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Examination Outline Cross-Reference: Q01

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	000007 G2.1.7	
Importance Rating	4.4	

K/A Statement: Reactor Trip: Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation

Proposed Question: Q1

1. An automatic reactor trip has occurred from 100% RTP.

   Approximately TWO MINUTES after the trip the following indications are present in the Control Room:
   - RCS pressure is 1675 psig
   - RCS temperature is 540ºF and slowly rising
   - Pressurizer level is 31 inches
   - SG1 pressure is 980 psig
   - SG2 pressure is 870 psig and slowly rising
   - SG1 level is 110 inches and lowering
   - SG2 level is 40 inches
   - CTMT pressure is 14.8 psia

   Based on the above conditions, which of the following actions is REQUIRED?

   A. Isolate Letdown.

   B. Trip ALL Reactor Coolant Pumps.

   C. Take manual control of Main Feedwater flow to BOTH SGs.

   D. Actuate BOTH channels of SFRCS INITIATION and ISOLATION.

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Proposed Answer: **A**

Explanation: Q1

A. Correct – Per DB-OP-02000, Attachment 1, Attachment 8, and their associated Carry-Over Steps Page, IF AT ANY TIME Pressurizer Level is less than 40 inches, isolate letdown by shutting MU2B.

B. Incorrect – SCM is greater than 20°F, therefore RCPs are NOT required to be tripped. Plausible since RCS Pressure is lower than expected for a plant trip without complications.

C. Incorrect – SG 1 level is 110 inches and lowering (normal for this event) It is not required to take manual control of feedwater flow unless they are being overfed or underfed. Plausible since SG level control setpoint is 40 inches for this scenario.

D. Incorrect - Based on the indications given the Overcooling event has been terminated therefore SFRCS Initiation and Isolation is not required. Plausible since it would be required to Initiate AND Isolate SFRCS using SFRCS MANUAL ACTUATION switches if the Overcooling Event cannot be stopped IAW DB-OP-02000 Step 7.10.

Technical Reference(s):	DB-OP-02000, RPS, SFAS, SFRCS TRIP, OR SG TUBE RUPTURE, R32 (p374)		
Reference(s) Provided	None		
Learning Objective:	OPS-GOP-306-06A		
Question Source:	Bank #	____	
	Modified Bank #	X	(note changes or attach parent)
	New	____	
Question History:	Last NRC Exam	2009 Q51	(if applicable)
Question Cognitive Level:	Memory / **F**undamental Knowledge		____
	Comprehension / Analysis / **H**igher Cognitive		X
10 CFR Part 55 Content:	55.41	(b)(5)	
	55.43	____	

Comments:
The question meets Tier 1 criteria as it tests information contained in the site's procedures, specifically the Emergency Operating Procedures.

Examination Outline Cross-Reference: Q02

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	000008 AK2.05	
Importance Rating	3.9	

K/A Statement: Knowledge of the relationship between PRESSURIZER VAPOR SPACE ACCIDENT and the following systems or components: RCS

Proposed Question: Q2

2. The Unit is at 100% RTP.

The following conditions are noted:
- RCS pressure is lowering
- ALL Pressurizer Heater Banks are ON
- Pressurizer UNCOMPENSATED Level is rising
- Containment radiation levels are rising

Which of the following describes the most likely event causing these indications?

A. RCS leak in the letdown piping.

B. Low failure of the pressure input the Pressurizer Heaters.

C. Pressurizer vapor space leak.

D. RC2, PRESSURIZER SPRAY VALVE, failed OPEN.

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Proposed Answer: **C**

Explanation: Q2

A. Incorrect – PRZ Level will lower during an RCS leak, not rise. Plausible since other indications could be an RCS Leak in the Letdown piping.

B. Incorrect – A low failure of the pressure input would not cause rising ctmt rad levels. Plausible since other indications could be caused by a low failure of the pressure input.

C. Correct – These are indications of a PZR Vapor Leak.

D. Incorrect – A failed open spray valve will not cause rad levels to change. Plausible since other indications could be caused by a failed open spray valve.

Technical Reference(s):	SD-039A R7 (p2-54), DB-OP-02513 R14 (p10, 11)		
Reference(s) Provided	None		
Learning Objective:	OPS-GOP-113-05K		
Question Source:	Bank #		
	Modified Bank #		(note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		(if applicable)
Question Cognitive Level:	Memory / **F**undamental Knowledge		
	Comprehension / Analysis / **H**igher Cognitive	X	
10 CFR Part 55 Content:	55.41	(b)(8) & (10)	
	55.43		

Comments

The question meets Tier 1 criteria as it tests the assessment of integrated plant response to an abnormal event across multiple plant systems.

Examination Outline Cross-Reference: Q03

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	000011 EK3.08	
Importance Rating	4.3	

K/A Statement: Large-Break LOCA: Knowledge of the reasons for the following responses and/or actions as they apply to LARGE-BREAK LOCA: Containment sump recirculation

Proposed Question: Q3

3. The plant is operating at 100% RTP.

A Design Basis Loss of Coolant Accident (DBLOCA) occurs.

Which of the following describes the bases for the Borated Water Storage Tank (BWST) level at which the operators transfer Low Pressure Injection (LPI) Suction to the Emergency Sump?

The specified BWST level for transfer of LPI suction to the Emergency Sump was selected to _____.

A. ensure the reactor remains subcritical during the Blowdown Phase

B. prevent boron precipitation in the core during the Recirculation Phase

C. ensure sufficient LPI Pump NPSH during the DBLOCA Recirculation Phase

D. prevent flooding of the Decay Heat Valve Tank in CTMT

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Proposed Answer: **C**

Explanation: Q3

A. Incorrect – Plausible since improper boron concentrations and volume in the BWST could result in a reduction of SDM. Incorrect since the CFTs add water to the RCS during the Blowdown Stage. The BWST will not reach the transfer level during the Blowdown phase.

Incorrect – Plausible since boron precipitation could occur post LOCA during the recirculation phase. Incorrect since boron precipitation is based on plant conditions during the recirculation phase and will not be a factor when the transfer occurs.

Correct – When ECCS pump suction is transferred to the sump, there must be sufficient water in the sump to ensure adequate net positive suction head (NPSH) for the LPI and containment spray pumps.

Incorrect – Incorrect since the vent pipe extends to a level above the maximum post-LOCA water level (SD-042 p2-31) Plausible since performing the transfer at a lower level would cause CTMT level to rise.

Technical Reference(s): SD-042 R7 item 2.1.2.3 (p2-6). TS Bases (pB 3.5.4-2).

Reference(s) Provided: None

Learning Objective: OPS-GOP-309-04K

Question Source: Bank # ____

Modified Bank # X (note changes or attach parent)

New ____

Question History: Last NRC Exam 2015 Q4 (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge X

Comprehension / Analysis / **H**igher Cognitive ____

10 CFR Part 55 Content: 55.41 (b)(5) & (10)

55.43 ____

Comments:

The question meets Tier 1 criteria as it tests the assessment of integrated plant response to an emergency event across multiple plant systems.

Examination Outline Cross-Reference: Q04

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	000015 AA1.02	
Importance Rating	3.2	

K/A Statement: Reactor Coolant Pump Malfunctions: Ability to operate and/or monitor the following as they apply to REACTOR COOLANT PUMP MALFUNCTIONS: RCP oil reservoir level and alarm indications

Proposed Question: Q4

4. Reactor power is 100% RTP.

A rapid drop in RCP 1-1 Motor Lower Bearing oil level occurs.

The following indications are noted in the Control Room:
- PPC Point L788, RCP 1-1 Mtr Lwr Brg Low Oil Level, is in alarm.
- PPC Point T787, RCP 1-1 Mtr Lwr Brg Mt, is in alarm.
- RCP 1-1 Motor Lower Bearing temperature is 195ºF and rising.

Given the above information, the operating crew MUST_____.

A. trip the reactor per DB-OP-02000, RPS, SFAS, SFRCS TRIP OR SG TUBE RUPTURE and then stop RCP 1-1

B. perform a power reduction to 72% and then stop RCP 1-1 per DB-OP-02515, REACTOR COOLANT PUMP AND MOTOR ABNORMAL OPERATION

C. complete a Reactor Shutdown to MODE 3 per DB-OP-02504, RAPID SHUTDOWN and then stop RCP 1-1 for inspection

D. add oil to RCP 1-1 oil system per DB-OP-06005, RC PUMP OPERATION and then monitor to ensure alarming conditions clear

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Proposed Answer: **B**

Explanation: Q4

A. Incorrect – Plausible as this would be the action directed by DB-OP-02515 if there were only 3 RCPs running when this condition occurred. Since the stem told us that the plant was at 100% reactor power when the alarms came in, we know that all 4 RCPs were running.

B. Correct – DB-OP-02515 section 4.6 directs the crew to Attachment 1 when oil level alarms are received and/or bearing temperatures exceed 190 °F. Attachment 1 has the crew perform a rapid down power to ≤72% and then stop RCP 1-1.

C. Incorrect – Plausible as the applicant may assume the plant must be shutdown to inspect for the oil leak as this could be construed as a fire hazard in containment.

D. Incorrect – Plausible as with the reactor shutdown and before RCP 1-1 is started operators would check oil levels and then monitor oil levels and bearing temperatures with the pump just started to ensure proper operation per DB-OP-06005. The RCP oil system is not to be breached with RCS temperature > 375 °F without Shift Manager and Operations Fire Protection Advisor permission due to flammability concerns.

Technical Reference(s): DB-OP-02515 R23, Section 4.6 (p44) and Attachment 1 (p51)

Reference(s) Provided	None		
Learning Objective:	OPS-GOP-115-03K		
Question Source:	Bank #		
	Modified Bank #	X	(note changes or attach parent)
	New		
Question History:	Last NRC Exam	TMI 2015 Q10	(if applicable)
Question Cognitive Level:	Memory / **F**undamental Knowledge		X
	Comprehension / Analysis / **H**igher Cognitive		
10 CFR Part 55 Content:	55.41	(b)(5) & (10)	
	55.43		

Comments:

The question meets Tier 1 criteria as it tests information contained in the site's procedures, specifically the Abnormal Operating Procedures.

Examination Outline Cross-Reference: Q05

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	000022 AA2.08	
Importance Rating	3.0	

K/A Statement: Ability to determine and/or interpret the following as they apply to LOSS OF REACTOR COOLANT MAKEUP: RCP seal flows, temperatures, pressures, and vibrations

Proposed Question: Q5

5. The plant is at 100% RTP.

Subsequently, the following conditions are received:

- MU32, PRESSURIZER LEVEL CONTROL, opening
- MU19, RCP SEAL INJECTION FLOW CONTROL, opening
- Pressurizer Level is lowering
- MUT Level is rising
- 6-6-C, SEAL INJ TOTAL FLOW is LIT
- RCP Seal Return temperatures are rising

Subsequently, the crew entered DB-OP-02512, MAKEUP AND PURIFICATION SYSTEM MALFUNCTIONS and isolated letdown.

The operating crew MUST now _____.

A. trip the Reactor and stop all RCPs

B. place the Alternate MU Injection line in service

C. place the standby Seal Injection Filter in service

D. isolate RCP Seal Injection and the Normal MU flowpath

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Proposed Answer: **D**

Explanation: Q5

A. Incorrect – Plausible as this would be the action if both Seal Injection (MUP) and seal cooling (CCW) are lost to the RCPs.

B. Incorrect – Plausible if the applicant correctly notes a loss of a MUP but assumes that makeup is recovered by establishing the Alternate Injection lineup.

C. Incorrect – Plausible as this is the action if the loss of seal injection was due to a clogged filter.

D. Correct – A loss of a MUP will cause a loss of RCS MU and seal injection flow to the RCPs. DB-OP-02512, Section 4.1 will require letdown to be isolated first then a check to verify CCW to the RCPs. There is no indication of a loss of CCW to the RCPs in the stem of this question. The seal injection line AND MU32 MUST be isolated prior to starting the standby MUP.

Technical Reference(s): DB-OP-02512 R19, Makeup and Purification System Malfunctions, (p12-14).

Reference(s) Provided: None

Learning Objective: OPS-GOP-112-02K

Question Source: Bank # _____

Modified Bank # _____ (note changes or attach parent)

New X

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge ____

Comprehension / Analysis / **H**igher Cognitive X

10 CFR Part 55 Content: 55.41 (b) (10)

55.43 _____

Comments:

The question meets Tier 1 criteria as it tests information contained in the site's procedures, specifically the Abnormal Operating Procedures.

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Examination Outline Cross-Reference: Q06

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	000025 AK3.01	
Importance Rating	3.8	

K/A Statement: Knowledge of the reasons for the following responses and/or actions as they apply to LOSS OF RESIDUAL HEAT REMOVAL SYSTEM: Shift to alternate flow path

Proposed Question: Q6

6. The Unit is in MODE 5 after completion of a refueling outage.

Current plant conditions:
- Decay Heat (DH) Pump 1 is running in decay heat removal mode
- DH Pump 2 is out of service due to a motor fault
- Fuel handling operations are taking place in the Spent Fuel Pool
- A Steam bubble has been drawn in the PZR
- RCS Temperature is 130ºF
- Pressurizer Level is 75 inches
- RCPs are not running

Subsequently, DH14B, DH Cooler 1 Outlet Flow Control Valve, fails Closed.
- Annunciator (3-1-H) LP INJ 1 FLOW LO alarms
- The Zone 3 operator reports DH14B cannot be manually opened

To restore core cooling, the operating crew MUST _____ per DB-OP-02527, LOSS OF DECAY HEAT REMOVAL.

A. throttle OPEN DH13B, DH CLR 1 BYPASS

B. place DH Pump 1 in service on LPI Injection line 2

C. align spent fuel pool cooling to the reactor vessel

D. establish feed and bleed using Gravity Drain of the BWST

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Proposed Answer: **B**

Explanation: Q6

A. Incorrect – Plausible since DH13B and DH14B are used in conjunction to set flow and the desired RCS temperature. Incorrect since opening DH13B (Cooler Bypass) will not provide core cooling.

B. Correct –DH Pump 1 is still available to use in this scenario. DH Pump 1 would be placed in service on LPI Injection line 2 IAW Attachment 1.

C. Incorrect – Plausible as this would be the correct action per Attachment 4 if the refueling gates were still removed and the plant was in Mode 6.

D. Incorrect – Plausible since feed and bleed cooling is presented as a possible option. However, gravity draining from the BWST is not possible with these plant conditions (RCS Pressurized).

Technical Reference(s): DB-OP-02527 R22, Loss of Decay Heat Removal (p20) Loss of flow path then RNO to loss of pump.

Reference(s) Provided: None

Learning Objective: OPS-GOP-127-03K

Question Source:
- Bank # _____
- Modified Bank # _____ (note changes or attach parent)
- New X

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level:
- Memory / **F**undamental Knowledge ____
- Comprehension / Analysis / **H**igher Cognitive X

10 CFR Part 55 Content:
- 55.41 (b) (5) & (10)
- 55.43 _____

Comments:

The question meets Tier 1 criteria as it tests information contained in the site's procedures, specifically the Abnormal Operating Procedures.

Examination Outline Cross-Reference: Q07

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	000026 AK1.02	
Importance Rating	3.8	

K/A Statement: Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to LOSS OF COMPONENT COOLING WATER: Loss of cooling to the CCWS

Proposed Question: Q7

7. Initial plant conditions:
   - Component Cooling Water (CCW) Pump 2 is in service
   - Component Cooling Water (CCW) Pump 1 is in standby
   - Service Water (SW) Pumps 1 and 2 are in service

   Subsequently:
   - 11-2-B, CCW HX 2 OUTLET TEMP HI is LIT
   - SW1434, COMP COOLING WATER HX 2 Outlet is observed to be closed
   - I&C reports the temperature transmitter to TIC1434 has failed sending a continuous close signal to SW1434

   Subsequently, a plant transient occurs resulting in an SFAS Level 2 Actuation.

   Regarding the Service Water System response, cooling to CCW HX2 will _____.

   A. remain isolated due to the TIC 1434 failure

   B. be maximized due to the SFAS Level 2

   C. control temperature at the TCV setpoint

   D. remain isolated until SW1434 is manually RESET

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Proposed Answer: **B**

Explanation: Q7

A. Incorrect – Plausible since there is a continuous close signal applied to the temperature controller.

B. Correct – SFAS 2 vents air from SW1434 controls thus overriding the temperature controller signal. SW1434 fails open to its SFAS position on loss of air.

