diagram is missing valve and needs revision.*

Technical Reference(s): APA-ZZ-00500

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # INPO  
Modified Bank # \_\_\_\_\_  
New \_\_\_\_\_

Question History: Last NRC Exam Robinson 04

Question Cognitive Level:

Memory or Fundamental Knowledge \_\_\_\_\_  
Comprehension or Analysis X

10 CFR Part 55 Content:

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Comments:

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Examination Outline Cross-reference:	Level	RO	SRO	
	Tier #	3		
	Group #			
	K/A #	2.2.36		
	Importance Rating	3.1		
Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.				

**Question #70**

Given the following plant conditions:

- Callaway Plant is operating at 75% power.
- Transformer checks were being conducted in the switchyard.
- A grid disturbance caused NB02 bus voltage fluctuations.
- MCB Annunciator "NB02 Bus Degraded Voltage" had been coming in intermittently, but has now been lit continuously for 60 seconds.

Which ONE of the following describes 1) the conditions that will trip the normal supply breaker NB0209; and 2) which TS LCO applies?

- A. 1) A Containment Spray Actuation Signal is actuated.  
2) TS LCO 3.8.1, AC Sources - Operating
- B. 1) NB02 voltage drops to 3800 volts.  
2) TS LCO 3.8.9, Distribution Systems - Operating
- C. 1) A Safety Injection Signal is actuated.  
2) TS LCO 3.8.1, AC Sources - Operating
- D. 1) The annunciator remains lit 25 seconds longer.  
2) TS LCO 3.8.9, Distribution Systems - Operating

*Justification*

- A. *Incorrect, SI signal, correct LCO*
- B. *Incorrect, <math>\leq 3761</math>, no indications that the EDG is inop*
- C. *Correct.*
- D. *Incorrect, need to be 87 to 104, no indications that the EDG is inop*

• *A time delay of 111 + 8 second allows time for the Control Room Operator or Grid Operations to correct the undervoltage condition before NB feeder breakers trip. The degraded voltage relay bistable also incorporates a time delay of 8 second for a total of 119 + 8.5 second.*

• *Alarm comes in after 22 + 1.0 seconds of Degraded Voltage Condition. This allows for the starting of a RCP motor without receiving an undervoltage trip. Load shed occurs after 119 seconds of degraded voltage condition and 97 seconds after alarm of this annunciator. However, if a Safety Injection Signal is present, load shed will occur after 8 seconds of degraded voltage.*

*When 87 to 104 seconds has elapsed, the following will occur:*

- *NB HIS-4, NB02 NORM SPLY BKR NB0209, opens*
- *NB HIS-5, NB02 ALT SPLY BKR NB0212, opens*

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- *IF no lockout exists, NE02, GEN STANDBY #2, starts and energizes NB02, SWGR 4.16 KV BUS.*
- *Steam Generator Blowdown Isolation Signal*
- *Turbine Driven Auxiliary Feedwater Actuation Signal*

*WHEN 87 to 104 seconds has elapsed, On RL015, CHECK the following breakers OPEN:*

- *NB HIS-4, NB02 NORM SPLY BKR NB0209*
- *NB HIS-5, NB02 ALT SPLY BKR NB0212*

Technical Reference(s): T61.0110 6, RO Systems, Lesson Plan #51  
OTA-RK-00016 (Add 22E)

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # \_R12215\_  
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\_  
55.43 \_\_\_\_\_

Comments:

NRC Site-Specific Written Examination  
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Examination Outline Cross-reference:	Level	RO	SRO	
	Tier #	3		
	Group #			
	K/A #	2.3.4		
	Importance Rating	3.2		
Knowledge of radiation exposure limits under normal or emergency conditions.				

**Question #71**

Given the following plant conditions:

- The plant is in MODE 6 with core off load in progress.
- The refueling machine gripper is to be replaced by a diver.
- While performing the gripper replacement, the diver left the approved diving area and went within 4.5 feet of some spent fuel assemblies for 10 minutes.
- Whole body dose received was 270 mrem.

Which ONE of the following is the correct calculation of whole body exposure the diver can receive without exceeding administrative limits and yet complete the task?

- A. 730 mrem
- B. 1730 mrem
- C. 2270 mrem
- D. 3730 mrem

*Justification*

- A. Incorrect, Uses incorrect admin limit of 1000.
- B. Correct.  $2000-270=1730$  see below. 2000 is limit at Callaway –  $270 = 1730$ .
- C. Incorrect, Adds 270 to 2000 instead of subtracting 270.
- D. Incorrect, Uses incorrect admin limit of 4000.

Technical Reference(s): APA-ZZ-01000 (Att. 1)

Proposed 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 \_\_\_\_\_  
Comprehension or Analysis \_\_\_X\_\_\_

10 CFR Part 55 Content:

55.41 \_12\_



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55.43 \_\_\_\_\_

Comments:

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<b>Examination Outline Cross-reference:</b>	<b>Level</b>	<b>RO</b>		<b>SRO</b>
	<b>Tier #</b>	3		
	<b>Group #</b>			
	<b>K/A #</b>	2.3.5		
	<b>Importance Rating</b>	2.9		
Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.				

**Question #72**

During a prejob briefing, Radiation Protection tells you the following:

- Electronic dosimeter (ED) dose alarm setting is 400 mrem.
- Electronic dosimeter (ED) dose rate alarm setting is 1000 mrem/hr.
- Assigned RWP work area dose rate is 1000 mr/hr.

Based on the conditions above, which ONE of the following describes when you would be required to leave the Radiological Control Area (RCA)?

- A. Immediately due to an ED dose alarm.
- B. Immediately due to an ED dose rate alarm.
- C. In 24 minutes due to an ED dose alarm.
- D. In 24 minutes due to an ED dose rate alarm.

*Justification:*

- A. *Incorrect. A dose alarm would be received in 24 minutes. Dose = 400 mrem/1000 mr/hr.*
- B. *Correct. A dose rate alarm would be received immediately since the work area dose rate is 1000 mr/hr; which is equal to the rate alarm setting. You are required by the ALARA program to exit the RCA upon receiving an ED alarm.*
- C. *Incorrect. A dose alarm would be received in 24 minutes. Dose = 400 mrem/1000 mr/hr.*
- D. *Incorrect. A dose rate alarm would be received immediately since the work area dose rate is 1000 mr/hr; which is equal to the rate alarm setting.*

Technical Reference(s): APA-ZZ-01004

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # \_\_\_\_\_  
Modified Bank # \_\_\_\_\_  
New  \_\_\_\_\_

Question History: Last NRC Exam  N/A

Question Cognitive Level:

Memory or Fundamental Knowledge \_\_\_\_\_  
Comprehension or Analysis

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10 CFR Part 55 Content:

55.41 \_10, 12\_\_\_\_  
55.43 \_\_\_\_\_

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO	
	Tier #	3		
	Group #			
	K/A #	2.4.5		
	Importance Rating	3.7		
Knowledge of the organization of the operating procedures network for normal, abnormal, and emergency evolutions.				