C. Incorrect – Plausible since it is a temperature control valve and various components have setpoints related to safety signals.

D. Incorrect – Plausible since this is the condition without the SFAS signal.

Technical Reference(s): OS-020 SH1 R104, OS-020 SH2 R54

Reference(s) Provided: None

Learning Objective: OPS-SYS-304-03K

Question Source: Bank # X

Modified Bank # ____ (note changes or attach parent)

New ____

Question History: Last NRC Exam 2016 Q6 (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge X

Comprehension / Analysis / **H**igher Cognitive ____

10 CFR Part 55 Content: 55.41 (b) (5) & (10)

55.43 ____

Comments:
The question meets Tier 1 criteria as it tests the applicant's knowledge of the plant's response to the progression of an emergency event.

Examination Outline Cross-Reference: Q08

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	000027 G2.2.22	
Importance Rating	4.0	

K/A Statement: Pressurizer Pressure Control System Malfunction (G2.2.22) EQUIPMENT CONTROL: Knowledge of limiting conditions for operation and safety limits

Proposed Question: Q8

8. The reactor was operating at 100% RTP with the RCS at NOP and NOT.

Pressurizer level is at 220 inches when Pressurizer Spray Valve, RC2 failed OPEN.

The crew entered DB-OP-02513, PRESSURIZER SYSTEM ABNORMAL OPERATION and closed Pressurizer Spray Block Valve, RC10.

No change in TAVE or Pressurizer level was observed.

RCS pressure is now 2035 psig.

Based on these conditions, which of the following is NOT met?

A. LCO 3.4.1, RCS Pressure, Temperature, and Flow DNB Limits

B. LCO 3.4.11, Pressurizer Pilot Operated Relief Valve (PORV)

C. TNC 8.4.4 Reactor Coolant System Vents

D. LCO 3.4.9, Pressurizer

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Proposed Answer: **A**

Explanation: Q8

A. Correct – When pressure decreased below 2064.8 psig with all four reactor coolant pumps running (implied since reactor power is 100% RTP), LCO 3.4.1 was no longer met. RCS pressure ≥ 2064.8 psig with four RCPs running is "above the line" knowledge and is therefore expected to be recalled by RO candidates.

B. Incorrect – Plausible because isolating the PORV is a step in the PZR Abnormal similar to isolating the spray valve. Isolating a PORV leads to entering TS 3.4.11.

C. Incorrect – Plausible because isolating the PORV is a step in the PZR Abnormal similar to isolating the spray valve. Closing the PORV Block valve and then opening it's breaker would require entry into TNC 8.4.4 c.

D. Incorrect – Plausible since this LCO addresses pressurizer parameters and pressure control devices, specifically level which is met through the transient and heaters which are not affected by this transient.

Technical Reference(s): DB-OP-02513 R14, PRESSURIZER SYSTEM ABNORMAL OPERATION (p26), ITS R340 LCO 3.4.1 (p3.4.1-1)

LCO 3.4.3, LCO 3.4.9, TNC 8.4.2

Reference(s) Provided: None

Learning Objective: OPS-GOP-434-02K

Question Source: Bank # _____

Modified Bank # _____ (note changes or attach parent)

New X

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge X

Comprehension / Analysis / **H**igher Cognitive _____

10 CFR Part 55 Content: 55.41 (b) (5)

55.43 _____

Comments:

The question meets Tier 1 criteria as it tests information contained in the site's procedures, specifically license documents.

Examination Outline Cross-Reference: Q09

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	000029 EK2.17	
Importance Rating	4.4	

K/A Statement: Knowledge of the relationship between ANTICIPATED TRANSIENT WITHOUT SCRAM (ATWS) and the following systems or components: Diverse scram system (BW)

Proposed Question: Q9

9. The reactor is operating at 100% RTP when a main turbine trip occurs.

The control rod drive mechanisms (CRDMs) remain energized.

The ATC RO has depressed both Reactor Trip Pushbuttons IAW DB-OP-02000, RPS, SFAS, SFRCS TRIP, OR SG TUBE RUPTURE without success.

The Diverse Scram System will deenergize the CRDMs when_____(1)_____ with _____(2)_____ logic.

A. (1) Loop T_{HOT} is > 625ºF
(2) 2 out of 2

B. (1) RCS Extended Wide Range Pressure is > 2450 psig
(2) 2 out of 2

C. (1) Loop T_{HOT} is > 625ºF
(2) 1 out of 2

D. (1) RCS Extended Wide Range Pressure is > 2450 psig
(2) 1 out of 2

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Proposed Answer: **B**

Explanation: Q9

A. Incorrect – Plausible as there is an RPS Trip signal when Loop T_{HOT} exceeds approximately 618F. 2 out of 2 logic is correct.

B. Correct – The Diverse Scram System will cause the degating of the CRDM Silicon Controlled Rectifiers and therefore the deenergization of the CRDM when both channels of PAM RCS Extended Range Pressure exceed 2450 psig.

C. Incorrect – Plausible as there is an RPS Trip signal when Loop T_{HOT} exceeds approximately 618F. 1 out of 2 logic is used in other applications.

D. Incorrect – Plausible as the Extended Range Pressure instruments provide the trip signal for the Diverse Scram System. 1 out of 2 logic is used in other applications.

Technical Reference(s):	SD 052 R4, System Description for Diverse Scram System (p1.2.1)		
Reference(s) Provided:	None		
Learning Objective:	OPS-SYS-510-02K		
Question Source:	Bank #	____	
	Modified Bank #	____	(note changes or attach parent)
	New	X	
Question History:	Last NRC Exam	_____	(if applicable)
Question Cognitive Level:	Memory / **F**undamental Knowledge		X
	Comprehension / Analysis / **H**igher Cognitive		____
10 CFR Part 55 Content:	55.41	(b) (8) & (10)	
	55.43	_____	

Comments:

The question meets Tier 1 criteria as it tests the applicant's knowledge of the plant's response to the progression of an emergency event.

Examination Outline Cross-Reference: Q10

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	000038 EK1.03	
Importance Rating	3.6	

K/A Statement: Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to STEAM GENERATOR TUBE RUPTURE: Natural circulation

Proposed Question: Q10

10. The following events have occurred:
    - Loss of Offsite Power (LOOP)
    - Steam Generator Tube Rupture

    Current plant conditions:
    - RCS pressure is 1500 psig
    - T_{HOT} is 550°F

    With these conditions which of the following describes the cooldown rate and target temperature per DB-OP-02000, RPS, SFAS, SFRCS TRIP, OR STEAM GENERATOR TUBE RUPTURE, Section 8, Steam Generator Tube Rupture?

    A. 50°F/hr to a T_{HOT} of 520°F

    B. 50°F/hr to a T_{HOT} of 500°F

    C. 100°F/hr to a T_{HOT} of 500°F

    D. 100°F/hr to a T_{HOT} of 520°F

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Proposed Answer: **A**

Explanation: Q10

A. Correct – Per step 8.31 of the Steam Generator Tube Rupture section of DB-OP-02000 cooldown to 520°F at 50°F per hour with natural circulation. A loss of offsite power causes a loss of forced circulation.

B. Incorrect – Plausible since 50°F per hour is correct for natural circulation and 500°F would be the correct target for forced circulation.

C. Incorrect – Plausible since 100°F per hour and 500°F is correct for forced circulation.

D. Incorrect – Plausible since 100°F per hour is correct for forced circulation and 520°F is correct for natural circulation system.

Technical Reference(s): DB-OP-02000 R32, RPS, SFAS, SFRCS TRIP, OR SG TUBE RUPTURE (Step 8.31, p134)

Reference(s) Provided: None

Learning Objective: OPS-GOP-307-05K

Question Source: Bank # X

Modified Bank # ___ (note changes or attach parent)

New ___

Question History: Last NRC Exam 2016 Q9 (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge ___

Comprehension / Analysis / **H**igher Cognitive X

10 CFR Part 55 Content: 55.41 (b) (5) & (7)

55.43 ___

Comments:

The question meets Tier 1 criteria as it tests information contained in the site's procedures, specifically the Emergency Operating Procedures.

Examination Outline Cross-Reference: Q11

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	000040 AA1.05	
Importance Rating	4.2	

K/A Statement: Ability to operate and/or monitor the following as they apply to STEAM LINE RUPTURE: Manual and automatic RPS trip initiation

Proposed Question: Q11

11. The plant is operating at 100% RTP when the following indications are observed in the Control Room:
    - Reactor power rising
    - T_{AVE} lowering
    - RCS Pressure lowering
    - MU Flow rising
    - Generated MWs begin to lower
    - CTMT pressure at 16.6 psia and rising
    - Containment radiation remains constant

    The crew verifies NO personnel are in CTMT.

    What operator actions MUST be taken NEXT?

    A. Immediately Trip the Reactor, ONLY then GO TO DB-OP-02000, RPS, SFAS, SFRCS Trip, OR SG TUBE RUPTURE.

    B. Immediately Trip the Reactor AND Initiate AND Isolate SFRCS using MANUAL ACTUATION Switches.

    C. Commence a Rapid Plant Shutdown until SGs are on Low Level Limit Control, then Trip the Reactor AND Initiate AND Isolate SFRCS using MANUAL ACTUATION Switches.

    D. Maintain Reactor power less than or equal to 100% RTP, THEN assess the plant conditions for leak isolation and continued operations.

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Proposed Answer: **C**

Explanation: Q11

A. Incorrect – Plausible as this is an immediate operator action based on other plant conditions, but operators are also directed to INITIATE and ISOLATE SFRCS to isolate a steam leak.

B. Incorrect – Plausible as the initial conditions provided in the question stem are the symptoms of a steam leak in containment per DB-OP-02525, STEAM LEAKS. The immediate actions of the procedure direct operators to Trip the Reactor and to INITIATE and ISOLATE SFRCS to place the plant in a safe condition and to attempt to isolate the Steam line fault <u>if personnel are in containment</u>.

C. Correct – Per the supplemental actions of DB-OP-02525 for a steam leak in containment, operators are directed to perform a plant shutdown to LLLs then Trip the Reactor, I&I SFRCS, then GOTO DB-OP-02000.

D. Incorrect – Plausible since the supplemental actions of DB-OP-02525 for a steam leak in containment, operators are initially directed to reduce reactor power to keep it below 100% RTP. Incorrect since the next step directs a plant shutdown, power operations will not be maintained.

Technical Reference(s): DB-OP-02525 R14, STEAM LEAKS. (p10-14)

DB-OP-02000, RPS, SFAS, SFRCS TRIP, OR SG TUBE RUPTURE: Davis-Besse Updated Final Safety Analysis Report, Section 15.4.4 Steam Line Break

Reference(s) Provided: None

Learning Objective: OPS-GOP-125-03K

Question Source: Bank # ___

Modified Bank # ___ (note changes or attach parent)

New X

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge _____

Comprehension / Analysis / **H**igher Cognitive X

10 CFR Part 55 Content: 55.41 (b) (5) & (7)

55.43 _____

Comments:

This question meets Tier 1 criteria as it tests the applicant's knowledge of actions in the Abnormal and Emergency Operating Procedures.

Examination Outline Cross-Reference: Q12

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	000054 AA2.05	
Importance Rating	3.0	

K/A Statement: Ability to determine and/or interpret the following as they apply to LOSS OF MAIN FEEDWATER: Status of MFW pumps, regulating and stop valves

Proposed Question: Q12

12. The reactor was at 100% RTP, when both MFW pumps trip.

Subsequently,
- ARTS initiated a Reactor Trip
- Both AFW Pumps initially start
- AFW Pump 1 then trips on overspeed
- Steam Generator 1 level is 22” and lowering

Based on these conditions, FW 612 and FW 601, SG 1 & 2 MAIN FEEDWATER STOP valves are _____(1)_____.

IAW DB-OP-02000, Specific Rule 4 the crew MUST start the _____(2)_____ and feed Steam Generator 1.

A. (1) open
(2) Emergency Feedwater pump

B. (1) open
(2) Motor Driven Feedwater pump

C. (1) closed
(2) Emergency Feedwater pump

D. (1) closed
(2) Motor Driven Feedwater pump

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Proposed Answer: **D**

Explanation: Q12

A. Incorrect – Plausible as OTSG levels < 23.5” would cause an SFRCS initiation but not an isolation. In this case the trip of both MFW pumps results in an SFRCS initiation AND isolation on reverse D/P across the feedwater check valves, therefore closing the MFW Isolation Valves. The second part is plausible as this would be the required action if both AFW pumps tripped and the MDFP did not start.

B. Incorrect – Plausible as OTSG levels < 23.5” would cause an SFRCS initiation but not an isolation. In this case the trip of both MFW pumps results in an SFRCS initiation AND isolation on reverse D/P across the feedwater check valves, therefore closing the MFW Isolation Valves. The second part is correct.

C. Incorrect – Plausible as the first part is correct for an SFRCS initiation and isolation. The second part is plausible as this would be the required action if both AFW pumps tripped and the MDFP did not start.

D. Correct – An SFRCS initiation and isolation occurs on reverse D/P when both MFW pumps trip. This results in a closing signal to the MFW STOP valves. With one AFW pump tripped and one running, SR4 directs the crew to start the MDFP and feed the affected SG.

Technical Reference(s): DB-OP-02000 R32, RPS, SFAS, SFRCS TRIP, OR SG TUBE RUPTURE (p285 SR4 & p497 Table 1). SD-010, SFRCS

Reference(s) Provided: None

Learning Objective: OPS-GOP-301-04K

Question Source: Bank # _____

Modified Bank # _____ (note changes or attach parent)

New X

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge _____

Comprehension / Analysis / **H**igher Cognitive X

10 CFR Part 55 Content: 55.41 (b) (10)

55.43 _____

Comments:

This question meets Tier 1 criteria as it tests the applicant’s knowledge of actions found in the Emergency Operating Procedures.

Examination Outline Cross-Reference: Q13

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	000055 EK1.07	
Importance Rating	4.0	

K/A Statement: Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to STATION BLACKOUT: RCP seal leakage and inventory control

Proposed Question: Q13

13. A large tornado impacted the site resulting in an Extended Loss of AC Power (ELAP).

There is NO indication of an RCS Leak.

To mitigate the loss of RCS inventory during an ELAP, DB-OP-02700, Station Blackout requires a Time Sensitive Operator Action for the crew to _____.

A. verify letdown is aligned to the Makeup Tank

B. isolate the Pressurizer PORV by closing the PORV Block Valve

C. CLOSE MU38 to isolate RCP Seal Return

D. CLOSE MU19 to isolate RCP Seal Injection

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Proposed Answer: **C**

Explanation: Q13

A. Incorrect – Letdown is isolated by closing MU3 (AOV). Plausible since this action maintains RCS Inventory when the MUPs are in service.

B. Incorrect – Plausible as this action is required by DB-OP-02000 and the Abnormal Operating Procedures if the Pressurizer PORV fails open. RC11 is powered by E16B which has been lost due to the station blackout.

C. Correct – DB-OP-02700, requires a 10-minute Time Sensitive Operator Action to minimize loss of RCS inventory by closing MU38 RCP SEAL RETURN.

D. Incorrect – Plausible as this is an action required in DB-OP-02522, SMALL RCS LEAKS ATTACHMENT 6, Isolation of Leaks in the Makeup System. Isolating MU19 will not prevent a loss of RCS inventory in this scenario.

Technical Reference(s): DB-OP-02700 R2, STATION BLACKOUT (p4)

DB-OP-02000, RPS, SFAS, SFRCS TRIP, OR SG TUBE RUPTURE; DB-OP-02522, SMALL RCS LEAKS.

Reference(s) Provided: None

Learning Objective: OPS-FLX-003

Question Source: Bank # _____

Modified Bank # _____ (note changes or attach parent)

New X

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge X

Comprehension / Analysis / **H**igher Cognitive _____

10 CFR Part 55 Content: 55.41 (b) (5) & (7)

55.43 _____

Comments:

This question meets Tier 1 criteria as it tests the applicant's knowledge of actions found in the Emergency and Abnormal Operating Procedures.

Examination Outline Cross-Reference: Q14

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	000057 AA1.05	
Importance Rating	3.7	

K/A Statement: Ability to operate and/or monitor the following as they apply to LOSS OF VITAL AC ELECTRICAL INSTRUMENT BUS: Backup instrument indications

Proposed Question: Q14

14. Initial conditions:
    - 100% RTP

    The following occurs:
    - Loss of power to 120 VAC Instrument Bus Y3

    With these current plant conditions which of the following combinations of Nuclear Instruments WILL be used to monitor reactor power?

    A. (1) NI3 Intermediate Range
       (2) NI7 Power Range

    B. (1) NI3 Intermediate Range
       (2) NI8 Power Range

    C. (1) NI4 Intermediate Range
       (2) NI7 Power Range

    D. (1) NI4 Intermediate Range
       (2) NI8 Power Range

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Proposed Answer: **A**

Explanation: Q14

A. Correct – Y3 powers RPS Cabinet 3 which powers NI 8 and NI 4 therefore NI 7 and NI 3 (powered from Y4) will be available to monitor reactor power.

B. Incorrect – NI8 will be de-energized. Plausible since NI3 is still energized.

C. Incorrect – NI4 will be de-energized. Plausible since NI7 is still energized.

D. Incorrect – Plausible if the examinee believes there is a backup power supply for Nuclear Instruments if the normal supply is de-energized. Y3 powers RPS Cabinet 3 which powers NI 8 and NI 4.

Technical Reference(s): DB-OP-06319, INSTRUMENT AC SYSTEM PROCEDURE

Reference(s) Provided: None

Learning Objective:

Question Source: Bank # _____

Modified Bank # X (note changes or attach parent)

New _____

Question History: Last NRC Exam DB 2020 Q57 (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge X

Comprehension / Analysis / **H**igher Cognitive _____

10 CFR Part 55 Content: 55.41 (b)(5) & (7)

55.43 _____

Comments:
This question was significantly modified as the question stem was changed in a way that made a distractor the new correct answer.

This question meets Tier 1 criteria as it tests the applicant's knowledge of actions found in the Abnormal Operating Procedures.

Examination Outline Cross-Reference: Q15

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	000058 AK2.07	
Importance Rating	3.8	

K/A Statement: Knowledge of the relationship between LOSS OF DC POWER and the following systems or components: AFW

Proposed Question: Q15

15. The Reactor is operating at 95% RTP due to the #2 Condensate Pump being OOS for maintenance.

Subsequently, the following annunciators are LIT:
- 1-5-F, DC PANEL VOLTAGE LO
- 1-6-A, INV YV1-YV3 TRBL
- 1-6-D, C1/D1 CONTROL PWR TRBL

The crew enters DB-OP-02537, LOSS OF D1P AND DAP and is required to Trip the Reactor and INITIATE and ISOLATE SFRCS.

Steam Generator 1 level will be maintained by placing _____.

A. Loop 1 Main and Startup FW Valves in AUTO

B. Loop 1 Main and Startup FW Valves in HAND

C. AF3870, AUX FEED 1 TO SG1 STOP in CLOSED with the MDFP in service

D. AF6452, SG1 AFW LEVEL CONTROL VALVE in AUTO supplied by AFPT 1

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Proposed Answer: **C**

Explanation: Q15

A. Incorrect – Plausible as this would be true if condensate pump #2 was not OOS.

B. Incorrect – Plausible if the operators confuse that MFW instead of AFW flow control would be affected by the loss in DC.

C. Correct – A loss of D1P and DAP results in a loss of the #1 and #3 condensate pumps. With #2 OOS, DB-OP-02537 Supplemental Actions direct the crew to trip the Reactor and Initiate and Isolate SFCRS. With AFW now running, this loss in DC power causes AF6452 to fail open and an overfeed condition on SG1 will occur. AF3870, AUX FEED 1 TO SG1 STOP can be closed since D1PA is available in this scenario. The MDFP will then be placed in service per Attachment 5.

D. Incorrect – Plausible as this is the normal alignment if there was not a loss of D1P and SFRCS was actuated.

Technical Reference(s):	DB-OP-02537 R20, LOSS OF D1P AND DAP (p3, p13) DB-OP-02707, LOSS OF DC POWER DB-OP-02000, DB-OP-02000, RPS, SFAS, SFRCS TRIP, OR SG TUBE RUPTURE		
Reference(s) Provided:	None		
Learning Objective:	OPS-GOP-137-4K		
Question Source:	Bank #		
	Modified Bank #		(note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		(if applicable)
Question Cognitive Level:	Memory / **F**undamental Knowledge		
	Comprehension / Analysis / **H**igher Cognitive	X	
10 CFR Part 55 Content:	55.41	(b)(8) & (10)	
	55.43		

Comments:

This question meets Tier 1 criteria as it tests the applicant's knowledge of actions found in the Emergency and Abnormal Operating Procedures.

Examination Outline Cross-Reference: Q16

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	000062 AA2.07	
Importance Rating	3.0	

K/A Statement: Ability to determine and/or interpret the following as they apply to LOSS OF SERVICE WATER: Implementation of TS requirements for loss of service water

Proposed Question: Q16

16. The Reactor is operating at 75% RTP and in power ascension following a refueling outage.
    - Service Water (SW) Pumps 1 and 2 are in service
    - SW Pump 2 is supplying the secondary loads
    - SW Pump 3 is available for operation

    Subsequently, the following annunciators are LIT and computer points in alarm:
    - 11-1-C, SW PMP 1 STRNR DISCH PRESS LO
    - 11-1-B, CCW HX 1 OUTLET TEMP HI
    - X002, SW PMP MTR TRBL
    - P945, SW HDR 1 PRESS

    The crew enters DB-OP-02511, LOSS OF SERVICE WATER PUMPS/SYSTEMS and determines that LCO 3.7.8, SERVICE WATER SYSTEM is not met.

    The crew MUST Align SW Pump 3 as 1 powered from ____(1)____.
    LCO 3.7.8, SERVICE WATER SYSTEM will be met when ____(2)____.

    A. (1) Bus D1
       (2) SW Pump 3 is in service as 1

    B. (1) Bus C1
       (2) SW Pump 3 is in service as 1

    C. (1) Bus D1
       (2) SW Pump 1 is repaired

    D. (1) Bus C1
       (2) SW Pump 1 is repaired

Proposed Answer: **B**

Explanation: Q16

A. Incorrect – Plausible because part (1) would be correct if SW Pump 2 had failed, and part (2) is correct.

B. Correct – This is the correct supplemental action per DB-OP-02511. LCO 3.7.8 Bases indicates that the spare SW Pump can be substituted for the normal loop SW pump as long as it is powered from the same bus as the pump it is replacing.

C. Incorrect –Plausible because part (1) would be correct if SW Pump 2 failed. Part (2) is plausible since SW Pump 1 is still damaged however, the LCO is permitted to be exited once SW Pump 3 is aligned to SW Loop 1.

D. Incorrect – Plausible as this is the correct Bus to power SW Pump 3 from, but the LCO is permitted to be exited once SW Pump 3 is aligned to loop 1 and powered from C1.

Technical Reference(s): DB-OP-02511 R22, LOSS OF SERVICE WATER PUMPS/SYSTEMS (p22, P65), TS Bases R34 (p B 3.7.8-2)

ITS LCO 3.7.8, Service Water Systems

Reference(s) Provided: None

Learning Objective: OPS-GOP-111-2K

Question Source: Bank # _____

Modified Bank # _____ (note changes or attach parent)

New X

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge _____

Comprehension / Analysis / **H**igher Cognitive X

10 CFR Part 55 Content: 55.41 (b)(10)

55.43 _____

Comments:

This question meets Tier 1 criteria as it tests the applicant's knowledge of actions found in the Abnormal Operating Procedures.

Examination Outline Cross-Reference: Q17

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	000065 AK3.14	
Importance Rating	4.2	

K/A Statement: Knowledge of the reasons for the following responses and/or actions as they apply to LOSS OF INSTRUMENT AIR: When to trip reactor if instrument air pressure is decreasing

Proposed Question: Q17

17. Initial conditions:
    - The Reactor is at 100% RTP
    - Station Air Compressor 2 is OOS for maintenance

    Subsequently, the following occurs:
    - 9-3-E, STA AIR HDR PRESS LO is LIT
    - 9-1-F, INSTR AIR HDR PRESS LO is LIT
    - PI 810, INSTRUMENT AIR HEADER PRESS indicates 85 psig and slowly lowering
    - PI 811, STATION AIR HEADER PRESS indicates 85 psig and slowly lowering
    - An Operator reports an air leak in Station Air Receiver 1
    - The plant is otherwise stable

    Per DB-OP-02528, INSTRUMENT AIR SYSTEM MALFUNCTIONS which of the following actions will mitigate this condition?

    A. Dispatch an EO to VERIFY SA6445, SA/IA CROSSTIE SOLENOID and SA236, IA/SA CROSSTIE SOLENOID (SA6445) BYPASS, are closed.

    B. Dispatch an EO to CLOSE SA18, STATION AIR HEADER BACK PRESSURE REG VALVE (SA2008) INLET ISOLATION VALVE.

    C. Notify Maintenance to immediately return Station Air Compressor 2 to service and verify all Air Compressors are running.

    D. IMMEDIATELY Trip the Reactor AND Initiate and Isolate SFRCS. Enter DB-OP-02000 RPS, SFAS, SFRCS TRIP, OR SG TUBE RUPTURE.

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Proposed Answer: **A**

Explanation: Q17

A. Correct – The EIAC will auto start. SA6445 and SA236 are verified closed by procedure to separate the rupture air receiver from Instrument air and allow Instrument air pressure to return to normal.

B. Incorrect – Plausible because this is a directed action for Instrument and Station air continuing to lower. This assumes the leak is in the instrument air header and also SA2008 is downstream of the station air receiver so this would not isolate the leak.

C. Incorrect – Plausible since verifying all available Air Compressors are running is directed but SAC 2 would not be considered available if it was out of service and the procedure does not direct a return to service.

D. Incorrect – Plausible since this is the direction for secondary plant upsets or when instrument air pressure lowers to 75 psig.

Technical Reference(s): DB-OP-02528 R28, INSTRUMENT AIR SYSTEM MALFUNCTIONS (p10)

Reference(s) Provided: None

Learning Objective: OPS-GOP-128-03K

Question Source: Bank # X

Modified Bank # _____ (note changes or attach parent)

New _____

Question History: Last NRC Exam 2016 Q14 (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge _____

Comprehension / Analysis / **H**igher Cognitive X

10 CFR Part 55 Content: 55.41 (b)(5) & (10)

55.43 _____

Comments:

This question meets Tier 1 criteria as it tests the applicant's knowledge of actions found in the Abnormal Operating Procedures.

Examination Outline Cross-Reference: Q18

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	BW E04 G2.1.30	
Importance Rating	4.4	

K/A Statement: Inadequate Heat Transfer – Loss of Secondary Heat Sink CONDUCT OF OPERATIONS: Ability to locate and operate components, including local controls

Proposed Question: Q18

18. The Reactor is operating at 100% RTP.
    - The MDFP is OOS for planned maintenance

    An event occurs which causes the following conditions:
    - Rising Containment temperatures and pressures without rising radiation levels
    - SFRCS has actuated due to low pressure in SG1
    - AF3869, Auxiliary Feed Pump 1 to Steam Generator 2 Stop, failed closed due to a stem/disc failure

    Subsequently, AFW Pump 2 trips and is damaged.
    - Steam Generator 2 level is 15 inches and lowering
    - Reactor pressure is 2000 psig and rising
    - RCS temperature is 540°F and rising slowly
    - The STBY MUP has been started, BOTH MUPs are in service

    What actions per DB-OP-02000, RPS, SFAS, SFRCS TRIP, OR SG TUBE RUPTURE, MUST operators take to recover heat transfer prior to RCS temperature reaching 600°F?

    A. Lock open RC2A, PORV until feedwater is restored IAW Attachment 4, Initiate MU/HPI Cooling.

    B. Open RC2A, PORV until RCS Subcooling Margin is 20°F IAW Section 6.0 Lack of Heat Transfer.

    C. Locally align then start the EFW pump to supply SG2 IAW Attachment 5, Guidelines for Restoring Feedwater.

    D. Place AVVs in HAND per ATTACHMENT 3: OPERATION OF AVV's and lower SG2 pressure until secondary T_{SAT} is 50°F lower than incore thermocouple temperature.

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Proposed Answer: **C**

Explanation: Q18

A. Incorrect – Plausible as this would be the required action if SG heat transfer is not available and RCS temperature exceeded 600°F.

B. Incorrect – Plausible as this would be the required action if while attempting to restore SG heat transfer, RCS pressure exceeded 2450 psig. The purpose of this step is to prevent repeated cycling of the PORV if the 2450 psig setpoint is reached. Repeated cycling may lead to failure.

C. Correct – With the MDFP OOS and AFW not able to feed SG2, operators are directed to restore FW with Attachment 5. Section B specifically directs operators to locally operate EFW and AFW valves to align the EFW pump to SG2.

D. Incorrect – Plausible as this would be the required action if SG2 had proper level but no heat transfer. The level in SG2 is < 16 inches which is not adequate/proper level to carry out this action.

Technical Reference(s): DB-OP-02000 R32, RPS, SFAS, SFRCS TRIP, OR SG TUBE RUPTURE (p286, 338)

Reference(s) Provided: None

Learning Objective: OPS-GOP-305-03K

Question Source: Bank # _____

Modified Bank # _____ (note changes or attach parent)

New X

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge _____

Comprehension / Analysis / **H**igher Cognitive X

10 CFR Part 55 Content: 55.41 (b)(7)

55.43 _____

Comments:

This question meets Tier 1 criteria as it tests the applicant's knowledge of actions found in the Emergency Operating Procedures.

Examination Outline Cross-Reference: Q19

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	000003 AK1.04	
Importance Rating	4.0	

K/A Statement: Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to DROPPED CONTROL ROD: Effect on power level and/or flux

Proposed Question: Q19

19. Initial plant conditions:
    - The plant is operating at 55% RTP
    - 3 RCP operation is in progress

    Subsequently, the following symptoms are observed:
    - 5-1-E, CRD SYSTEM FAULT is in alarm
    - 5-2-E, CRD ASYMMETRIC ROD is in alarm
    - Computer Point Q178 CRD ASYMMETRIC ROD ALARM is in alarm
    - CRD Mechanism Position Indication Panel - 0 PERCENT light for Rod 6 – 5 is LIT
    - IN LIMIT on rod control panel is lit for Group 6

    Based on the above plant conditions AND assuming no Cross Limits are reached:

    PRIOR TO the required actions of the crew, if plant conditions are permitted to stabilize, Heat Balance Power will be ___(1)___ the initial Heat Balance Power.

    IAW DB-OP-02516, CRD MALFUNCTIONS, the ULD MAX LOAD LIMIT will be set to ___(2)___ RTP.

    A. (1) substantially less than
       (2) 50%

    B. (1) approximately the same as
       (2) 50%

    C. (1) substantially less than
       (2) 33%

    D. (1) approximately the same as
       (2) 33%

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Proposed Answer: **D**

Explanation: Q19

A. Incorrect – Part (1) is plausible as initially power decreases on a dropped rod. Part (2) is plausible as power will be reduced to 50% IAW DB-OP-02516 for a dropped rod when 4 RCPs are in operation.

B. Incorrect – Plausible as Part (1) is correct. See explanation for answer D. Part (2) is plausible as power will be reduced to 50% IAW DB-OP-02516 for a dropped rod when 4 RCPs are in operation.

C. Incorrect – Part (1) is plausible as initially power decreases on a dropped rod. Part (2) is correct as power will be reduced to 33% IAW DB-OP-02516 for a dropped rod when 3 RCPs are in operation.

D. Correct – Part (1) is correct since initially power decreases on a dropped rod, but as T_{AVE} lowers with a negative MTC the coolant becomes more dense and less neutrons escape the core and are absorbed by fuel thereby causing reactor power to raise back to approximately the same value as it was initially with a lower RCS temperature. Part (2) is correct as power will be reduced to 33% IAW DB-OP-02516 for a dropped rod when 3 RCPs are in operation.

Technical Reference(s):	Davis Besse UFSAR Chapter 15.2 Class 1 Events Leading to No Radioactive Release at the Exclusion Zone, Section 15.2.3.2.3; DB-OP-02516 R19, CRD MALFUNCTIONS (p12)		
Reference(s) Provided:	None		
Learning Objective:	OPS-GOP-116-03K		
Question Source:	Bank #		
	Modified Bank #	X	(note changes or attach parent)
	New		
Question History:	Last NRC Exam	2022 Q82	(if applicable)
Question Cognitive Level:	Memory / **F**undamental Knowledge		
	Comprehension / Analysis / **H**igher Cognitive	X	
10 CFR Part 55 Content:	55.41	(b)(5) & (7)	
	55.43		

Comments:

This question is significantly modified as the question stem was changed to ask about the response of reactor power and the answer, and all distractors were changed.