**Question #73**

Given the following events and conditions:

- The Callaway Plant was conducting control rod drop tests during a plant startup at 2% reactor power when a complete loss of 'A' Train CCW occurred.
- Control room operators enter OTO-EG-00001, CCW System Malfunction.
- CCW cooling to the Reactor Coolant Pumps is lost for 10 minutes.
- The operators manually trip the reactor but the trip breakers fail to open.
- Reactor power has risen to 5%.
- Pressurizer pressure = 1930 psig.

Which ONE of the following statements correctly describes the proper procedural flow path for these conditions?

- A. Remain in OTO-EG-00001, trip all RCPs and commence a reactor shutdown.
- B. Implement FR-S.1, Response to Nuclear Power Generation/ATWS, concurrently with OTO-EG-00001.
- C. Terminate actions of OTO-EG-00001 and immediately transition to FR-S.1.
- D. Enter E-0 and immediately transition to FR-S.1 while continuing in OTO-EG-00001 as time and conditions permit.

*Justification:*

- A. Incorrect, per reference. Do not trip RCP's during ATWS, common simulator error
- B. Incorrect, per reference. Nothing is implemented with S.1, common to do in E series
- C. Incorrect, per reference. Wrong procedure flowpath
- D. Correct, per reference.

Technical Reference(s): ODP-ZZ-00025

Proposed 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 \_\_\_\_\_  
Comprehension or Analysis   X  

10 CFR Part 55 Content:

55.41   10    
55.43 \_\_\_\_\_

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO	
	Tier #	3		
	Group #			
	K/A #	2.4.11		
	Importance Rating	4.0		
Knowledge of abnormal condition procedures.				

**Question #74**

Which ONE of the following events would require the Control Room to implement OTO-SK-00001, Plant Security Event-Hostile Intrusion?

- A. An intrusion is detected into the Owner Controlled Area
- B. An imminent aircraft threat is received from the NRC
- C. Announcement by Security of a "CODE RED"
- D. A tornado touches down resulting in a loss of off-site power

*Justification*

- A. *Incorrect, not an entry condition, would be a security force response*
- B. *Incorrect, different security response/procedure*
- C. *Correct*
- D. *Incorrect, different OTO procedures response*

Technical Reference(s): OTO-SK-00001

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank #   R8396    
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    
55.43           

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #			
	K/A #	2.4.43		
	Importance Rating	3.2		
Knowledge of emergency communications systems and techniques.				

**Question #75**

Given the following plant conditions:

- The unit is stable at 100% power
- There are calibration activities on secondary plant instruments (feed flow) in progress.
- MCB annunciator 61A, Process Rad HiHi, alarms in the Control Room.

Which ONE of the following CORRECTLY describes the required communication between the Reactor Operator and Control Room Supervisor?

- A. Expected Alarm
- B. Unexpected Alarm
- C. Process Rad HiHi - Expected
- D. Process Rad HiHi - Unexpected

*Justification*

- A. *Incorrect, not an expected alarm. Expected alarms occur as a result of action being taken. Common mistake.*
- B. *Incorrect, missing annunciator # or description not an expected alarm.*
- C. *Incorrect. Not an expected alarm. Exp. alarms occur as a result of action being taken.*
- D. *Correct.*

Technical Reference(s): ODP-ZZ-00001 Addendum 01

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank #   R8645    
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    
55.43

NRC Site-Specific Written Examination  
Callaway Plant  
Reactor Operator

Comments:



NRC Site-Specific Written Examination  
Callaway Plant  
Senior Reactor Operator

<b>Examination Outline Cross-reference:</b>	<b>Level</b>	<b>RO</b>		<b>SRO</b>
	<b>Tier #</b>			1
	<b>Group #</b>			1
	<b>K/A #</b>	0008 AA2.14		
	<b>Importance Rating</b>			4.4
Ability to determine and interpret the following as they apply to the Pressurizer Vapor Space Accident: Saturation temperature monitor				

**Question #76**

Given the following plant conditions:

- A large vapor space LOCA has occurred.
- The operating crew has implemented the appropriate emergency procedures and is currently in E-1, Loss of Reactor or Secondary Coolant.
- The STA is monitoring status trees.
- The following indications are observed in the Main Control Room:
  - Train "A" Thermocouples indicate 720°F
  - Train "B" Thermocouples are de-energized
  - RVLIS indicates 40%
  - RCS pressure is 350 psig
  - No Reactor Coolant Pumps are in service

Which ONE of the following describes status of the reactor coolant, core cooling status, and mitigating actions?

The coolant status is \_\_\_\_\_, core cooling is \_\_\_\_\_ and will be mitigated by performing \_\_\_\_\_.

- A. superheated; DEGRADED; FR-C.2, Response to Degraded Core Cooling
- B. superheated; INADEQUATE; FR-C.1, Response to Inadequate Core Cooling
- C. saturated; SATURATED; FR-C.2, Response to Saturated Core Cooling
- D. saturated; ADEQUATE; E-1, Loss of Reactor or Secondary Coolant

*Justification*

- A. *Incorrect, superheated, inadequate, incorrect procedure.*
- B. *Correct*
- C. *Incorrect, superheated, degraded, incorrect procedure*
- D. *Incorrect, superheated, inadequate, incorrect procedure*

Technical Reference(s): CSF-1

Proposed references to be provided to applicants during examination: None

Learning Objective:

NRC Site-Specific Written Examination  
Callaway Plant  
Senior Reactor Operator

Question Source: Bank # \_\_\_\_\_  
Modified Bank # \_\_\_\_\_  
New  \_\_\_\_\_

Question History: Last NRC Exam  N/A

Question Cognitive Level:  
Memory or Fundamental Knowledge   
Comprehension or Analysis

10 CFR Part 55 Content:  
55.41   
55.43  5

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Senior Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	0011 2.3.4	
	Importance Rating		3.7
Knowledge of radiation exposure limits under normal or emergency conditions.			

**Question #77**

Given the following plant conditions:

- A LOCA outside containment has occurred 15 minutes ago at 0130.
- The Shift Manager has declared a SITE AREA EMERGENCY.
- The faulted line was manually isolated locally, however the Operations Technician performing the task was injured and CANNOT leave the area on his own.
- Initial dose estimates for the area are 90 R/hr primarily due to gamma radiation.
- The recovery time using one individual is estimated to take 10 minutes with a maximum time of 15 minutes.

Which ONE of the following describes the conditions concerning a rescue attempt?

- A. NO attempted rescue may be made since the exposure will exceed the allowed dose guidelines.
- B. A qualified individual selected by the Shift Manager may attempt the rescue with the approval of the Emergency Coordinator.
- C. Only a volunteer, after being made aware of all risks, can attempt the rescue when authorized by the Emergency Coordinator.
- D. A qualified individual selected by the Shift Manager may attempt the rescue once the authorization of the Vice President - Nuclear is obtained and concurrence given by the Radiological Protection Director.

*Justification*

- A. *Incorrect, exposures to save a life can be allowed*
- B. *Incorrect, must be a volunteer cannot be selected.*
- C. *Correct.*
- D. *Incorrect, must be a volunteer cannot be selected.*

Technical Reference(s): APA-ZZ-01000

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # \_\_\_\_\_  
Modified Bank # \_\_\_\_\_  
New   X  \_\_\_\_\_

NRC Site-Specific Written Examination  
Callaway Plant  
Senior 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 \_\_\_\_\_  
55.43 \_\_\_4\_\_\_

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Senior Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	029 EA2.01	
	Importance Rating		4.7
Ability to determine or interpret the following as they apply to a ATWS: Reactor nuclear instrumentation			

**Question #78**

Given the following plant conditions:

- A large LOCA has occurred resulting in a plant trip.
- The reactor trip breakers fail to open on the trip signal.
- The following plant conditions exist:
  - Reactor Power 40% and lowering.
  - Pressurizer Level 0%.
  - Pressurizer Pressure 1300 psig and lowering.
  - RVLIS - Pumps OFF 38%.
  - Core Exit TCs 1250°F and rising.
  - Containment Temp 175°F.

Which ONE of the following would be the correct implementation of the Emergency Operating Procedures after implementation of E-0, Reactor Trip or Safety Injection?

- A. E-1, Loss of Reactor or Secondary Coolant, to SACRG-1, Severe Accident CR Guideline Initial Response.
- B. FR-S.1, Response to Nuclear Power Generation, to FR-C.1, Response to Inadequate Core Cooling.
- C. FR-S.1, Response to Nuclear Power Generation, to SACRG-1, Severe Accident CR Guideline Initial Response.
- D. E-1, Loss of Reactor or Secondary Coolant, to FR-C.1, Response to Inadequate Core Cooling.

*Justification*

- A. *Incorrect. S.1 required, E-1 plausible due to LOCA.*
- B. *Incorrect. S.1 required, C.1 plausible due to CET.*
- C. *Correct.*
- D. *Incorrect. S.1 required, E-1 plausible due to LOCA, C.1 plausible due to CET.*

Technical Reference(s): E-0 and FR-S.1

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # 8632  
Modified Bank # \_\_\_\_\_

NRC Site-Specific Written Examination  
Callaway Plant  
Senior Reactor Operator

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 \_\_\_\_\_

55.43 \_\_5\_\_

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Senior Reactor Operator

<b>Examination Outline Cross-reference:</b>	<b>Level</b>	<b>RO</b>		<b>SRO</b>
	<b>Tier #</b>			1
	<b>Group #</b>			1
	<b>K/A #</b>	054 2.1.6		
	<b>Importance Rating</b>			4.8
Ability to manage the control room crew during plant transients.				

**Question #79**

Given the following plant conditions:

- The Callaway Plant is operating at 82% power.
- Both MFPs are in service.
- MFPs and main feed regulating valves are in automatic.
- "A" MFP trips.
- Steam flow is greater than Feed flow after the MFP trips.

Which ONE of the following describes correct procedure and the action directed by the SRO in response to above conditions?

**Procedure**

**Action**

- |   |   |
|---|---|
| A. OTO-AE-00001, Feedwater System Malfunction | A manual reactor trip.                                |
| B. OTO-MA-00008, Rapid Load Reduction         | A manual turbine load reduction to restore SG levels. |
| C. OTO-AE-00001, Feedwater System Malfunction | A manual start of AFW Pumps to restore SG levels.     |
| D. OTO-MA-00008, Rapid Load Reduction         | A manual turbine trip.                                |

*Justification:*

- A. *Correct.*
- B. *Incorrect. Manual load reduction required if power is <80% power.*
- C. *Incorrect. OTO requires unit trip and does not specify AFW pump start.*
- D. *Incorrect. OTO requires reactor trip if >80% power.*

Technical Reference(s): OTO-AE-00001

Proposed 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  
Senior Reactor Operator

Question Cognitive Level:

Memory or Fundamental Knowledge \_\_\_\_\_  
Comprehension or Analysis   X  

10 CFR Part 55 Content:

55.41 \_\_\_\_\_  
55.43   5  

Comments:



NRC Site-Specific Written Examination  
Callaway Plant  
Senior Reactor Operator

<b>Examination Outline Cross-reference:</b>	<b>Level</b>	<b>RO</b>		<b>SRO</b>
	<b>Tier #</b>			1
	<b>Group #</b>			1
	<b>K/A #</b>	055 EA2.04		
	<b>Importance Rating</b>			4.1
Ability to determine or interpret the following as they apply to a Station Blackout: Instruments and controls operable with only dc battery power available				

**Question #80**

Given the following plant conditions:

- The unit is at 100% power
- The Callaway Plant has just experienced a loss of all off site power
- Both "A" and "B" Diesel Generators failed to start and cannot be started

Which ONE of the following Control Room controls or indications will remain usable to control the initial response and the impact on the event classification?

- A. Digital Rod Position Indication (DRPI)  
Declare a Site Area Emergency
- B. Steam Generator ASD Controllers  
Declare an Alert
- C. Digital Rod Position Indication (DRPI)  
Declare an Alert
- D. Steam Generator ASD Controllers  
Declare a Site Area Emergency

*Justification:*

- A. Incorrect - PN07, non-safety related, alternate from PA01. Correct call*
- B. Incorrect - NN01/NN04. Wrong call*
- C. Incorrect - PN07/8, non-safety related, alternate from PA01/2. Wrong call*
- D. Correct. NN01/NN04, Group SS1.1 is the correct call*

Technical Reference(s): ECA-0.0, EIP-ZZ-00101, Addendum 1

Proposed 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:

NRC Site-Specific Written Examination  
Callaway Plant  
Senior Reactor Operator

Memory or Fundamental Knowledge \_\_\_\_\_  
Comprehension or Analysis   X  

10 CFR Part 55 Content:

55.41 \_\_\_\_\_  
55.43   1  

Comments:

**COMMENT:** The question asks the student to determine power supplies and to make an Emergency Action Level (EAL) declaration given the conditions:

- Offsite power is lost
- Both emergency diesel generators did not start automatically and cannot be started

The power supply portion of the question leads to answers B and D due to the Steam Generator ASD Controllers being supplied by NN01/NN04.

The EAL for loss of power requires a greater than 15 minute loss which is implied for the diesels but not given for the offsite power. This makes the question unclear. If all offsite power is restored in less than 15 minutes there is no EAL classification. If a single offsite power source is restored, the classification would be an Alert. If no offsite source is restored, the classification would be a Site Area Emergency.

In addition, the lesson plan objective in the Radiological Emergency Response operations lesson plan for EAL classification states "Determine the emergency classification for given indications and/or symptoms per EIP-ZZ-00101." The applicable sections of this procedure were not provided.

The KA reference for this question is for the power supply portion only.

Based on the stated information, both B and D are acceptable answers.

**NRC RESOLUTION:** Based on the sentence, "The Callaway Plant has just experienced a loss of all offsite power," and based on the fact that both diesel generators were lost and would not be restored, the applicant is asked to make an immediate EAL classification. The procedure governing EAL classification, EIP-ZZ-00101, states that EAL SS1.1, loss of offsite and both class 1E 4KV buses, is not applicable until 15 minutes has elapsed. The stem of the question is not clear as to when the applicant should make the classification in that it is asking what the impact will be on the event classification. Immediately following the loss of offsite power, there is no impact since no EAL is in effect until 15 minutes has elapsed. Presumably, the operators would use this time to contact dispatch to determine when power would be restored. This information is not given. Additionally, the stem should have asked what the impact would be if the conditions were to not change during a 15 minute interval.