This question meets Tier 1 criteria as it tests the applicant's knowledge of actions found in the Abnormal Operating Procedures.

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Examination Outline Cross-Reference: Q20

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	000005 AA2.05	
Importance Rating	3.8	

K/A Statement: Ability to determine and/or interpret the following as they apply to INOPERABLE/STUCK CONTROL ROD: Ex-core NIS

Proposed Question: Q20

20. The plant is operating at 75% RTP.
    - Rod Index is at 265
    - A power ascension to 100% is started with Integrated Control System (ICS) in Full Automatic

    At 80% RTP, the following conditions are noted:
    - 5-2-E, CRD ASYMMETRIC ROD is LIT
    - ASYMMETRY FAULT light on the Rod Control Panel is LIT

    The crew stabilizes the plant at 80% and determines Control Rod 7-1 is actually STUCK at 65%.

    The Axial Power Imbalance signal produced by the Nuclear Instrument (NI) nearest Control Rod 7-1 will become ___(1)___ negative COMPARED to the other three Power Range NIs.

    IAW DB-OP-02516, CRD MALFUNCTIONS, 4.4 Stuck Control Rods the crew is required to reduce power ____(2)____.

A. (1) less
   (2) until rod alignment is restored

B. (1) more
   (2) until rod alignment is restored

C. (1) less
   (2) to MODE 3, continued operation with a known stuck rod is not permitted

D. (1) more
   (2) to MODE 3, continued operation with a known stuck rod is not permitted

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Proposed Answer: **D**

Explanation: Q20

A. Incorrect – Plausible if the applicant confuses how Axial Power Imbalance is calculated. Part (2) is plausible as this is the required action for a misaligned control rod from Control Rod Group 7 which is not stuck. The question stem states that Rod 7-1 is stuck.

B. Incorrect – Plausible as part (1) is correct and part (2) is plausible as this is the required action for a misaligned control rod from Control Rod Group 7 which is not stuck. The question stem states that Rod 7-1 is stuck.

C. Incorrect – Plausible if the applicant confuses how Axial Power Imbalance is calculated. Part (2) is correct.

D. Correct – With Control Rod 7-1 stuck at its original position. Flux as detected by the upper detector of the affected PRNI will be suppressed as compared to the other three channels. Per DB-OP-02516, CRD 7-1 must be declared inoperable, and a rapid shutdown must be performed to be in MODE 3 in 6 hours per TS 3.1.4, Cond D.

Technical Reference(s): DB-OP-02516 R19, CRD MALFUNCTIONS (p40); SD 044 R4, (p2-4)

Reference(s) Provided: None

Learning Objective: OPS-GOP-116-03K

Question Source: Bank # _____

Modified Bank # X (note changes or attach parent)

New _____

Question History: Last NRC Exam 2015 Q19 (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge _____

Comprehension / Analysis / **H**igher Cognitive X

10 CFR Part 55 Content: 55.41 (b)(10)

55.43 _____

Comments:

This question is significantly modified as the question stem was changed to ask about Axial Power Imbalance and the required actions per DB-OP-02516.

This question meets Tier 1 criteria as it tests the applicant's knowledge of actions found in the Abnormal Operating Procedures.

Examination Outline Cross-Reference: Q21

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	000028 G2.4.4	
Importance Rating	4.5	

K/A Statement: (SF2 PZR LCS) PRESSURIZER LEVEL CONTROL SYSTEM (G.2.4.4) EMERGENCY PROCEDURES/PLAN: Ability to recognize abnormal indication for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.

Proposed Question: Q21

21. The plant is operating at 100% RTP.

An event occurs which causes the following conditions:

- Tave is 582ºF and stable
- RCS pressure is 2155 psig and stable
- Pressurizer level is 220 inches and lowering
- MUT level is 74 inches and rising
- MU flow is 0 GPM
- LD flow is 70 GPM

The crew enters the appropriate Abnormal Procedure.

At which Pressurizer Level is the crew required to Trip the Reactor prior to reaching Steam Generator Low Level Limits?

A. 200 inches

B. 160 inches

C. 140 inches

D. 100 inches

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Proposed Answer: **B**

Explanation: Q21

A. Incorrect – Plausible if the applicant believes that the reactor must be tripped when PZR level reaches 20” below setpoint. There is an action to take in RCS Leaks when PZR level is 20 inches below setpoint. Annunciator 4-2-E, PZR LVL LO also comes in at 200 inches.

B. Correct – IAW DB-OP-02512, Step 4.1.3 (Loss of RCS Makeup Pump(s)) IF AT ANY TIME PZR level is less than the minimum required level in accordance with Curve CC 4.3, Minimum Pressurizer Level vs. RC Temperature of DB-PF-06703, Miscellaneous Operation Curves (minimum level for Tave 582°F is 160 inches), THEN Trip the Reactor.

C. Incorrect – Plausible since 140 inches is the limit if Tave was 573°F which is possible if the crew started a down power.

D. Incorrect – Plausible since this is the level at which the Reactor is required to be tripped for an RCS Leak. The applicant may believe there is a leak in progress since PZR level is lowering. MUT level does not rise during an RCS Leak.

Technical Reference(s): DB-OP-02512, MAKEUP AND PURIFICATION SYSTEM MALFUNCTIONS; SD 048, System Description for MAKEUP AND PURIFICATION, DB-PF-06703 MISCELLANEOUS OPEARTION CURVES, DB UFSAR

Reference(s) Provided: None

Learning Objective:

Question Source: Bank # _____

Modified Bank # _____ (note changes or attach parent)

New X

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge X

Comprehension / Analysis / **H**igher Cognitive ____

10 CFR Part 55 Content: 55.41 (b)(10)

55.43 _____

Comments:

This question meets Tier 1 criteria as it tests the applicant’s knowledge of actions found in the Abnormal Operating Procedures.

Examination Outline Cross-Reference: Q22

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	000033 G2.4.45	
Importance Rating	4.1	

K/A Statement: LOSS OF INTERMEDIATE RANGE NUCLEAR INSTRUMENTATION (G2.4.45) EMERGENCY PROCEDURES/PLAN: Ability to prioritize and interpret the significance of each annunciator or alarm.

Proposed Question: Q22

22. The plant is in Mode 2 during a planned reactor startup.
    - NI-3 indicates off scale low (< $1x10^{-11}$ AMPS)
    - NI-4 indicates normal at $2x10^{-10}$ AMPS
    - ROD INDEX is at 160

    The CSRO has just declared NI-3 INOPERABLE due to improper overlap.

    Prior to any other actions being taken, the following indications are observed:
    - 5-5-E, SUR ROD WITHDRAW INHIBIT is LIT
    - NI-3 still indicates < $1x10^{-11}$ AMPS
    - NI-4 is observed to be failed high

    The crew enters DB-OP-02505, NUCLEAR INSTRUMENTATION FAILURES.

    What action is REQUIRED by the control room IAW DB-OP-02505, NUCLEAR INSTRUMENTATION FAILURES?

    A. Deenergize the Source Range Detectors immediately.

    B. OPEN the CRD trip breakers within 1 hour.

    C. Suspend ALL reactivity changes until BOTH NI-3 and NI-4 are restored.

    D. Reduce power to < $1x10^{-10}$ AMPS within 10 hours.

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Proposed Answer: **B**

Explanation Q22

A. Incorrect – Plausible since this is the action if the IR detectors fail low and cause the SR detectors to energize. The detector failing high will not cause the SR detectors to inadvertently energize.

B. Correct – NI-3 INOPERABLE due to improper overlap. NI-4 is INOPERABLE because its failed high. Because 2 Intermediate range NI's are INOPERABLE, DB-OP-2505 NUCLEAR INSTRUMENT FAILURES step 4.2.1 directs the operator to suspend all positive reactivity addition manipulations AND OPEN the CRD breakers within 1 hr. DB ITS LCO 3.3.10 (1 hour action).

C. Incorrect – Plausible since suspending positive reactivity changes is a required action in this condition.

D. Incorrect – Plausible since reducing power to < 1x10-10 AMPS within 2 hours is an action for 1 IR detector inop, not 10 hours. 10 hour action to reduce High Flux trip setpoints is required for other plant conditions.

Technical Reference(s): DB-OP-02505, NUCLEAR INSTRUMENTATION FAILURES, SD 044 System Description for NUCLEAR INSTRUMENTATION/REACTOR PROTECTION SYSTEM, DB-OP-02005 PRIMARY INSTRUMENTATION ALARM PANEL 5 ANNUNCIATORS, DB UFSAR, DB ITS LCO 3.3.10

Reference(s) Provided: None

Learning Objective:)

Question Source: Bank # _____

Modified Bank # _____ (note changes or attach parent)

New X

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge _____

Comprehension / Analysis / **H**igher Cognitive X

10 CFR Part 55 Content: 55.41 (b)(10)

55.43 _____

Comments:

This question meets Tier 1 criteria as it tests the applicant's knowledge of actions found in the Abnormal Operating Procedures.

Examination Outline Cross-Reference: Q23

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	BW A01 AA1.08	
Importance Rating	3.5	

K/A Statement: Ability to operate and/or monitor the following as they apply to (BW A01) PLANT RUNBACK: ICS

Proposed Question: Q23

23. The plant is operating at 100% RTP with SG/RX DEMAND in HAND due to a malfunction in AUTO.

A transient occurs and the following is reported:

- 14-3-D, ICS MFP LOSS OR LO DEAR RUNBACK is LIT

The cause of this condition is ___(1)___.

With SG/RX DEMAND in HAND, the plant will ___(2)___ reduce power to 55%.

A. (1) Deaerator Storage Tank level of 3.5 feet
(2) automatically respond to

B. (1) Deaerator Storage Tank level of 3.5 feet
(2) require the operators to manually

C. (1) MFP 1 discharge pressure of 1400 psig
(2) automatically respond to

D. (1) MFP 1 discharge pressure of 1400 psig
(2) require the operators to manually

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Proposed Answer: **B**

Explanation: Q23

A. Incorrect – Plausible since during normal operations, a runback signal generated would cause ICS to automatically runback to 55% power; however, the stem of the question has the SG/RX DEMAND in HAND. Per DB-OP-2014, annunciator 14-3-D, the immediate action when any ICS H/A station is in HAND is to MANUALLY perform the runback. The first part of the question is correct.

B. Correct – Per DB-OP-02014, the condition that gives this alarm (and, thus a MFW runback) are 75 psig sensed by PS 2729 or 2730, LSLL 425 or 435 (Deaerator level) of 4 feet, or MFP discharge pressure of 1433 psig. Deaerator storage tank level presented in this question meets the runback condition. Additionally, since stem of the question has the SG/RX DEMAND in HAND. Per DB-OP-2014, annunciator 14-3-D, the immediate action when any ICS H/A station is in HAND is to MANUALLY perform the runback.

C. Incorrect – Plausible since multiple conditions can cause this annunciator (and a MFW runback), including MFP discharge pressure. If the candidate does not recall the setpoint, (1433 psig), this distractor may be selected. Additionally, since during normal operations, a runback signal generated would cause ICS to automatically runback to 60% power; however, the stem of the question has the SG/RX DEMAND in HAND. Per DB-OP-2014, annunciator 14-3-D, the immediate action when any ICS H/A station is in HAND is to MANUALLY perform the runback.

D. Incorrect – Plausible since multiple conditions can cause this annunciator (and a MFW runback), including MFP discharge pressure. If the candidate does not recall the setpoint, (1433 psig), this distractor may be selected. The second part is correct.

Technical Reference(s): DB-OP-02014, ARP for Annunciator 14-3-D:
IMMEDIATE ACTIONS - IF any ICS H\A Station is in HAND,
THEN manually perform the runback function on those stations.
REFER TO DB-OP-06401, Integrated Control System Operating Procedure

Proposed reference: None

Learning Objective:

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

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge _____
Comprehension / Analysis / **H**igher Cognitive X

10 CFR Part 55 Content: 55.41 (b)(7)
55.43 _____

Comments:

This question meets Tier 1 criteria as it tests the applicant's knowledge of actions found in the Normal Operating Procedures.

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KA Change

Examination Outline Cross-Reference: Q24

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	000037 AA2.03	
Importance Rating	4.0	

K/A Statement: Ability to determine and interpret the following as they apply to the Steam Generator Tube Leak: Verification that the expected indication on main steam lines from the S/Gs should show increasing radiation levels

Proposed Question: Q24

24. DB-OP-02012, STM GEN/SFRCS ALARM PANEL 12 ANNUNCIATOR procedure directs Radiation Protection to be notified to take local surveys of the Main Steam Line area when annunciator 12-1-A, MN STM LINE 1 RAD HI comes into alarm.

Which of the following is the reason for this direction?

A. To evaluate for initiating conditions into RA-EP-02861, RADIOLOGICAL INCIDENTS

B. To obtain data to support leak rate calculation for DB-OP-02522, SMALL RCS LEAKS

C. To project off site doses from the Station Vent in accordance with RA-EP-02240, OFFSITE DOSE ASSESSMENT

D. To verify affected SG diagnosis in accordance with DB-OP-02531, STEAM GENERATOR TUBE LEAK

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Proposed Answer: **D**

Explanation: Q24

A. Incorrect – Plausible because high radiation levels would be an initiating condition for the radiological incidents off normal procedure, but this alarm is to support indications of a steam generator tube leak.

B. Incorrect – Plausible because there are leak rate calculations in DB-OP-02522. None of these calculations use rad surveys.

C. Incorrect – Plausible Steam Generator Tube Leaks will cause a release of radioactive material but checking radiation levels in the Main Steam Line area will not allow determination of dose from the station vent.

D. Correct – DB-OP-02012 directs checking symptoms in accordance with DB-OP-02531 along with alarm verification to access entry conditions into the steam generator tube leak abnormal procedure.

Technical Reference(s): DB-OP-02012 12-1-A, DB-OP-02531 (p4)

Reference(s) Provided: None

Learning Objective:

Question Source: Bank # X

Modified Bank # ____ (note changes or attach parent)

New ____

Question History: Last NRC Exam DB2020 Q22 (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge X

Comprehension / Analysis / **H**igher Cognitive ____

10 CFR Part 55 Content: 55.41 (b)(8), (b)(10)

55.43 ____

Comments:

This question meets Tier 1 criteria as it tests the applicant's knowledge of the bases for actions found in the Abnormal Operating Procedures.

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Examination Outline Cross-Reference: Q25

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	BW E08 EK 3.12	
Importance Rating	4.0	

K/A Statement: Knowledge of the reasons for the following responses and/or actions as they apply to LOCA COOLDOWN: Check that LPI flow meets flow criteria to secure HPI

Proposed Question: Q25

25. The Reactor has tripped from 100% RTP due to a large RCS leak.
    - SFAS Level 4 has actuated; all safety systems responded as expected
    - The crew is preparing to transfer ECCS Pump Suctions to the Emergency Sump
    - LPI Injection flow has been 1500 gpm for > 20 minutes in BOTH LPI Lines (3000 gpm total LPI Injection Flow)

    HPI pumps ___(1)___ be stopped prior to the transfer.

    Piggyback Valves DH63 and DH64 are required to be ___(2)____.

    A. (1) CAN
       (2) OPEN

    B. (1) CAN
       (2) CLOSED

    C. (1) CANNOT
       (2) OPEN

    D. (1) CANNOT
       (2) CLOSED

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Proposed Answer: **B**

Explanation Q25

A. Incorrect –Incorrect. Part (1) is correct. HPI can be stopped if adequate LPI flow is GREATER THAN 1350 GPM for 20 minutes or more.
Part 2 is incorrect. Plausible since DH 63 and DH 64 are required to be open if HPI pumps remain in service.

B. Correct – Part (1) is correct. HPI can be stopped if adequate LPI flow is GREATER THAN 1350 GPM for 20 minutes or more.
Part (2) is correct. Piggyback valves DH63 and DH64 are required to be closed IAW SR3 and they are verified closed prior to the sump transfer.

C. Incorrect – Part (1) is incorrect. HPI should be TERMINATED if adequate LPI flow is GREATER THAN 1350 GPM for 20 minutes or more.
Part 2 is incorrect. Plausible since DH 63 and DH 64 are required to be open if HPI pumps remain in service.