Based on this, there is no correct answer for the question, and the question has been removed from the examination.

NRC Site-Specific Written Examination  
Callaway Plant  
Senior Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	E12 2.4.44	
	Importance Rating		4.4
Knowledge of emergency plan protective action recommendations.			

**Question #81**

You are the Shift Manager and receive the following information:

- A Steam Line Break on 'A' Main Steam Line
- 125 gpm primary to secondary leakage on 'A' S/G
- Lab analysis indicates RCS activity is 350 uCi/cc dose equivalent Iodine 131
- ALL MSIVs failed to close following the reactor trip

Which ONE of the responses below describes the proper initial protective action recommendation?

- A. SHELTER 2 mile radius and EVACUATE 5 miles downwind and SHELTER remainder of 10 mile EPZ.
- B. EVACUATE 2 mile radius and 5 miles downwind and SHELTER remainder of 10 mile EPZ.
- C. EVACUATE 2 mile radius and SHELTER remainder of 10 mile EPZ.
- D. EVACUATE 2 mile radius and EVACUATE 5 miles downwind.

*Justification*

- A. *Incorrect. Would evacuate 2 mile radius.*
- B. *Incorrect. Sheltering would not be done.*
- C. *Incorrect. Does not consider 5 miles. Implies they are sheltered.*
- D. *Correct.*

Technical Reference(s): EIP-ZZ-00212

Proposed references to be provided to applicants during examination: None

Learning Objective: T68.1020.6, 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\_\_\_

NRC Site-Specific Written Examination  
Callaway Plant  
Senior Reactor Operator

10 CFR Part 55 Content:

55.41 \_\_\_\_\_

55.43 \_\_1\_\_

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Senior Reactor Operator

<b>Examination Outline Cross-reference:</b>	<b>Level</b>	<b>RO</b>		<b>SRO</b>
	<b>Tier #</b>			1
	<b>Group #</b>			2
	<b>K/A #</b>	001 2.2.22		
	<b>Importance Rating</b>		4.7	
Knowledge of limiting conditions for operations and safety limits.				

**Question #82**

The plant is stable at 85% power with the following conditions:

- RCS  $T_{avg}$  is on program
- Pressurizer pressure is 2230 psig
- Control Bank 'D' is at 160 steps withdrawn
- The Control Rod Bank Selector is in AUTO

Control Bank 'D' then begins to step out at minimum rod speed. Rod Control System automatic rod blocks fail to function.

With no operator action, which ONE of the following describes the appropriate procedure to enter and what would generate the reactor trip to provide protection?

<u>Procedure</u>	<u>Generating Signal</u>
A. OTO-BB-00006, Pressurizer Pressure Control Malfunction	Pressurizer low pressure reactor trip
B. OTO-SF-00001, Rod Control Malfunction	Overtemperature $\Delta T$ reactor trip
C. OTO-SE-00001, Nuclear Instrument Malfunction	Power range positive rate trip
D. OTO-BB-00004, RCS RTD Channel Failures	Overpower $\Delta T$ reactor trip

*Justification*

- A. Incorrect. Rods stepping out would increase temp, which would increase pressure.  
 B. Correct.  $T_{avg}$  and pressure increase lowers the setpoint to trip first.  
 C. Incorrect. Would not reach setpoint at minimum rod speed.  
 D. Incorrect. Runback would occur first.

*As stated in TS Bases, the  $OT\Delta T$  reactor trip provides core protection against DNB for all combinations of pressure, power, coolant temperature and axial power distribution, provided only that : 1) the transient is slow with respect to piping transit delays from the core the the temperature detectors (about 2 seconds), and 2) pressure is within the range between the high and low pressure reactor trips. The USAR Accident Analysis confirms the  $OT\Delta T$  reactor trip is expected to limit this transient. For a slow RCCA withdrawal ( $3.0E-5 \Delta k/sec$ ) from full power... Reactor trip occurs on Overtemperature  $\Delta T$  reactor trip... The minimum DNBR reached during the transient is greater than the MDNBR (Minimum DNB Ratio).*

NRC Site-Specific Written Examination  
Callaway Plant  
Senior Reactor Operator

*OPΔT reactor trip prevents the power density anywhere in the core from exceeding that value at which fuel pellet centerline melting would occur (as compare to DNB).*

*The positive rate trip is designed for a rod ejection or an uncontrolled RCCS bank withdrawal. The setpoint would not be reached for this event prior to OTΔT.*

*Pressurizer pressure is expected to rise during the rod bank withdrawal accident and no challenge is provided to low pressure reactor trip.*

Technical Reference(s): OTO-SF-00001 and Tech Spec Basis B 3.3.1

Proposed 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         
Comprehension or Analysis   X  

10 CFR Part 55 Content:  
55.41         
55.43   2, 5  

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Senior Reactor Operator

<b>Examination Outline Cross-reference:</b>	<b>Level</b>	<b>RO</b>		<b>SRO</b>
	<b>Tier #</b>			1
	<b>Group #</b>			2
	<b>K/A #</b>	0067 AA2.17		
	<b>Importance Rating</b>			4.3
Ability to determine and interpret the following as they apply to the Plant Fire on Site: Systems that may be affected by the fire				

**Question #83**

Given the following plant conditions:

- The Callaway Plant is at 100% power.
- The Control Room has been evacuated due to a fire.

Which ONE of the following lists the equipment that would be available following the evacuation of the Control Room due to a fire and the appropriate EAL classification?

<u>Equipment Available</u>	<u>EAL Classification</u>
A. Reactor Coolant Pump "B" BBPCV0456A, PZR PORV ABPV0004, SG 'D' ASD	Alert
B. ABPV0004, SG 'D' ASD CCW Pump "D" TD Aux FW Pump	Alert
C. Reactor Coolant Pump "B" CCW Pump "D" TD Aux FW Pump	Unusual Event
D. ABPV0004, SG 'D' ASD BBPCV0456A, PZR PORV TD Aux FW Pump	Unusual Event

*Justification*

- A. *Incorrect, RCP's are tripped, PORV's power isolated, correct EAL*
- B. *Correct.*
- C. *Incorrect, RCP's are tripped, wrong EAL*
- D. *Incorrect, PORV's power isolated, wrong EAL*

Technical Reference(s): OTO-ZZ-00001

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # \_\_\_\_\_  
Modified Bank # \_R8496\_\_\_\_\_  
New \_\_\_\_\_

NRC Site-Specific Written Examination  
Callaway Plant  
Senior 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 \_\_\_\_\_  
55.43 \_\_\_5\_\_\_

Comments:



NRC Site-Specific Written Examination  
Callaway Plant  
Senior Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	E06 2.4.1	
	Importance Rating		4.8
Degraded Core Cooling – Knowledge of EOP entry conditions and immediate action steps.			

**Question #84**

Given the following plant conditions:

- The crew is responding to a large break LOCA.
- The following plant conditions exist:
  - Core Exit Temperature            750-800°F rising
  - RCS Subcooling                    100°F superheat
  - RCPs                                    Secured
  - PZR Level                            Off scale low
  - RVLIS (Pumps Off)                55% stable
  - IR SUR                                0.0 dpm
  - Containment Pressure              30 psig stable

Which ONE of the following procedures should the CRS directly transition to?

- A. FR-S.2, Response to Loss of Core Shutdown
- B. FR-I.3, Response to Voids in Reactor Vessel
- C. FR-C.2, Response to Degraded Core Cooling
- D. FR-Z.1, Response to High Containment Pressure

*Justification*

- A. *Incorrect. Do not meet entry conditions, 0 SUR instead of negative may make them choose it.*
- B. *Incorrect, Voids = PZR level high, may pick because of subcooling/superheat.*
- C. *Correct.*
- D. *Incorrect. lower priority orange path*

Technical Reference(s): CSF-1

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source:     Bank # \_\_\_\_\_  
                               Modified Bank # **\_ R12132** \_\_\_\_\_  
                               New \_\_\_\_\_

Question History: Last NRC Exam \_\_\_\_ N/A \_\_\_\_\_

NRC Site-Specific Written Examination  
Callaway Plant  
Senior Reactor Operator

Question Cognitive Level:

Memory or Fundamental Knowledge \_\_\_\_\_  
Comprehension or Analysis   X  

10 CFR Part 55 Content:

55.41 \_\_\_\_\_  
55.43   5  

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Senior Reactor Operator

<b>Examination Outline Cross-reference:</b>	<b>Level</b>	<b>RO</b>		<b>SRO</b>
	<b>Tier #</b>			1
	<b>Group #</b>			2
	<b>K/A #</b>	E14 EA2.2		
	<b>Importance Rating</b>			3.8
Ability to determine and interpret the following as they apply to the (High Containment Pressure) Adherence to appropriate procedures and operation within the limitations in the facility*s license and amendments.				

**Question #85**

The plant has experienced a large break LOCA. An SI, CISB, and CSAS have all actuated due to high containment pressure.

Which ONE of the following indications would be used by the Control Room Supervisor to transfer the Containment Spray Pump Suctions to the Recirc Sump?

- A. RWST EMPTY
- B. RWST LO-LO 2
- C. RWST LEV HI/LO
- D. RWST LO-LO 1 AUTO XFR

*Justification:*

- A. *Incorrect. Pump would be secured at this indication.*
- B. *Correct.*
- C. *Incorrect. This is the level to warn of Tech Spec limits being approached..*
- D. *Incorrect. This is the level at which the RHR pumps are realigned, not the CS pumps.*

Technical Reference(s): ES-1.3

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank #   R11795    
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         
55.43   5  

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Senior Reactor Operator

<b>Examination Outline Cross-reference:</b>	<b>Level</b>	<b>RO</b>		<b>SRO</b>
	<b>Tier #</b>			2
	<b>Group #</b>			1
	<b>K/A #</b>	026 A2.08		
	<b>Importance Rating</b>			3.7
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: Safe securing of containment spray when it can be done				

**Question #86**

Given the following plant conditions:

- Crew is performing actions of ES-1.3, Transfer to Cold Leg Recirculation.
- Neither RHR pump can be started.
- Containment pressure is 12.5 psig.
- Both Containment Spray pumps are running and aligned to the RWST.
- RWST level is 5%.
- SI has been reset.

Which ONE of the following describes the appropriate procedure to use and the crew actions regarding the Containment Spray pumps?

<u>Procedure</u>	<u>Action</u>
A. ECA-1.3, Sump Blockage Mitigation	Close HIS 8812A/B, RWST to RHR Pump A/B Suction.
B. ECA-1.3, Sump Blockage Mitigation	Place Containment Spray pumps in Pull-To-Lock.
C. ECA-1.1, Loss of Emergency Coolant Recirculation	Place Containment Spray pumps in Pull-To-Lock.
D. ECA-1.1, Loss of Emergency Coolant Recirculation	Close HIS 8812A/B, RWST to RHR Pump A/B Suction.

*Justification:*

- A. *Incorrect. Wrong procedure, wrong action.*
- B. *Incorrect. Wrong procedure, correct action.*
- C. *Correct.*
- D. *Incorrect. Correct procedure, wrong action.*

Technical Reference(s): ES-1.3, step 3, ECA-1.1

Proposed references to be provided to applicants during examination: None

Learning Objective:

NRC Site-Specific Written Examination  
Callaway Plant  
Senior Reactor Operator

Question Source: Bank # \_\_\_\_\_  
Modified Bank # \_\_\_\_\_  
New   X  

Question History: \_\_\_\_\_

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

10 CFR Part 55 Content:  
55.41 \_\_\_\_\_  
55.43   5  

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Senior Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	061 A2.03	
	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: Loss of dc power			

**Question #87**

Given the following plant conditions:

- A reactor trip occurs from 100% power.
- A fault on NK04 occurs resulting in a loss of the bus, after the reactor trip.

Which ONE of the following describes 1) the impact to the AFW system; and 2) the procedure to select for control of AFW flow for these conditions?

- A. 1) Normal control power to the "B" AFW pump is lost.  
2) E-0, Reactor Trip or Safety Injection.
- B. 1) Normal control power to the TDAFW pump is lost.  
2) OTO-NK-00002, Loss of Vital 125 VDC Bus, Attachment K.
- C. 1) Normal control power to the TDAFW pump is lost.  
2) E-0, Reactor Trip or Safety Injection.
- D. 1) Normal control power to the "B" AFW pump is lost.  
2) OTO-NK-00002, Loss of Vital 125 VDC Bus, Attachment K.

*Justification*

- A. Correct.
- B. Incorrect. See below. Stay in E-0, step 10 for control of AFW flow
- C. Incorrect. See below.
- D. Incorrect. See below. Stay in E-0, step 10 for control of AFW flow

*NK01 and NK04 supply additional DC loads such as diesel field flashing, breaker control power, main control board power and emergency lighting. These loads are not supplied by the other two buses, NK02 and NK03. For this reason, batteries NK11 and NK14 require additional capacity. Each battery is designed to have sufficient stored energy to supply the required emergency loads for 200 minutes following a loss of AC power.*

Technical Reference(s): E-0 and OTO-NK-00002, Att. K

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # \_\_\_\_\_  
Modified Bank # \_\_\_\_\_  
New   X  \_\_\_\_\_

NRC Site-Specific Written Examination  
Callaway Plant  
Senior 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 \_\_\_\_\_  
55.43 \_\_\_5\_\_\_

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Senior Reactor Operator

<b>Examination Outline Cross-reference:</b>	<b>Level</b>	<b>RO</b>		<b>SRO</b>
	<b>Tier #</b>			2
	<b>Group #</b>			1
	<b>K/A #</b>	063 A2.02		
	<b>Importance Rating</b>			3.1
Ability to (a) predict the impacts of the following malfunctions or operations on the DC electrical systems; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of ventilation during battery charging				

**Question #88**

Given the following plant conditions:

- The Callaway Plant is in MODE 5, preparing for a refueling outage.
- RHR Train B is in service, providing RCS cooling.
- RCS TEMPERATURE                      175°F
- RCS LEVEL                                      50 INCHES
- SG A WR LEVEL                              88 %
- SG D WR LEVEL                              90 %
- RHR Pump B trips due to a Ground on ESF Bus NB02.