D. Incorrect – Part (1) is incorrect. HPI should be TERMINATED if adequate LPI flow is GREATER THAN 1350 GPM for 20 minutes or more.
Part (2) is correct. Piggyback valves DH63 and DH64 are required to be closed IAW SR3 and they are verified closed prior to the sump transfer.

Technical Reference(s): DB-OP-02000 RPS, SFAS, SFRCS TRIP, OR SG TUBE RUPTURE SR3, 3.4 (close piggyback valves) and 3.5 (HPI termination criteria), DP-OP-02000 BASES AND DEVIATION DOCUMENT, DB UFSAR

Reference(s) Provided: None

Learning Objective:

Question Source: Bank # _____

Modified Bank # _____ (note changes or attach parent)

New X

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge ___

Comprehension / Analysis / **H**igher Cognitive X

10 CFR Part 55 Content: 55.41 (b)(5),(b)(10)

55.43 _____

Comments:

This question meets Tier 1 criteria as it tests the applicant's knowledge of actions found in the Emergency Operating Procedures.

Examination Outline Cross-Reference: Q26

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	BW E13 EK 1.15	
Importance Rating	3.3	

K/A Statement: Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to EOP RULES: Tube to shell delta T.

Proposed Question: Q26

26. The reactor has tripped from 100% RTP due to a Loss of Off-Site power.
    - AFW Pump 1 has tripped on overspeed
    - Due to equipment issues, Steam Generator (SG) 1 can NOT be fed by any other pumps
    - #2 Atmospheric Vent Valve is being utilized to maintain RCS temperature constant OR slightly lowering IAW SR 4, Steam Generator Control

    Subsequently, AFW Pump 1 is reset and is now available for use.
    - SG 1 level is 14"
    - SG 1 pressure is 750 psig and stable
    - SG 1 Shell avg. temperature: 534°F
    - RCS Loop 1 Tave: 516°F

    Based on current conditions, Steam Generator 1 _____.

    A. CAN be fed because SG Tube to Shell ΔT is ≤ 50°F

    B. CAN be fed because NO Loop 1 RCPs are in service

    C. CANNOT be fed because feeding of a Dry SG is prohibited with tubes in tension

    D. CANNOT be fed because feeding of a Dry SG is prohibited when at least one SG is providing heat transfer

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Proposed Answer: **A**

Explanation Q26

A. Correct - The given parameters indicate that the SG Tube to Shell ΔT is -18°F. RCS Loop 1 Tave: 489F - SG# 1 Shell avg temp: 534F = -18 ΔT. The step in DB-OP-2000 Specific Rule 4 Steam Generator Control 4.4 states to check that SG Tube to Shell ΔT meets the following: Less than or Equal to 50°F with tubes in TENSION.

B. Incorrect - First part is correct. Second part is incorrect. Plausible since RCPs do determine how much flow can be supplied to a dry SG NOT if you can feed it or not.

C. Incorrect – Per specific Rule 4 the SGs can be fed when tubes are in tension as long as the ΔT is <50°F.

D. Incorrect – Plausible since the plant can be cooled down with one SG. A candidate may believe that it is conservative to not feed a dry steam generator if it's not needed. Feeding a dry SG could damage it if it is not within limits.

Technical Reference(s): DB-OP-02000 RPS, SFAS, SFRCS TRIP, OR SG TUBE RUPTURE, SR4, Step 4.4 and Attachment 5, page 329 for feeding a dry SG, DP-OP-02000 BASES AND DEVIATION DOCUMENT SR4, p452

Reference(s) Provided None

Learning Objective:

Question Source: Bank # _____

Modified Bank # _____ (note changes or attach parent)

New X

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge _____

Comprehension / Analysis / **H**igher Cognitive X

10 CFR Part 55 Content: 55.41 (b)(5), (b)(7)

55.43 _____

Comments:

This question meets Tier 1 criteria as it tests the applicant's knowledge of actions found in the Emergency Operating Procedures.

Examination Outline Cross-Reference: Q27

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	00003 A2.06	
Importance Rating	3.5	

K/A Statement: Ability to (a) predict the impacts of the following on the REACTOR COOLANT PUMP SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: CCWS malfunction

Proposed Question: Q27

27. The Reactor is operating at 100% RTP with the following indications:
    - 11-3-A, CCW SURGE TK LVL LO is in alarm
    - 11-2-A, CCW SURGE TK LVL LO-LO is in alarm
    - CCW Surge tank indicates 34" and lowering

    What operator action(s) MUST be performed for the stated conditions?

    A. Start the Standby CCW Pump.

    B. Trip the Reactor then Trip ALL RCPs.

    C. Stop ALL RCPs and ISOLATE Seal Return flow.

    D. Verify CCW is available to provide cooling to the RCPs.

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Proposed Answer: **B**

Explanation Q27

A. Incorrect - Plausible because the STBY CCW Pump is started per Attachment 2, Shutdown of a Leaking CCW Essential Header when level is at 33 inches.

B. Correct - The direction from DB-OP-2523, CCW MALFUNCTIONS step 4.1.1 states that IF AT ANY TIME CCW surge tank reaches 35 inches, Trip the reactor, trip all RCPs and ensure non-essential CCW headers are isolated.

C. Incorrect - Plausible because at 35" in the CCW surge tank, non- essential CCW loops are isolated. The applicant may believe that this would cause a loss of seal injection from the loss of CCW cooling to the Makeup pump and for seal return temperature to be > 200F due to the loss of CCW cooling to the seal return coolers. DB-OP-2515 RCP Malfunctions step 4.4.1 states that on a loss of CCW and Seal Injection to all RCPs is confirmed, then trip the reactor, stop all RCPs and return to the AOP following immediate and supplemental actions. Step 4.4.2 would then direct the isolating Seal Return to have an idle RCP with Seal Return temperature greater than 200F.

D. Incorrect - Plausible because verifying CCW is available to provide cooling to the RCPs is directed by attachment 2 in DB-OP-2515 RCP Malfunctions step 4.3.4: Determine the cause of the loss of CCW flow. Refer to attachment 2 Verification of CCW flow to RCPs. However, CCW flow to ALL RCPs is lost once CCW surge tank drops below 35 inches and the non-essential header isolations automatically close.

Technical Reference(s): DB-OP-02515 RCP MALFUNCTIONS, DB-OP-02523 CCW MALFUNCTIONS Step 4.1.1 for lowering CCW surge tank level at 35 inches, ITS for DB, DB UFSAR, SD 016 CCW System

Reference(s) Provided None

Learning Objective:

Question Source: Bank # _____

Modified Bank # _____ (note changes or attach parent)

New X

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge X

Comprehension / Analysis / **H**igher Cognitive _____

10 CFR Part 55 Content: 55.41 (b)(5)

55.43 _____

Comments:

Examination Outline Cross-Reference: Q28

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	00003 G2.2.44	
Importance Rating	4.2	

K/A Statement: REACTOR COOLANT PUMP SYSTEM: (EQUIPMENT CONTROL) Ability to interpret control indications to verify the status and operation of a system and understand how operator actions and directives affect plant and system conditions.

Proposed Question: Q28

28. The plant is starting up following a refueling outage.
- RCPs 2-1 and 2-2 are running
- RCPs 1-1 and 1-2 are off

Preparations are being made to start RCP 1-2:
- RCS loop 1 WR Tc is 420°F
- Seal injection flow to RCP 1-2 is 3.5 gpm
- Upper and Lower bearing oil levels are normal
- All Bus voltages are normal
- CCW flows are normal
- HISMU59D, RCP 1-2 Seal Return Isolation, Red Light is ON
- Computer point P800, AC Oil Pmp Disch Press indicates NORM for RCP 1-2

Based on the above conditions, RCP 1-2 will ____(1)____.

Undervoltage Bypass Pushbuttons may be held in the "Bypass" position for a MAXIMUM of ____(2)____.

A. (1) NOT start
(2) 30 seconds

B. (1) start
(2) 30 seconds

C. (1) NOT start
(2) 1 minute

D. (1) start
(2) 1 minute

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Proposed Answer: **D**

Explanation: Q28

A. Incorrect - (1) Incorrect. All RCP start interlocks are met and the RCP will start. Plausible because the starting interlock based on RCS WR Tc is > 440ºF for the 4th RCP only. (2) Incorrect. Undervoltage Bypass Pushbuttons may be operated for up to 1 minute. 30 seconds is plausible since it would allow enough time for the RCP amps to stabilize.

B. Incorrect - (1) is correct because all starting interlocks are met. (2) Incorrect. Undervoltage Bypass Pushbuttons may be operated for up to 1 minute. 30 seconds is plausible since it would allow enough time for the RCP amps to stabilize.

C. Incorrect. (1) Incorrect. All RCP start interlocks are met and the RCP will start. Plausible because the starting interlock based on RCS WR Tc is > 440ºF for the 4th RCP only. (2) Correct - Undervoltage Bypass Pushbuttons may be operated for up to 1 minute.

D. Correct. All RCP start interlocks are met and the RCP will start. Bus voltage > 10.56 Kv, Seal Return Valve open, Seal Injection > 3 gpm, CCW flows normal, upper bearing oil level > -1.25”, lower bearing oil level > -1.25”, oil lift pressure > 60 psig, RX power < 60% RTP. (2) Correct - Undervoltage Bypass Pushbuttons may be operated for up to 1 minute.

Technical Reference(s): DB-OP-06005 RCP OPERATIONS Step 3.1.23 for RCP start interlocks and Caution 3.1.31 for holding UV pushbutton for less than 1 minute, DB-OP-06900 PLANT HEATUP, DB-OP-06901 PLANT STARTUP, DB UFSAR, TS 3.3.8

Reference(s) Provided: None

Learning Objective:

Question Source: Bank # _____

Modified Bank # _____ (note changes or attach parent)

New X

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge _____

Comprehension / Analysis / **H**igher Cognitive X

10 CFR Part 55 Content: 55.41 (b)(5)

55.43 _____

Comments:

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Examination Outline Cross-Reference: Q29

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	00004 K5.11	
Importance Rating	3.4	

K/A Statement: Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the CHEMICAL AND VOLUME CONTROL SYSTEM: PTS

Proposed Question: Q29

29. A transient has occurred which caused the Reactor to Trip due to low RCS Pressure:
    - All Immediate Actions have been completed
    - RCPs were tripped IAW Specific Rule 2
    - HPI/LPI/MU has been placed in service per Attachment 8

    Current Conditions:
    - One Control Rod has failed to fully insert
    - Containment pressure is 0.5 psig and stable
    - PZR level is 10"
    - RCS temperature is 530°F and lowering
    - RCS pressure is 1540 psig rising

    Which of the following is the greatest concern while attempting to mitigate this event?

    A. Raise Subcooling Margin by ensuring full High Pressure Injection flow is maintained.

    B. Restoring PZR level by injecting with both the normal and alternate MU injection lines.

    C. Minimize Subcooling Margin by throttling Makeup Injection flow.

    D. Ensure adequate Shut Down Margin by borating the RCS per Specific Rule 1.

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Proposed Answer: **C**

Explanation: Q29

A. Incorrect. Plausible because full HPI flow is required when SCM is < 20ºF subcooled. SCM must be correctly calculated in this scenario. In this instance SCM is > 20ºF subcooled.

B. Incorrect. Plausible since normal post trip PZR level setpoint is 100 inches. However, PZR level is not the greatest concern. Restoring SCM is the basis behind maximizing MU flow with both trains. In this instance SCM is > 20ºF subcooled.

C. Correct. The greatest concern for this condition is the combination of low RCS Temp and high RCS pressure, which will lead to a pressurized thermal shock condition under natural circulation conditions. Specific Rule 5 for Pressurized Thermal Shock is applicable with no RCPs running and MUP injecting via normal and alternate paths from the BWST per Attachment 8.

D. Incorrect. Plausible because SDM is a concern. However, emergency boration is only required if more than one Control Rod failed to insert or if IR/SR indication is rising.

Technical Reference(s): DB-OP-02000 RPS, SFAS, SFRCS TRIP or SGTR Attachment 8 puts normal and alternate injection in service, SR 5 is applicable. DB UFSAR

Reference(s) Provided: None

Learning Objective:

Question Source:

Bank #		
Modified Bank #		(note changes or attach parent)
New	X	

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level:

Memory / **F**undamental Knowledge	
Comprehension / Analysis / **H**igher Cognitive	X

10 CFR Part 55 Content:

55.41	(b)(5)
55.43	

Comments:

Examination Outline Cross-Reference: Q30

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	005K3.01	
Importance Rating	4.5	

K/A Statement: Knowledge of the effect that a loss or malfunction of the RESIDUAL HEAT REMOVAL SYSTEM will have on the following systems or components: RCS

Proposed Question: Q30

30. The plant has just been shutdown for a refueling outage.
    - Mode 5, RCS Pressure and Temperature are stable after stopping all RCPs
    - RCS drain preparations are in progress
    - DHR Train 1 cooling flow is 3000 GPM and stable

    Suddenly, power to the solenoid valve for DH 14B, DH COOLER 1 OUTLET FLOW CONTROL is lost.

    RCS temperature will _____(1)_____.
    DH Flow will be restored by throttling _____(2)_____.

    A. (1) rise
       (2) DH 13B, DH CLR 1 BYPASS

    B. (1) rise
       (2) DH 1B, DH PUMP 1 TO RCS ISO

    C. (1) lower
       (2) DH 13B, DH CLR 1 BYPASS

    D. (1) lower
       (2) DH 1B, DH PUMP 1 TO RCS ISO

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Proposed Answer: **D**

Explanation Q30

A. Incorrect - Part (1) Plausible for a fail position of CLOSED. However, DH 14B fails open. Part (2) is plausible since DH 13B is used to control temperature and flow during normal ops. It will not restore flow to it's original value when closed.

B. Incorrect - Part (1) Plausible for a fail position of CLOSED. However, DH 14B fails open. Part (2) is correct.

C. Incorrect – Part (1) is correct. DH 14B will fail open to the mechanical stop which will cause RCS Temperature to lower. Part (2) is plausible since DH 13B is used to control temperature and flow during normal ops.

D. Correct – Part (1) is correct. DH 14B will fail open to the mechanical stop which will cause RCS Temperature to lower. Part (2) is correct. IAW DB-OP-02003, 3-2-H LP INJ 1 FLOW HI, DH 14B, DH 13B, or DH 1B are throttled to restore flow of 3000 GPM or less. DH 14B is failed open and throttling DH 13B will not restore flow to it's original value.

Technical Reference(s): SD 42 system description for DECAY HEAT REMOVAL, DB-OP-02003 R19, 3-2-H LP INJ 1 FLOW HI Step 3.2, OS-004 Sh1, Decay Heat Removal

Reference(s) Provided: None

Learning Objective:

Question Source: Bank # _____

Modified Bank # _____ (note changes or attach parent)

New X

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge _____

Comprehension / Analysis / **H**igher Cognitive X

10 CFR Part 55 Content: 55.41 (b)(7)

55.43 _____

Comments:

Examination Outline Cross-Reference: Q31

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	00006 K1.02	
Importance Rating	4.4	

K/A Statement: Knowledge of the physical connections and/or cause and effect relationships between the EMERGENCY CORE COOLING SYSTEM and the following systems: ESFAS

Proposed Question: Q31

31. The plant is operating at 100% RTP with HPI Pump 1 removed from service for maintenance and its breaker racked out.

The following conditions occur:
- RCS pressure is 1590 psig and lowering
- Containment pressure is 18.0 psia and rising
- Pressurizer level is 265 inches and rising

With NO operator action, what is the status of the HPI pump Safety Actuation Monitoring (SAM) lights?

HPI Pump 1 SAM light is ____(1)_____.
HPI Pump 2 SAM light is ____(2)_____.

A. (1) OFF
(2) OFF

B. (1) OFF
(2) DIM

C. (1) DIM
(2) DIM

D. (1) FLASHING
(2) DIM

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Proposed Answer: **B**

Explanation Q31

A. Incorrect. SFAS level 2 actuation will occur if RCS pressure is </= 1600 psig OR CTMT pressure is >/= 18.7 psia. Level 2 starts both HPI pumps, however HPI #1 is tagged out for maintenance. The SAM light for HPI 1 would be off and remain off. The SAM light for HPI 2 would be DIM following the SFAS actuation due to low RCS pressure below the setpoint.