Which ONE of the following describes the appropriate procedure and action that is required?

<u>Procedure</u>	<u>Action</u>
A. OTO-EJ-00001, Loss of RHR	Dispatch an Equipment Operator to vent the RHR suction header prior to starting RHR Pump A
B. OTO-EJ-00001, Loss of RHR	Evacuate non-essential personnel from containment and complete containment closure
C. OTO-EJ-00003, Loss of RHR While Operating at Reduced Inventory	Evacuate non-essential personnel from containment and complete containment closure
D. OTO-EJ-00003, Loss of RHR While Operating at Reduced Inventory.	Dispatch an Equipment Operator to vent the RHR suction header prior to starting RHR Pump A

*Justification*

- A. *Incorrect, wrong procedure, wrong action, action is for different plant condition*
- B. *Incorrect, wrong procedure, correct action*
- C. *Correct*
- D. *Incorrect, correct procedure, wrong action, action is for different plant condition*



NRC Site-Specific Written Examination  
Callaway Plant  
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Technical Reference(s):

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # \_\_\_\_\_  
Modified Bank #R12085 \_\_\_\_\_  
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 \_\_\_\_\_  
55.43 \_\_\_5\_\_\_

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Senior Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	064 2.4.45	
	Importance Rating		4.3
Ability to prioritize and interpret the significance of each annunciator or alarm.			

**Question #89**

Given the following plant conditions:

- The Control Room receives "NE01 Trouble" annunciator.
- The Secondary Operations Technician reports that annunciator 6E, DC Control Power Failure Alarm, is lit at the NE01 local alarm panel.
- On panel KJ121, IL1 and IL2 lights are OFF, IL3 and IL4 lights are ON.

Which ONE of the following describes 1) the effect on the Diesel Generator; and 2) the Technical Specification implications?

- A. 1) NE01 is OPERABLE if starting air pressure is maintained 610 to 640 psig.  
2) No LCO actions are required.
- B. 1) NE01 is INOPERABLE since the fuel oil transfer pump is disabled.  
2) Verify Off-site power circuits aligned properly within 1 hour.
- C. 1) NE01 is INOPERABLE since diesel start circuits are disabled.  
2) Verify Off-site power circuits aligned properly within 1 hour.
- D. 1) NE01 is OPERABLE since the fuel oil transfer pump is disabled and not required for operability.  
2) No LCO actions are required.

*Justification*

- A. *Incorrect, EDG is inop, TS Action B is required*
- B. *Incorrect, wrong failure mode, correct TS action*
- C. *Correct.*
- D. *Incorrect, EDG is inop, TS Action B is required*

*Diesel Start circuits have lost power (lights 1 and 2) making the EDG inop.*

Technical Reference(s): TS 3.8.1 and T61.0110.6, Standby Generation

Proposed 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  
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Question Cognitive Level:

Memory or Fundamental Knowledge \_\_\_\_\_  
Comprehension or Analysis   X  

10 CFR Part 55 Content:

55.41 \_\_\_\_\_  
55.43   2  

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Senior Reactor Operator

<b>Examination Outline Cross-reference:</b>	<b>Level</b>	<b>RO</b>		<b>SRO</b>
	<b>Tier #</b>			2
	<b>Group #</b>			1
	<b>K/A #</b>	073 2.1.28		
	<b>Importance Rating</b>			4.1
Process Radiation Monitoring (PRM) System / Knowledge of the purpose and function of major system components and controls.				

**Question #90**

Callaway Plant RCS is at 220°F and stable with a maintenance outage in progress. The RM-11 console alarms due to GT-RE-59, Containment Area Radiation Monitor, indicating LIGHT BLUE.

The alarm message “Monitor Loss of RM-23 Communications” is received on the printer. No other alarm messages are received from the RM-11.

Which ONE of the following is the required Tech Spec action for this condition?

- A. Initiate the preplanned alternate method of monitoring containment radiation. Submit a report within 14 days with alternate method, cause and restoration schedule.
- B. Verify GT-RE-60 operating and communicating with its RM-23 and repair GT-RE-59 within 30 days.
- C. Restore GT-RE-59 to OPERABLE within 7 days of failure.
- D. No ACTION required. GT-RE-59 not required for this mode of operation.

*Justification*

- A. Incorrect. Plant is in Mode 4, GT-RE-59 only required to be operable in Modes 1-3.
- B. Incorrect. Plant is in Mode 4, GT-RE-59 only required to be operable in Modes 1-3.
- C. Incorrect. Plant is in Mode 4, GT-RE-59 only required to be operable in Modes 1-3.
- D. Correct.

Technical Reference(s): Tech Spec, PAM Instrumentation, 3.3.3

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank #   R8459    
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  
Senior Reactor Operator

10 CFR Part 55 Content:

55.41 \_\_\_\_\_  
55.43   7  

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Senior Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	016 A2.02	
	Importance Rating		3.2*
Ability to (a) predict the impacts of the following malfunctions or operations on the NNIS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of power supply			

**Question #91**

The following conditions exist:

- The unit is stable at 100% power
- All systems are properly aligned in automatic

Control rods start to move in and many annunciators go into alarm. You notice that the controlling narrow range level channels for 2 out of 4 Steam Generators have gone to zero and the feed regulating valves for 2 out of 4 Steam Generators are ramping open.

Which ONE of the following Off-Normal Operating Procedures should the Control Room Supervisor use for this event?

- A. OTO-NN-00001, Loss of Safety Related Instrument Power
- B. OTO-KA-00001, Partial or Total Loss of Instrument Air
- C. OTO-NK-00001, Failure of NK Battery Charger
- D. OTO-NB-00001, Loss of Power to NB01

*Justification:*

- A. *Correct.*
- B. *Incorrect. Loss of air would affect all steam generators.*
- C. *Incorrect. Would still have battery for power if a battery charger fails.*
- D. *Incorrect. Would affect more instrumentation would affect major components.*

Technical Reference(s): OTO-NN-00001

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # L13352  
Modified Bank # \_\_\_\_\_  
New \_\_\_\_\_

Question History: Last NRC Exam N/A

Question Cognitive Level:

Memory or Fundamental Knowledge \_\_\_\_\_  
Comprehension or Analysis X

NRC Site-Specific Written Examination  
Callaway Plant  
Senior Reactor Operator

10 CFR Part 55 Content:

55.41 \_\_\_\_\_  
55.43 \_\_5\_\_

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Senior Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	034 K1.03	
	Importance Rating		2.7*
Knowledge of the physical connections and/or cause-effect relationships between the Fuel Handling System and the following systems: CVCS			

**Question #92**

Given the following plant conditions:

- The Callaway Plant is in Mode 6.
- Fuel movement in progress.
- Audible and Source Range counts rising.
- Annunciator 65A, SR High Flux At Shutdown, alarms.

Which ONE of the following describes:

- 1) action(s) that should be directed
- 2) procedure that should be entered?

- A. 1) Place the high flux at shutdown switch for each SRM to block.  
2) OTO-ZZ-00003, Loss of Shutdown Margin
- B. 1) Suspend core alterations and emergency borate.  
2) OTO-ZZ-00003, Loss of Shutdown Margin
- C. 1) Place the high flux at shutdown switch for each SRM to block.  
2) OTO-KE-00001, Fuel Handling Accident
- D. 1) Suspend core alterations and emergency borate.  
2) OTO-KE-00001, Fuel Handling Accident

*Justification:*

- A. *Incorrect. These actions for an invalid alarm. Correct procedure.*
- B. *Correct. EIP/OTO requires.*
- C. *Incorrect. These actions for an invalid alarm. SRM counts increasing make this alarm valid.*
- D. *Incorrect. Correct action, wrong procedure.*

Technical Reference(s): ETP-ZZ-00035 and OTO-ZZ-00003

Proposed 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  
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Question Cognitive Level:

Memory or Fundamental Knowledge \_\_\_\_\_  
Comprehension or Analysis   X  

10 CFR Part 55 Content:

55.41 \_\_\_\_\_  
55.43   5  

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Senior Reactor Operator

<b>Examination Outline Cross-reference:</b>	<b>Level</b>	<b>RO</b>		<b>SRO</b>
	<b>Tier #</b>			2
	<b>Group #</b>			2
	<b>K/A #</b>	055 2.4.16		
	<b>Importance Rating</b>			4.4
Knowledge of EOP implementation hierarchy and coordination with other support procedures or guidelines such as, operating procedures, abnormal operating procedures, and severe accident management guidelines.				

**Question #93**

Given the following plant conditions:

- Callaway Plant startup is in progress following a refueling outage.
- The turbine load was being raised per OTO-ZZ-00003, Plant Startup Hot Zero Power to 30% Power.
- Annunciator 116B, Cond A Vac Lo, alarmed.
- Turbine load is currently at 300 MWe and condenser backpressure is 12.5 inches HgA and stable.

Which ONE of the following actions will the CRS take to stabilize the plant?

- A. Secure from the load increase and immediately start reducing load per OTG-ZZ-00005, Plant Shutdown 20% Power to hot Standby.
- B. Secure from the load increase, stabilize the plant at the current power level, and monitor condenser vacuum.
- C. Monitor condenser vacuum and continue with the load increase.
- D. Trip the turbine and go to OTO-AC-00001, Turbine Trip.

*Justification*

- A. *Incorrect; These are the actions that would be performed if the condenser vacuum was in the operating range and vacuum still decreasing.*
- B. *Incorrect; These are the actions that would be performed if the condenser vacuum was in the operating range.*
- C. *Incorrect; The load increase should be stopped.*
- D. *Correct.*

Technical Reference(s): OTO-AD-00001

Proposed 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  
Senior Reactor Operator

Question Cognitive Level:

Memory or Fundamental Knowledge \_\_\_\_\_  
Comprehension or Analysis   X  

10 CFR Part 55 Content:

55.41 \_\_\_\_\_  
55.43   5  

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Senior Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		
	K/A #	2.1.5	
	Importance Rating		3.9
Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc.			

**Question #94**

Which ONE of the following should the Control Room Supervisor do if an employee calls from home and reports he will not be coming to work due to an occupational injury?

- A. Inform the individual he must see a Company authorized medical provider that day.
- B. Inform the individual he must have a doctor's permission prior to returning to work.
- C. Inform the individual he must see a Company authorized medical provider the first day back to work.
- D. Complete a Form 70 or CAR with the individual.

*Justification*

- A. Correct.
- B. Incorrect. *Permission slip is not needed.*
- C. Incorrect. *Doctor must be seen the day of the call, not the first day back to work.*
- D. Incorrect. *Both of these will be done, but not by the CRS over the phone.*

Technical Reference(s): APA-ZZ-00835

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # \_003A0H02A\_  
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\_\_  
55.43 \_\_5\_\_

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Senior Reactor Operator

<b>Examination Outline Cross-reference:</b>	<b>Level</b>	<b>RO</b>		<b>SRO</b>
	<b>Tier #</b>			3
	<b>Group #</b>			
	<b>K/A #</b>	2.2.15		
	<b>Importance Rating</b>			4.3
Ability to determine the expected plant configuration using design and configuration control documentation, such as drawings, line-ups, tag-outs, etc.				

**Question #95**

The Shift Manager can authorize which ONE of the following operations of a component that has a Local Control Tag hanging on it?

- A. Operation of MCB switch BB HIS-38 by Relay Test personnel during a surveillance.
- B. Removal of control power fuse block from NB0202 cubicle.
- C. Racking a 4160VAC breaker when work is scheduled on a downstream component.
- D. Installation of grounds on PA01.

*Justification*

- A. *Incorrect. Relay Test personnel not licensed, cannot operate CR components.*
- B. *Correct.*
- C. *Incorrect. Local Control would not be used.*
- D. *Incorrect. Local Control would not be used.*

Technical Reference(s): APA-ZZ-00310

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank #   R8621    
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             
55.43   3  

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Senior Reactor Operator

<b>Examination Outline Cross-reference:</b>	<b>Level</b>	<b>RO</b>		<b>SRO</b>
	<b>Tier #</b>			3
	<b>Group #</b>			
	<b>K/A #</b>	2.2.40		
	<b>Importance Rating</b>			4.7
Ability to apply Technical Specifications for a system.				

**Question #96**

Callaway Plant is in Mode 2 when the following equipment problems occur:

- The “B” CCP is declared inoperable at 1200 on 11/25/08
- The “A” SI pump is declared inoperable at 1200 on 11/26/08

Which ONE of the following actions satisfies Technical Specifications?

- A. Restore the “B” CCP and the “A” SI pump by 1200 on 11/28/08
- B. Restore the “B” CCP or the “A” SI pump by 1200 on 11/28/08
- C. Restore the “B” CCP and the “A” SI pump by 1200 on 11/29/08
- D. Immediately enter TS LCO 3.0.3

*Justification:*

- a. Correct.
- b. Incorrect Both pumps must be restored.
- c. Incorrect. The CCP must be operable by 11/28.
- d. Incorrect. TS 3.0.3 is not required.

Technical Reference(s): TS 3.5.2 and TS 1.3

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank #   R13610    
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         
55.43   2  

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Senior Reactor Operator

<b>Examination Outline Cross-reference:</b>	<b>Level</b>	<b>RO</b>		<b>SRO</b>
	<b>Tier #</b>			3
	<b>Group #</b>			
	<b>K/A #</b>	2.3.13		
	<b>Importance Rating</b>			3.8
Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.				

**Question #97**

Given the following plant conditions:

- Core off-load is in progress.
- Increased bubbling from the fuel bundle.
- Increased radiation indicated at the refueling machine area radiation monitor.

Which ONE of the following describes the appropriate procedure and the required actions in response to this event?