B. Correct. SFAS level 2 actuation will occur if RCS pressure is </= 1600 psig OR CTMT pressure is >/= 18.7 psia. Level 2 starts both HPI pumps, however HPI #1 is tagged out for maintenance. The SAM light for HPI 1 would be off and remain off and HPI 2 would be DIM following the SFAS actuation due to low RCS pressure below the setpoint.

C. Incorrect. SFAS level 2 actuation will occur if RCS pressure is </= 1600 psig OR CTMT pressure is >/= 18.7 psia. Level 2 starts both HPI pumps, however HPI #1 is tagged out for maintenance. The SAM light for HPI 1 would be off and remain off. The SAM light for HPI 2 would be DIM following the SFAS actuation due to low RCS pressure below the setpoint. Plausible because both DIM is the expected condition for both pumps following SFAS actuation if both pumps are in operation.

D. Incorrect. SFAS level 2 actuation will occur if RCS pressure is </= 1600 psig OR CTMT pressure is >/= 18.7 psia. Level 2 starts both HPI pumps, however HPI #1 is tagged out for maintenance. The SAM light for HPI 1 would be off and remain off. The SAM light for HPI 2 would be DIM following the SFAS actuation due to low RCS pressure below the setpoint. Plausible because a FLASHING light is observed when the SFAS actuation occurs, is blocked and positioned out of its SFAS required position, which would be the case for an open breaker on the HPI pump, however the component would not have been blocked.

Technical Reference(s): DB-OP-06405, SAFETY FEATURES ACTUATION SYSTEM PROCEDURE, Attachment 2, SAM lights; SD 02: System Description for SFAS F 2.3-3-1

Reference(s) Provided None

Learning Objective:

Question Source: Bank # _____

Modified Bank # _____ (note changes or attach parent)

New X

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge X

Comprehension / Analysis / **H**igher Cognitive _____

10 CFR Part 55 Content: 55.41 (b)(2)

55.43 _____

Comments:

Examination Outline Cross-Reference: Q32

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	00007 K4.05	
Importance Rating	2.6	

K/A Statement: Knowledge of PRESSURIZER RELIEF/QUENCH TANK SYSTEM design features and/or interlocks that provide for the following: Draining PRT/Quench tank.

Proposed Question: Q32

32. Quench Tank (QT) conditions:
    - Level is at 9ft 6in and slowly rising
    - Temperature is at 85°F
    - Pressure is 25 psig

    Which of the following statements is correct?

    The QT _____

    A. Circ Pump will AUTO start and pump the QT to the Miscellaneous Waste Drain Tank.

    B. Circ Pump will AUTO start and pump the QT to the Reactor Coolant Drain Tank.

    C. Discharge Valve will AUTO open, and the QT will drain to the Reactor Coolant Drain Tank.

    D. Discharge Valve will AUTO open, and the QT will drain to the Miscellaneous Waste Drain Tank.

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Proposed Answer: **C**

Explanation: Q32

A. Incorrect. The QT Recirc pump Auto starts on high temperature not level. Plausible since the QT Recirc Pump can be used to drain the QT if manually aligned and started. Also, many systems drain or are pumped to the Misc Waste Drain Tank (MWDT).

B. Incorrect. The QT Recirc pump Auto starts on high temperature not level. Plausible since the QT Recirc Pump can be used to pump the QT to the RCDT if manually aligned and started.

C. Correct. The Quench tank is normally aligned to automatically drain to the RCDT via RC225A, QUENCH TANK DISCHARGE TO RCDT, when level is greater than 9 ft 6 in.

D. Incorrect. The Quench tank is normally aligned to automatically drain to the RCDT via RC225A when level is greater than 9 ft 6 in. Plausible since many systems drain to the MWDT.

Technical Reference(s): DB-OP-06004, System procedure for QUENCH TANK Limit and Precaution 2.2.4, and Attachment 3; SD 039B, System Description for REACTOR COOLANT SYTEM ACCESSORY SYSTEMS Section 2.1.2.1.4, pages 2-3 and 2-4

Reference(s) Provided: None

Learning Objective:

Question Source: Bank # _____

Modified Bank # _____ (note changes or attach parent)

New X

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge X

Comprehension / Analysis / **H**igher Cognitive _____

10 CFR Part 55 Content: 55.41 (b)(7)

55.43 _____

Comments:

Examination Outline Cross-Reference: Q33

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	00008 K2.02	
Importance Rating	3.9	

K/A Statement: Knowledge of electrical power supplies to the following (SF8 CCW) COMPONENT COOLING WATER SYSTEM: CCW pumps

Proposed Question: Q33

33. The plant is operating at 100% RTP.

5 minutes ago, the following alarm was received:

- 1-3-D, BUS C1 LOCKOUT

What is the status of CCW Pump 1?

A. DEENERGIZED.

B. ENERGIZED and powered from the #1 EDG.

C. ENERGIZED and powered from C2 Bus.

D. ENERGIZED and powered via the alternate source breaker.

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Proposed Answer: **A**

Explanation Q33

A. Correct. The C1 Bus lockout condition opens the supply breaker and all load breakers. 1-1 CCW pump is powered from the C1 Bus.

B. Incorrect. The C1 Bus lockout condition opens the supply breaker and all load breakers. Plausible because the EDG starts on C1 Bus undervoltage but the EDG would run unloaded until it is locally shutdown due to no cooling water available.

C. Incorrect. The C1 Bus lockout condition opens the normal supply breaker from C2 and opens all load breakers. Plausible because the C2 bus would still be energized in this condition, and if the CCW pump were powered from the C2 bus it would be energized.

D. Incorrect. The C1 Bus lockout condition opens the supply breaker and all load breakers. Plausible because if the applicant believes the lockout was on the C2 bus, the C1 bus would dead bus fast transfer to the alternate source, and the CCW pump would still be energized, but via the alternate source.

Technical Reference(s): DB-OP-02521, Step 3.1.1 for C1 lockout, emergency shutdown the EDG. DB-OP-02001, Electrical distribution alarm panel 1 annunciators; 1-3-D p 60, SD 003A, System Description for 4160 VOLT AUXILIARY SYSTEM Section 2.1.2.6,b. bus lockout for essential buses, C1, page 2-4

Reference(s) Provided None

Learning Objective:

Question Source: Bank # _____

Modified Bank # _____ (note changes or attach parent)

New X

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge X

Comprehension / Analysis / **H**igher Cognitive _____

10 CFR Part 55 Content: 55.41 (b)(7)

55.43 _____

Comments:

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Examination Outline Cross-Reference: Q34

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	00008 K6.04	
Importance Rating	3.9	

K/A Statement: Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the Component Cooling Water System: CCW pump(s)

Proposed Question: Q34

34. Initial conditions:
- 100% RTP
- Component Cooling Water (CCW) Pump 2 is in service

The following occurs:
- CCW Pump 2 shears its shaft
- CCW Pump 1 fails to auto start

One minute later the Reactor Operator starts CCW Pump 1 and stops CCW Pump 2

The following Annunciator alarms have been received and have **NOT** been acknowledged:
- 11-4-B, CCW PMP 1 FLOW LO
- 11-5-B, CCW PMP 2 FLOW LO
- 6-5-B, SEAL CCW FLOW LO
- 2-3-A, LETDOWN TEMP HI ALARMS

All automatic actions have completed, and no other operator actions have been taken.

Based on these conditions, which of the following identifies the actions required to restore CCW system flow to its previous value?

A. Open Reactor Coolant Pump Seal Cooler CCW Return Valves

B. Close Loop 2 Non-Essential CCW Isolation Valves

C. Restore the Aux Building Non-Essential Header

D. Restore RCS Letdown

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Proposed Answer: **D**

Explanation: Q34

A. Incorrect – plausible since the low flow alarm is flashing and there is an automatic closure interlock due to high pressure

B. Incorrect – plausible since the valves would not auto close on a sheared shaft until CCW 2 pump breaker was opened

C. Incorrect – plausible since it's a "non-essential" header and CC1495 has a close interlock due to low CCW Surge Tank Level. However, it does not close due to low flow.

D. Correct – CCW valves to Letdown Coolers automatically close following a high temperature isolation

Technical Reference(s): SD-048 R4 Page 2-26

Reference(s) Provided None

Learning Objective:

Question Source: Bank # X

Modified Bank # ____ (note changes or attach parent)

New ____

Question History: Last NRC Exam DB2016 Q35 (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge ____

Comprehension / Analysis / **H**igher Cognitive X

10 CFR Part 55 Content: 55.41 (b)(7)

55.43 ____

Comments:

Examination Outline Cross-Reference: Q35

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	000010 K2.02	
Importance Rating	3.1	

K/A Statement: Knowledge of electrical power supplies to the PZR pressure controller: SF3 PZR PCS, PRESSURIZER PRESSURE CONTROL SYSTEM

Proposed Question: Q35

35. Essential Bank 2 Pressurizer heater bank control switch is in the ON position.

If RCS pressure is stable at the normal operating point, and Pressurizer level fails to 37", which ONE of the following explains the status of the Essential Bank 2 heaters?

The heater bank is _____.

A. energized because manual control overrides the Pressurizer low-low level heater cutoff

B. de-energized because the Pressurizer low-low level heater cutoff overrides manual control

C. energized because Pressurizer level is above the low-low level heater cutoff setpoint

D. de-energized because normal RCS pressure is above the heater bank cycle setpoint

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Proposed Answer: **A**

Explanation Q35

A. Correct - In automatic, the design of the Pressurizer Heaters removes power on LOW LOW pressurizer Level (40 inches). Operating the Pressurizer in manual (ON) overrides this design feature.

B. Incorrect - Plausible if the candidate does not understand that the "ON" position for the heaters overrides the Low-level cutoff.

C. Incorrect – Plausible if the candidate knows the heater banks start to uncover at 28 inches and believes 28 inches is the setpoint for the low pressurizer level heater cutoff.

D. Incorrect - Plausible because this bank of heaters is off at normal RCS Pressure when in Auto.

Technical Reference(s): DB-OP-06003 R36 Attachment 7, Pressurizer Pressure And Level Setpoints, PZR Heater Control Panel Placard, DB-OP-02513 R14 Step 4.6.4 RNO

Reference(s) Provided None

Learning Objective:

Question Source: Bank # X

Modified Bank # _____ (note changes or attach parent)

New _____

Question History: Last NRC Exam DB2013 Q36 (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge X

Comprehension / Analysis / **H**igher Cognitive _____

10 CFR Part 55 Content: 55.41 (b)(7)

55.43 _____

Comments:

Examination Outline Cross-Reference: Q36

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	000012 G2.4.35	
Importance Rating	3.8	

K/A Statement: Knowledge of nonlicensed operator responsibilities during an emergency: REACTOR PROTECTION SYSTEM (2.4.35) EMERGENCY PROCEDURES/PLAN

Proposed Question: Q36

36. The plant is operating at 100% RTP when a loss of both MFW pumps occurs.

The following conditions are noted:
- RCS pressure is 2400 psig and rising
- Rx power is 100% RTP
- The turbine is NOT tripped

Assuming previously performed required actions have failed, what is the NEXT anticipated action(s) for the given conditions?

A. Attempt to Trip the Main Turbine by pressing the EMERGENCY TRIP pushbuttons.

B. Locally open A, B and C Reactor Trip Breakers from the low voltage switchgear room.

C. Shift the MDFP from MFW to the AFW mode per Attachment 5: Guidelines for Restoring Feedwater, section F.

D. Initiate AND Isolate SFRCS using MANUAL ACTUATION switches.

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Proposed Answer: **B**

Explanation: Q36

A. Incorrect - The reactor should have tripped with RCS pressure greater than the RPS setpoint of 2355 psig. Plausible because tripping the turbine is an expected action when the reactor is tripped, however, DB-OP-02000 specifically prohibits continuing past RNO step 3.3 until the reactor has been shutdown.

B. Correct - The reactor should have tripped with RCS pressure greater than the RPS setpoint of 2355 psig. Per the Immediate Action RNO step 3.3 locally open the RTBs is a required action to shutdown the reactor.

C. Incorrect - The reactor failed to trip when RCS pressure exceeded 2355 psig setpoint. Plausible because the action to swap the MDFP from MFW mode to AFW mode is a possible action for the loss of AFW at lower power levels.

D. Incorrect - The reactor should have tripped with RCS pressure greater than the RPS trip setpoint of 2355 psig, This is the expected action when the Reactor trips but the Turbine does not.

Technical Reference(s): DB-OP-02000, Immediate actions, Step 3.3 RNO #4.

Reference(s) Provided None

Learning Objective:

Question Source: Bank # _____

Modified Bank # _____ (note changes or attach parent)

New X

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge _____

Comprehension / Analysis / **H**igher Cognitive X

10 CFR Part 55 Content: 55.41 (b)(10)

55.43 _____

Comments:

Examination Outline Cross-Reference: Q37

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	000013 A3.12	
Importance Rating	3.9	

K/A Statement: Ability to monitor automatic features of the (SF2 ESFAS) ENGINEERED SAFETY FEATURES ACTUATION SYSTEM including ESFAS permissives.

Proposed Question: Q37

37. The plant is in MODE 3 and being shut down for a refueling outage. All 4 RCPs are running.

During the controlled RCS depressurization, the following indications and alarms are present:

- RC2A3 SFAS CH 4 RCS Pressure is: 1665 psig
- RC2B3 SFAS CH 3 RCS Pressure is: 1677 psig
- RC2A4 SFAS CH 2 RCS Pressure is: 1662 psig
- RC2B4 SFAS CH 1 RCS Pressure is: 1679 psig
- 5-2-C, RC Low Pressure Trip Block Permit is LIT

Based on the above indications, conditions are ___(1)____ to block the SFAS Low RCS Pressure Trip actuation.

____(2)____ to adequately block the SFAS Low RCS Pressure Trip during the controlled shutdown.

A. (1) met
(2) SFAS CH 1 and CH 2 LO TRIP BLK pushbuttons are used

B. (1) met
(2) SFAS CH 2 and 4 equipment block pushbuttons are used

C. (1) NOT met
(2) The SFAS Shutdown Bypass Key switch MUST be used

D. (1) NOT met
(2) At least 3 SFAS Channels MUST be blocked

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Proposed Answer: **D**

Explanation: Q37

A. Incorrect. Plausible because SFAS has two actuation channels. SFAS requires that a minimum of 3 of 4 trip bistables must be blocked (<1670 psig) to adequately block SFAS operation.

B. Incorrect. Plausible because the applicant may believe that only 2 of 4 bistables are required to be below the block setpoint. Also plausible because the SFAS equipment can be blocked at the component level using the SFAS equipment block pushbuttons after an actuation.

C. Incorrect. Part 1 is correct. Part 2 is plausible because the applicant may believe that the shutdown bypass key switch could be used to prevent actuation, however use of this key switch is restricted to modes 5, 6 or defueled.

D. Correct. The SFAS Block permissive logic requires that a minimum of 3 of 4 trip bistables must be blocked (<1670 psig) to adequately block SFAS operation.

Technical Reference(s): System Description 002 for (SFAS) Step 2.1.2.6 three of four needed to block; DB-OP-02005, Primary Instrumentation Alarm Panel 5 Annunciators; DB-OP-06903, Caution 3.37.2 SFAS RCS LO PRESS TRIP BLOCK PERMISSIVE is 1670 psig.

Reference(s) Provided None

Learning Objective:

Question Source: Bank # _____

Modified Bank # _____ (note changes or attach parent)

New X

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge X

Comprehension / Analysis / **H**igher Cognitive _____

10 CFR Part 55 Content: 55.41 (b)(7)

55.43 _____

Comments:

Examination Outline Cross-Reference: Q38

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	000013 K6.10	
Importance Rating	4.0	

K/A Statement: Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the (SF2 ESFAS ENGINEERED SAFETY FEATURES ACTUATION SYSTEM: Feedline break.

Proposed Question: Q38

38. A plant transient occurred causing the Reactor to trip.

The following indications were noted just prior to the trip:

- Rx power is at 97% RTP and lowering
- Steam Generator 1 pressure 900 psig and rising
- Steam Generator 2 pressure 880 psig and stable
- Feedwater Header Pressure at the outlet of HPFW Heaters 1-6 and 2-6 is 750 psig and lowering
- Pressurizer level is 225" and rising
- Pressurizer pressure is 2260 psig and rising

Based on the indications above, the SFRCS (Steam and Feed Rupture Control System) actuation will feed _____ Steam Generator(s) with Auxiliary Feedwater.

A. ONLY #1

B. ONLY #2

C. BOTH

D. NEITHER

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Proposed Answer: **C**

Explanation: Q38

A. Incorrect - Plausible because on a loss of MFW, as indicated by SG - FW reverse d/p high > 125 psid, SFRCS will start AFW and feed both SGs via the AFW system piping. On a low main steam line pressure of < 630 psig, the SFRCS system will isolate the first SG to reach the setpoint and align AFW to feed the second SG. Since neither SG has reached the 630psig setpoint, AFW would not be isolated to either SG.

B. Incorrect - Plausible because on a loss of MFW, as indicated by SG - FW reverse d/p high > 125 psid, SFRCS will start AFW and feed both SGs via the AFW system piping. On a low main steam line pressure of < 630 psig, the SFRCS system will isolate the first SG to reach the setpoint and align AFW to feed the second SG. Since neither SG has reached the 630 psig setpoint, AFW would not be isolated to either SG.

C. Correct – The conditions in the stem are indicative of a feed line rupture. With SG – FW d/p > 125 psid, SFRCS will initiate and feed both SGs with AFW.

D. Incorrect - Plausible because on a SG - FW reverse d/p high of > 125 psid, SFRCS will start AFW and feed both SGs via the AFW system piping. The applicant may believe incorrectly that both SGs are required to have a high feedwater line check valve reverse d/p to get the SFRCS initiation, in which case main feedwater would still be feeding both SG's, not AFW.

Technical Reference(s): SD 010, System Description for SFRCS Section 3.3 Setpoints; OS12A, MFW system drawing, DB UFSAR

Reference(s) Provided None

Learning Objective:

Question Source: Bank # _____

Modified Bank # _____ (note changes or attach parent)

New X

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge ____

Comprehension / Analysis / **H**igher Cognitive X

10 CFR Part 55 Content: 55.41 (b)(7)

55.43 _____

Comments:

Examination Outline Cross-Reference: Q39

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	000022 A1.02	
Importance Rating	3.9	

K/A Statement: Ability to predict and/or monitor changes in parameters associated with operation of the (SF4 CCS) CONTAINMENT COOLING SYSTEM, including containment pressure.

Proposed Question: Q39

39. The plant is operating at 100% RTP with 1 AND 2 Containment Air Coolers (CAC)s in FAST speed.

    Which of the following conditions or malfunctions would require Technical Specification action within one hour or less?

    A. Loss of a running CAC resulting in Containment pressure rising to 28” H2O gauge.

    B. Outside air temperature changes resulting in Containment Average Temperature rising to 118°F.

    C. One Containment Spray train is removed from service.

    D. CV5070, Containment Vacuum Relief Isolation, failed its closed stroke time and is declared inoperable.

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Proposed Answer: **A**

Explanation: Q39

A. Correct. Containment pressure exceeding LCO 3.6.4 maximum value of 25" H20 requires restoring pressure to within limits in one hour.

B. Incorrect. Plausible because Containment Average Temperature above 120F requires entering LCO 3.6.5.

C. Incorrect. Plausible because one containment spray train removed from service requires entering LCO 3.6.6 condition A, however Condition A has a 7-day completion time.

D. Incorrect. Plausible because CV5070 is a containment isolation valve and would require entering LCO 3.6.3 condition A, however it is a 4-hour completion time.

Technical Reference(s): ITS rev 338 for Davis Bessie TS 3.6, Containment Systems

Reference(s) Provided None

Learning Objective:

Question Source: Bank # _____

Modified Bank # _____ (note changes or attach parent)

New X

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge X

Comprehension / Analysis / **H**igher Cognitive _____

10 CFR Part 55 Content: 55.41 (b)(5)

55.43 _____

Comments:

Examination Outline Cross-Reference: Q40

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	000026 A2.09	
Importance Rating	2.8	

K/A Statement: Ability to (a) predict the impacts of the following on the (SF5 CSS) CONTAINMENT SPRAY SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations; Radiation hazard potential of BWST/RWST

Proposed Question: Q40

40. A Large Break Loss of Coolant Accident occurs causing a Loss of SCM.

The crew has completed the required actions in DB-OP-02000, Section 5, LACK OF ADEQUATE SUBCOOLING MARGIN and routed to Section 10, Large LOCA Cooldown.

(1) When are the BREAKERS for DH7A, DH7B, DH9A, DH9B and HP31, CLOSED during a large break LOCA?

(2) How is the radiation hazard minimized when restoring power to DH7A, DH7B, DH9A, DH9B and HP31?

DH7B, BWST LINE 1 ISO VALVE
DH7A, BWST LINE 2 ISO VALVE
DH9B, DH PUMP 1 SUCTION FROM EMERG SUMP
DH9A, DH PUMP 2 SUCTION FROM EMERG SUMP
HP31, HPI PUMP 2 MIN RECIRC ISO

A. (1) When annunciator 5-3-A, BWST LOLO LVL XFER TO EMER SUMP alarms.
(2) A Radiation Protection Technician is required to accompany the operator during the task to ensure the minimum dose is received.

B. (1) When annunciator 5-3-A, BWST LOLO LVL XFER TO EMER SUMP alarms.
(2) The operator must follow the prescribed route to avoid an exposure significantly higher than the projected 2 rem.

C. (1) Within 23 minutes from the time RCS leakage indicated a Large Break LOCA.
(2) A Radiation Protection Technician is required to accompany the operator during the task to ensure the minimum dose is received.

D. (1) Within 23 minutes from the time RCS leakage indicated a Large Break LOCA.
(2) The operator must follow the prescribed route to avoid an exposure significantly higher than the projected 2 rem.

Proposed Answer: **D**

Explanation: Q40

A. Incorrect - Part (1) Plausible because this annunciator comes in at 9.0 ft in the BWST and the ARP directs the transfer of the suction from the BWST to the Emergency Sump if DB-OP-2000 is in effect. Incorrect because insufficient time would be available to close the breakers necessary to perform the transfer before losing NPSH to the operating ECCS Pumps Part (2) is plausible since RP coverage is required for many dose significant jobs and/or the dose rates are unknown.

B. Incorrect - Part (1) Plausible because this annunciator comes in at 9.0 ft in the BWST and the ARP directs the transfer of the suction from the BWST to the Emergency Sump if DB-OP-2000 is in effect. Incorrect because insufficient time would be available to close the breakers necessary to perform the transfer before losing NPSH to the operating ECCS Pumps. Part (2) is correct

C. Incorrect. Plausible because Part (1) is correct. Part (2) is plausible since RP coverage is required for many dose significant jobs and/or the dose rates are unknown.

D. Correct. Part (1) is correct. This is an TCOA (23 MIN) to ensure the ECCS suction is ready to transfer when BWST level is 9.0 ft. Part (2) is correct. A worst-case dose of 2 REM was predicted. Failure to follow the prescribed route could result in a significantly higher dose. This dose is within the limits established for annual personnel dose and planned special exposure.

Technical Reference(s): DB-OP-02000, RPS, SFAS, SFRCS TRIP or SG TUBE RUPTURE Attachment 7, words at the front of Attachment and Warning Attachment 7, DB-OP-02000 Basis document, DB-OP-02005 PRIMARY INSTRUMENTATION ALARM PANEL 5 ANNUNCIATORS PURIFICATION

Reference(s) Provided: None

Learning Objective:

Question Source: Bank # _____

Modified Bank # _____ (note changes or attach parent)

New X

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge X

Comprehension / Analysis / **H**igher Cognitive _____

10 CFR Part 55 Content: 55.41 (b)(5)

55.43 _____

Comments:

Examination Outline Cross-Reference: Q41

Level	RO	SRO
Tier #	2	
Group #	1	

Examination Outline Cross-Reference: Q41

K/A #	000026 K2.01
Importance Rating	3.9

K/A Statement: Knowledge of electrical power supplies to the following CONTAINMENT SPRAY SYSTEM: Containment spray pumps

Proposed Question: Q41

41. The plant is operating at 100% RTP.

The following events then occur:

- A Large Break LOCA
- Total loss of offsite power
- Containment pressure is 42 psia
- Emergency Diesel Generator 1 fails to start and cannot be manually started

Based on these conditions, what will be the status of the Containment Spray Pumps?

Containment Spray Pump 1 will be _____(1)_____ and
Containment Spray Pump 2 will be _____(2)_____.

A. (1) running
(2) running

B. (1) running
(2) stopped

C. (1) stopped
(2) stopped

D. (1) stopped
(2) running

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Proposed Answer: **D**

Explanation: Q41

A. Incorrect – CTMT Spray Pump 1 (P56-1) will have no power since EDG 1 did not start - Plausible as both pumps would have received a start signal on high containment pressure of 40 psia.

B. Incorrect – CTMT Spray Pump 2 (P56-2) will have a start signal from high containment pressure, and it will have power from the EDG 2. CTMT Spray Pump 1 (P56-1) will have no power since EDG 1 did not start – Plausible as the actual condition is one containment spray pump running and one NOT running due to containment pressure and the loss of power to a safety bus.

C. Incorrect – CTMT Spray Pump 2 (P56-2) will have a start signal from high containment pressure, and it will have power from the EDG 2. CTMT Spray Pump 1 (P56-1) will have no power since EDG 1did not start – Plausible if candidate does not know SFAS Level 4 setpoint for high containment pressure.

D. Correct – CTMT Spray Pump 2 (P56-2) will have a start signal from high containment pressure, and it will have power from the EDG 2. CTMT Spray Pump 1 (P56-1) will have no power since EDG 1 did not start.

Technical Reference(s): DB-OP-02000 R32, RPS, SFAS, SFRCS TRIP, OR SG TUBE RUPTURE (p503). SD 002 R6, SFAS (T1.2-2-1)
SD 022A, SYSTEM DESCRIPTION FOR CONTAINMENT SPRAY

Reference(s) Provided: None

Learning Objective: OPS-SYS-306-10K

Question Source: Bank # X

Modified Bank # _____ (note changes or attach parent)

New _____

Question History: Last NRC Exam DB 2011 Q44 (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge ____

Comprehension / Analysis / **H**igher Cognitive X

10 CFR Part 55 Content: 55.41 (b)(7)

55.43 _____

Comments:

Examination Outline Cross-Reference: Q42

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	000039 A3.01	
Importance Rating	2.8	

K/A Statement: Ability to monitor automatic features of the MAIN AND REHEAT STEAM SYSTEM, including: Moisture separator reheater steam supply

Proposed Question: Q42

42. The Reactor is being maneuvered from 100% to 5% RTP in accordance with DB-OP-06902, POWER OPERATIONS:

MS 195A and MS 195B RSHLVs (Reheat Steam High Load Valves), will automatically CLOSE when unit load reaches approximately _____.

A. 850 MWe

B. 600 MWe

C. 270 MWe

D. 90 MWe

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Proposed Answer: **B**

Explanation: Q42

A. Incorrect – 850 Mwe corresponds to ~90% RTP which is where the Auxiliary Feed Pump Turbine Main Steam Minimum flow lines are removed from service.

B. Correct – 2nd Stage Reheat HI Load Valves (MS 195A & B) Auto Close at approximately 600 MWe (sensed by Cross around pressure).

C. Incorrect – Plausible as 270 MWe (LLLs) is power when DB-OP-06902, POWER OPERATIONS directs that S/G REACTOR DEMAND H/A Station is placed in HAND.

D. Incorrect – Plausible as 10% turbine load (90 MWe) (sensed by Cross around pressure) is when the 2nd Stage Reheat Source valves automatically close.

Technical Reference(s): DB-OP-06902 R72, POWER OPERATIONS (p81)

OS-008 SH 3

Reference(s) Provided: None

Learning Objective: OPS-SYS-204

Question Source: Bank # _____

Modified Bank # _____ (note changes or attach parent)

New X

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge X

Comprehension / Analysis / **H**igher Cognitive _____

10 CFR Part 55 Content: 55.41 (b)(7)

55.43 _____

Comments:

Examination Outline Cross-Reference: Q43

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	000053 A1.09	
Importance Rating	3.6	

K/A Statement: Ability to predict and/or monitor changes in parameters associated with operation of the INTEGRATED CONTROL SYSTEM (BW), including: RCS flow (RCP status)

Proposed Question: Q43

43. Plant conditions:
    - The plant is at 79% RTP
    - Normal equipment lineups exist
    - ICS is in automatic

    Subsequently, Reactor Coolant Pump 1-1 trips.

    With the above condition, _____.

    A. RPS will trip the reactor

    B. ICS will reduce reactor power to 45%

    C. ICS will reduce reactor power to 55%

    D. ICS will reduce reactor power to 75%

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Proposed Answer: **D**

Explanation: Q43

A. Incorrect – Plausible if the examinee believes the reactor will trip due to an RPS Flux/Δ Flux/Flow Trip or being above the 3 RCP Hi Flux Trip Setpoint power value; however, The Hi Flux Trip Setpoint for three RCP operations (80.6%) is manually set by I&C. In this case the Hi Flux Trip Setpoint is still at 104.7%.

B. Incorrect – Plausible as TS 3.1.4 power limit is 45% when the combination of a dropped rod and 3 RCPs exists.

C. Incorrect – Plausible as the limit for 1 MFW pump in operation is 55% and the limit when deaerator level is low is also 55%.

D. Correct – DB-OP-02014, MSR/ICS ALARM PANEL 14 ANNUNCIATORS alarm 14-3-C, ICS RCP LOSS RUNBACK has the operator verify the runback to 75% RTP due to the tripped RCP.

Technical Reference(s): DB-OP-02515 R22, REACTOR COOLANT PUMP AND MOTOR ABNORMAL OPERATION (p51).
DB-OP-02014, R17 MSR/ICS ALARM PANEL 14 ANNUNCIATORS alarm 14-3-C, ICS RCP LOSS RUNBACK (p28)

SD 045, SYSTEM DESCRIPTION FOR INTEGRATED CONTROL SYSTEM.

Reference(s) Provided: None

Learning Objective: OPS-SYS514-04K

Question Source: Bank # _____

Modified Bank # X (note changes or attach parent)

New _____

Question History: Last NRC Exam DB 2008 Q61 (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge X

Comprehension / Analysis / **H**igher Cognitive _____

10 CFR Part 55 Content: 55.41 (b)(5)

55.43 _____

Comments:

Modified by changing the event, and distractor with 55% value and reordered distractors.

Examination Outline Cross-Reference: Q44

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	000059 K1.13	
Importance Rating	2.9	

K/A Statement: Knowledge of the physical connections and/or cause and effect relationships between the MAIN FEEDWATER SYSTEM and the following system: S/GB system

Proposed Question: Q44

44. Initial Conditions:
    - Startup is in progress at 10% RTP
    - Steam Generator blowdown lines are in service

    Which of the following WILL automatically isolate this flow path?

    A. Manual SFRCS Initiate ONLY actuation.

    B. Loss of all Reactor Coolant Pumps.

    C. Feedwater deviation that results in Steam Generator Low Level.

    D. Feedwater deviation that results in Steam Generator High Level.

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Proposed Answer: **D**

Explanation: Q44

A. Incorrect – Manual initiation of SFRCS would result in the start of AFW but would not cause an isolation of blowdown. Plausible as SFRCS can isolate blowdown when isolation is manually actuated.

B. Incorrect – Loss of All RCPs trip provides mitigation for a loss of off-site power resulting in loss of four reactor coolant pumps, which causes loss of reactor coolant flow and therefore auxiliary feedwater is needed to establish reactor coolant natural circulation flow. Plausible as SFRCS can isolate blowdown and SFRCS would INITIATE. In this instance, it would not cause isolation of blowdown.

C. Incorrect – The low-level trip does not differentiate which steam generator has low inventory or which steam generator signaled low level first. The SFRCS will not isolate main feedwater flow to and main steam from either steam generator but will initiate turbine driven auxiliary feedwater flow from both auxiliary feedwater trains to their respective steam generators. Plausible as SFRCS can isolate blowdown and SFRCS would INITIATE. In this instance, it would not cause isolation of blowdown.

D. Correct – Steam Generator High Level causes an SFRCS INITIATION and ISOLATION. This isolation would cause blowdown to be isolated.

Technical Reference(s): DB-OP-02000 R32, RPS, SFAS, SFRCS TRIP, OR SG TUBE RUPTURE (p497)

SD 10, SYSTEM DESCRIPTION FOR STEAM AND FEEDWATER LINE RUPTURE CONTROL SYSTEM

Reference(s) Provided: None

Learning Objective:

Question Source: Bank # X

Modified Bank # ____ (note changes or attach parent)

New ____

Question History: Last NRC Exam DB 2016 Q47 (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge ____

Comprehension / Analysis / **H**igher Cognitive X

10 CFR Part 55 Content: 55.41 (b)(2 to 9)

55.43 ____

Comments:

Examination Outline Cross-Reference: Q45

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	000061 A1.02	
Importance Rating	3.8	

K/A Statement: Ability to predict and/or monitor changes in parameters associated with operation of the AUXILIARY/EMERGENCY FEEDWATER SYSTEM, including: S/G pressure

Proposed Question: Q45

45. The plant tripped from 100% RTP.
    - Steam Generator 1 pressure is 600 psig; level is 52 inches
    - Steam Generator 2 pressure is 640 psig; level is 58 inches

    Which of the following is the expected plant response to these plant conditions?

    A. AFW is feeding Steam Generator 1 ONLY.

    B. AFW is feeding Steam Generator 2 ONLY.

    C. MFW is feeding BOTH Steam Generators.

    D. AFW is feeding BOTH Steam Generators.

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Proposed Answer: **B**

Explanation: Q45

A. Incorrect – Plausible as AFW starts on low SG water level (23.5”). SG 1 has a lower level but is above the AFW start setpoint. SG 1 is below the AFW auto start on SFRCS actuation and isolation for low steam pressure (< 630 psig). Plausible if student assesses that the low pressure SG requires AFW flow.

B. Correct – The SFRCS Actuation /Isolation setpoint for Steam Line Low Pressure is 630 psig per Table 3.3-1, SFRCS Setpoints, located in System Description (SD-10), Rev. 7. Therefore, SG 1 will be isolated and SG 2 will be fed from AFW.

C. Incorrect – Plausible as MFW was feeding both SGs at the start of the event and would continue to feed both SGs until pressure < 630 psig”.

D. Incorrect – Plausible as SG 1 is below the SFRCS low pressure actuation and isolation set point which causes AFW to auto start.

Technical Reference(s): DB-OP-02000 R32, RPS, SFAS, SFRCS TRIP, OR SG TUBE RUPTURE (p497 & p498) SD-10 R7, STEAM AND FEEDWATER LINE RUPTURE CONTROL SYSTEM (SFRCS) (T3.3-1-1)

Drawing OS-017A, Sheet 1, AUXILIARY FEEDWATER SYSTEM, Rev. 35;

Reference(s) Provided: None

Learning Objective: OPS-SYS-523-04K

Question Source: Bank # X

Modified Bank # _____ (note changes or attach parent)

New _____

Question History: Last NRC Exam DB 2004 Q47 (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge ___

Comprehension / Analysis / **H**igher Cognitive X

10 CFR Part 55 Content: 55.41 (b)(5)

55.43 _____

Comments:

Examination Outline Cross-Reference: Q46

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	000061 A4.03	
Importance Rating	4.1	

K/A Statement: Ability to manually operate and/or monitor the AUXILIARY/EMERGENCY FEEDWATER SYSTEM in the control room: AFW pump turbines

Proposed Question: Q46

46. The plant was at 100% RTP when an event occurred which caused a loss of off-site power.
- Both Emergency Diesel Generators are running and are supplying the C1 and D1 busses
- The SBO Diesel Generator is powering the D2 bus
- Both Auxiliary Feedwater Pumps (AFP) are running being supplied by Main Steam
- CST levels are 4.0 feet in both tanks and lowering

Subsequently,
- 10-5-G, AFP 1 SUCTION PRESS LO is LIT
- 10-5-H, AFP 2 SUCTION PRESS LO is LIT
- 10-4-G, AFP 1 TRBL is LIT
- SW1382, SERVICE WATER SUPPLY TO AFPT 1 has failed in the CLOSED position

As the Control Room Operator, you MUST _____.

A. block and close MS106 and MS106A (Main Steam supply to AFPT 1)

B. close MS5889A, Steam Admission Valve to AFPT 1

C. stop BOTH AFPs and initiate MU/HPI cooling

D. start the Emergency Feedwater Pump

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Proposed Answer: **A**

Explanation: Q46

A. Correct – With a AFP 1 TRBL ALARM in (pump suction <3.8 psig for >60 seconds) the AFPT steam inlet valves auto close to prevent pump damage.

B. Incorrect – Plausible since MS5889A is normally closed to prevent steam supplying the pump. It cannot be closed when an SFRCS signal is present.

C. Incorrect – Plausible since this would be the mode of cooling if all feedwater was lost; however, feedwater will not be lost in this situation since SW will supply water to AFP 2.

D. Incorrect – Plausible if main and auxiliary feedwater were lost this would be the required action. In this instance service water is still available to AFP2.

Technical Reference(s): DB-OP-02010 R27 Annunciator 10-5-G, AFP 1 SUCT PRESS LO (p75). OS-017A R4 SH 2 CL-3,

SD 015, SYSTEM DESCRIPTION FOR AUXILIARY FEEDWATER SYSTEM; DB_AFW Simplified Powerpoint

Reference(s) Provided: None

Learning Objective: OPS-SYS-523-04K

Question Source: Bank # _____

Modified Bank # X (note changes or attach parent)

New _____

Question History: Last NRC Exam DB 2008 Q24 (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge _____

Comprehension / Analysis / **H**igher Cognitive X

10 CFR Part 55 Content: 55.41 (b)(7)

55.43 _____

Comments:

Question was modified by changing stem conditions to make distractor A now the correct and answer. Original correct answer D was replaced with a new distractor.

Examination Outline Cross-Reference: Q47

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	000062 K6.02	
Importance Rating	3.4	

K/A Statement: Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the AC ELECTRICAL DISTRIBUTION SYSTEM: Breakers, relays, and disconnects

Proposed Question: Q47

47. Given the following plant conditions:
- HPI 1 pump breaker is in the TEST position
- Control power fuses are installed

The primary breaker connection _____(1)_____ connected to the bus.
The breaker can be operated _____(2)_____.

Which of the following completes the statements above?

A. (1) is NOT
(2) remotely ONLY

B. (1) is NOT
(2) locally or remotely

C. (1) is
(2) remotely ONLY

D. (1) is
(2) locally or remotely

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Proposed Answer: **B**

Explanation: Q47

A. Incorrect – First part is correct. Second part is incorrect. Plausible if the applicant believes that with the breaker in test, it can only be operated with the same control room switch used to start the pump.

B. Correct – Each breaker unit of 4160 switchgear has operating, test, and withdrawn positions. In the test position, the primary contacts are disconnected, and the auxiliary contacts for breaker and control circuit testing from local or remote points are operable. In the withdrawn position, all connections are disconnected.

C. Incorrect – First part is incorrect. Plausible because when in TEST the auxiliary contacts for breaker and control circuit testing are operable. Second part is incorrect. Plausible if the applicant believes that with the breaker in test, it can only be operated with the same control room switch used to start the pump.

D. Incorrect – First part is incorrect. Plausible because when in TEST the auxiliary contacts for breaker and control circuit testing are operable. Second part is correct.

Technical Reference(s): SD 003A R6, 4160 VOLT AUXILIARY SYSTEM (p2-13 first bullet)

Reference(s) Provided: None

Learning Objective: OPS-SYS-405-5K

Question Source: Bank # X

Modified Bank # ____ (note changes or attach parent)

New ____

Question History: Last NRC Exam Oconee 2018 Q49 (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge ____

Comprehension / Analysis / **H**igher Cognitive X

10 CFR Part 55 Content: 55.41 (b)(7)

55.43 ____

Comments:

Examination Outline Cross-Reference: Q48

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	000063 A2.04	
Importance Rating	3.9	

K/A Statement: Ability to (a) predict the impacts of the following on the DC ELECTRICAL DISTRIBUTION SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Battery malfunctions

Proposed Question: Q48

48. DB-ME-03004, Station Battery Monthly Surveillance (Technical Specification Surveillance Requirement 3.8.6.2) is in progress on Station Battery 1P when the following information is determined:
    - A Float Charge in progress
    - Pilot Cell 42 has an Individual Cell Voltage (ICV) of 1.88 VDC
    - Electrolyte temperature is 85°F
    - Float current is 3 amps

    Based on this information what must the crew IMMEDIATELY do?

    A. Declare Battery 1P INOPERABLE.

    B. Raise battery charger DBC1P DC output voltage.

    C. Start an additional ventilation fan in the Battery 1P room.

    D. Perform a Battery Discharge Performance Test on Station Battery 1P.

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Proposed Answer: **A**

Explanation: Q48

A. Correct – With an ICV < 2.07 V or float current > 2 amps battery 1P must be declared INOPERABLE immediately per TS 3.8.6, Condition F.

B. Incorrect – Plausible as the applicant will have recognized that ICV is low, and that TS 3.8.6 requires correction in 24 hours most likely by performing an equalizing battery charge which includes raising charger output voltage.

C. Incorrect – Plausible as increasing ventilation is a requirement if hydrogen concentration or temperature in the battery compartment is out of specification.

D. Incorrect – Plausible as this is a required action if the applicant assesses that the battery shows signs of its battery capacity < 80%.

Technical Reference(s): TECHNICAL SPECIFICATION 3.8.6, BATTERY PARAMETERS; DB_DC INST POWER POWERPOINT (p3.8.6-1)

Reference(s) Provided: None

Learning Objective: OPS-SYS-409-08K

Question Source: Bank # _____

Modified Bank # _____ (note changes or attach parent)

New X

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge _____

Comprehension / Analysis / **H**igher Cognitive X

10 CFR Part 55 Content: 55.41 (b)(5)

55.43 _____

Comments:

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Examination Outline Cross-Reference: Q49

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	000064 K4.01	
Importance Rating	4.0	

K/A Statement: Knowledge of EMERGENCY DIESEL GENERATOR SYSTEM design features and/or interlocks that provide for the following: Trips while loading the EDG (frequency, voltage, and speed)

Proposed Question: Q49

49. Emergency Diesel Generator (EDG) 1 started and connected to its associated bus in response to a loss of offsite power:

    If protective trip _____ actuates, AC101, EDG 1 Output Breaker will OPEN AND the engine will shutdown.

    A. engine overspeed

    B. low lube oil pressure

    C. high jacket water temperature

    D. reverse power

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Proposed Answer: **A**

Explanation: Q49

A. Correct – The Engine Overspeed trip will open the output breaker and cause the engine to shutdown during an under voltage emergency start condition.

B. Incorrect – Plausible if the applicant does not recall that this trip is bypassed during emergency starts.

C. Incorrect – Plausible if the applicant does not recall that this trip is bypassed during emergency starts.

D. Incorrect – Plausible if the applicant does not recall that this trip is bypassed during emergency starts.

Technical Reference(s):	SD 03B R7, SYSTEM DESCRIPTION FOR EMERGENCY DIESEL GENERATORS AND STATION BLACKOUT DIESEL GENERATORS (p1-10, p2-21)	
Reference(s) Provided:	None	
Learning Objective:	OPS-SYS-406-12K	
Question Source:	Bank #	
	Modified Bank #	(note changes or attach parent)
	New	X
Question History:	Last NRC Exam	(if applicable)
Question Cognitive Level:	Memory / **F**undamental Knowledge	
	Comprehension / Analysis / **H**igher Cognitive	X
10 CFR Part 55 Content:	55.41	(b)(7)
	55.43	

Comments:

Examination Outline Cross-Reference: Q50

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	000073 A4.04	
Importance Rating	3.2	

K/A Statement: Ability to manually operate and/or monitor the PROCESS RADIATION MONITORING SYSTEM in the control room: Alarm and/or interlock setpoint checks and adjustments

Proposed Question: Q50

50. The BOP Reactor Operator inadvertently set the WARN Alarm setpoint higher than the HIGH Alarm setpoint on RI 1770A1, CLEAN WASTE SYSTEM OUTLET radiation monitor.

How will the individual be notified of this error?

A. The digits will flash when the HIGH pushbutton is depressed to check the HIGH Alarm setpoint.

B. Annunciator (7-1-B) CLN WST SYS OUT RAD HI will alarm after entering the HIGH Alarm setpoint.

C. An error will be displayed if setpoints are adjusted such that the High alarm set-point is lower than the Warn alarm set-point.

D. The WARN lamp will flash when the module is placed back into the panel.

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Proposed Answer: **C**

Explanation: Q50

A. Incorrect – Plausible since the digits normally flash when entering the setpoints.

B. Incorrect – Plausible since 7-1-B comes into alarm when doing the functional test.

C. Correct – IAW DB-OP-03011, Attachment 20 NOTE 2, an error message will be displayed if the HIGH Alarm setpoint is adjusted lower than the WARN Alarm setpoint.

D. Incorrect – Plausible as this is a valid indication for the RE1770A when the WARN(ALERT) Alarm setpoint has been exceeded. The module has to be partially pulled from the panel when setting the alarms.

Technical Reference(s): DB-OP-03011 R32, RADIOACTIVE LIQUID BATCH RELEASE (p168)
SD 017A, SYSTEM DESCRIPTION FOR PROCESS RADIATION MONITORS Table 2.5-2, T2.5-2-9 RE1770A

Reference(s) Provided: None

Learning Objective: OPS-SYS-111-05K

Question Source: Bank # _____

Modified Bank # _____ (note changes or attach parent)

New X

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge X

Comprehension / Analysis / **H**igher Cognitive ____

10 CFR Part 55 Content: 55.41 (b)(7)

55.43 _____

Comments:

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Examination Outline Cross-Reference: Q51

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	076 K3.04	
Importance Rating	2.7	

K/A Statement: Knowledge of the effect that a loss or malfunction of the SERVICE WATER SYSTEM will have on the following systems or system parameters: Turbine building CCW

Proposed Question: Q51

51. The Reactor is at 75% RTP with the following lineup:
    - Service Water (SW) Pumps 1 and 2 are in service
    - SW Pump 2 is supplying the secondary loads
    - Turbine Plant Cooling Water (TPCW) Heat Exchanger 1 and 2 are in service
    - TIC630, #1 TPCW HX TEMPERATURE CONTROLLER is in MANUAL, controlling at 85°F
    - TIC631, #2 TPCW HX TEMPERATURE CONTROLLER is in AUTO, set point at 85°F

    Subsequently,
    A crack develops in the sensing line for PSL 1377A, SERVICE WTR PMP 2 DISCH, resulting in pressure lowering to 47 psig on PSL 1377A.

    What is the expected plant response to these conditions?

    A. SW1395, SW HDR 2 TO TPCW HX, CLOSES.

    B. SW1399, SW HDR 1 TO TPCW HX, OPENS.

    C. SW1395 AND SW1399 receive CLOSE signals.

    D. SW630, TPCW HX 1 TEMPERATURE REG OUTLET fully OPENS.

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Proposed Answer: **A**

Explanation: Q51

A. Correct – SW 1395 is interlocked with and CLOSES when PSL 1377A drops below 48.5 psig (I&C Data package) to protect essential SW loads from a potential break in the non-essential SW piping.

B. Incorrect – Plausible if the applicant assesses that a loss of cooling to the secondary loads would be automatically accounted for by aligning SW train 1.

C. Incorrect – Plausible as SW1399 is interlocked with PSL1376A and would close if PSL1376A was also reading < 50 psig. PSL1376A is not affected by this leak and would be reading normally at ~100 psig.

D. Incorrect – Plausible as isolation of Service Water to the TPCW Heat Exchanger would cause temperatures in the TPCW system to rise and if TC630 was in AUTO, SW630 would OPEN to counter rising temperature.

Technical Reference(s): OS-020 SH1 R104, OS-020 SH2 R54 CL-6, DB-OP-06261 R75 (p23), SW SYSTEM OPERATING PROCEDURE (Note 3.1.9.e through 3.1.9.1)

SD-018, SW SYSTEM DESCRIPTION; SD-021, TPCW SYSTEM DESCRIPTION; DB-OP-02511, LOSS OF SW PUMPS/SYSTEMS; DB-OP-02514, TURBINE PLANT COOLING WATER MALFUNCTIONS; AND M-041A, B, & C, SW PIPING DIAGRAMS

Reference(s) Provided: None

Learning Objective: OPS-SYS-305-04K

Question Source: Bank # _____

Modified Bank # _____ (note