	<u><b>Procedure</b></u>	<u><b>Appropriate Action</b></u>
A.	OTO-KE-00001, Fuel Handling Accident	Return fuel assembly to reactor vessel, evacuate unnecessary personnel from containment, close one air lock door
B.	OTS-KE-00013, Refueling Machine	Return fuel assembly to reactor vessel, initiate Containment Purge Isolation Signal, place both RHR trains in service
C.	OTS-KE-00013, Refueling Machine	Contact Reactor Engineering, place damaged fuel assembly in change fixture, notify HP
D.	OTO-KE-00001, Fuel Handling Accident	Contact Reactor Engineering, initiate Containment Purge Isolation Signal, evacuate all personnel from containment

*Justification*

- A. *Correct.*  
 B. *Incorrect. Wrong procedure for actions. This procedure is used for moving the fuel. Incomplete actions*  
 C. *Incorrect. Wrong procedure for actions. Wrong location to store assembly*  
 D. *Incorrect. Correct procedure, incomplete/incorrect actions*

Technical Reference(s): OTO-KE-00001

Proposed references to be provided to applicants during examination: None

Learning Objective:

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         
Comprehension or Analysis   X  

10 CFR Part 55 Content:  
55.41         
55.43   5  

Comments:



NRC Site-Specific Written Examination  
Callaway Plant  
Senior Reactor Operator

<b>Examination Outline Cross-reference:</b>	<b>Level</b>	<b>RO</b>		<b>SRO</b>
	<b>Tier #</b>			3
	<b>Group #</b>			
	<b>K/A #</b>	2.3.14		
	<b>Importance Rating</b>			3.8
Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.				

**Question #98**

Given the following plant conditions:

- Refueling is in progress
- A spent fuel element is being moved from the reactor to the upender
- The spent fuel element is dropped to the bottom of the canal

Which ONE of the following products released from the ruptured spent fuel element will present the most immediate hazard and what is the first procedural action to be directed?

<u><b>Hazard</b></u>	<u><b>Procedural Action</b></u>
A. Hydrogen gas.	Initiate CRVIS and evacuate Containment
B. Alpha radiation from fission products.	Initiate CPIS and evacuate Containment
C. Gamma radiation from fission and corrosion products.	Initiate CPIS and evacuate Containment
D. Gamma radiation from Iodine and Krypton gases.	Initiate CRVIS and evacuate Containment

*Justification*

- A. *Incorrect. Wrong Hazard, right action*
- B. *Incorrect. Wrong Hazard, wrong action, step 12*
- C. *Incorrect. Wrong Hazard, wrong action, step 12*
- D. *Correct.*

Technical Reference(s): OTO-KE-00001, step 2

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # \_\_\_\_\_  
Modified Bank # \_\_\_\_\_  
New  \_\_\_\_\_

Question History: Last NRC Exam \_\_\_\_ N/A \_\_\_\_\_

NRC Site-Specific Written Examination  
Callaway Plant  
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Question Cognitive Level:

Memory or Fundamental Knowledge \_\_\_\_\_  
Comprehension or Analysis   X  

10 CFR Part 55 Content:

55.41 \_\_\_\_\_  
55.43   6  

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Senior Reactor Operator

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		
	K/A #	2.4.21	
	Importance Rating		4.6
Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.			

**Question #99**

Given the following plant conditions:

- Reactor Power is 100%.
- “A” S/G tube ruptures (300 gpm).
- “A” S/G safety valve fails open when the turbine is tripped.
- Automatic and manual reactor trips from the Control Room fail to trip the reactor **BUT** it can be tripped locally.

Which ONE of the following describes the required procedure sequences?

E-0, Reactor Trip or Safety Injection, to FR-S.1, Response to Nuclear Power generation/ATWS, to . . .

- A. E-2, Faulted Steam generator Isolation, to  
E-3, Steam Generator Tube Rupture, to  
ECA-3.1, SGTR With Loss of Reactor Coolant – Subcooled Recovery Desired
- B. E-0, Reactor Trip or Safety Injection, to  
E-2, Faulted Steam generator Isolation, to  
ECA-3.1, SGTR With Loss of Reactor Coolant – Subcooled Recovery Desired
- C. E-0, Reactor Trip or Safety Injection, to  
E-3, Steam Generator Tube Rupture, to  
ECA-3.1, SGTR With Loss of Reactor Coolant – Subcooled Recovery Desired
- D. E-0, Reactor Trip or Safety Injection, to  
E-2, Faulted Steam generator Isolation, to  
E-3, Steam Generator Tube Rupture, to  
ECA-3.1, SGTR With Loss of Reactor Coolant – Subcooled Recovery Desired

*Justification:*

- A. *Incorrect. Entry criteria for FR-S.1 has been met.*
- B. *Incorrect. E-2 transitions to E-3. E-3 should transition ECA-3.1.*
- C. *Incorrect. E-0 will transition to E-2.*
- D. *Correct.*

Technical Reference(s): FR-S.1, E-0, E-2, E-3

Proposed references to be provided to applicants during examination: None

NRC Site-Specific Written Examination  
Callaway Plant  
Senior Reactor Operator

Learning Objective:

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         
55.43   5  

Comments:

NRC Site-Specific Written Examination  
Callaway Plant  
Senior Reactor Operator

<b>Examination Outline Cross-reference:</b>	<b>Level</b>	<b>RO</b>		<b>SRO</b>
	<b>Tier #</b>			3
	<b>Group #</b>			
	<b>K/A #</b>	2.4.46		
	<b>Importance Rating</b>			4.2
Ability to verify that the alarms are consistent with the plant conditions.				

**Question #100**

During a normal reactor startup, reactor power is rising on a stable 0.5 dpm SUR with control bank D rods at 125 steps.

As the operator inserts rods to level power at  $10^{-8}$  amps, the following annunciators alarm:

- 79C, Control Rod Dev
- 81B, Rod At Bottom

Which ONE of the following describes the cause of the alarms and the appropriate procedure?

**Alarm Cause**

**Procedure Selection**

- |                          |  |
|--------------------------|--|
| A. Multiple dropped rods | OTO-SF-00001, Rod Control Malfunctions |
| B. One dropped rod       | OTO-SF-00001, Rod Control Malfunctions |
| C. Multiple dropped rods | E-0, Reactor Trip or Safety Injection  |
| D. One dropped rod       | E-0, Reactor Trip or Safety Injection  |

*Justification*

- A. *Incorrect. Annunciators do not support multiple dropped rods (81A). Correct procedure*  
 B. *Correct.*  
 C. *Incorrect. Annunciators do not support multiple dropped rods (81A). Wrong procedure*  
 D. *Incorrect. Correct indications, wrong procedure*

*Requires synthesis of information in ARPs with theoretical knowledge of reactivity effects of dropped rod.*

Technical Reference(s): OTA-RK-00022 (Add 81B), OTO-SF-00001, Steps 1 through 5, E-0

Proposed 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  
Senior Reactor Operator

Question Cognitive Level:

Memory or Fundamental Knowledge \_\_\_\_\_  
Comprehension or Analysis   X  

10 CFR Part 55 Content:

55.41 \_\_\_\_\_  
55.43   5  

Comments: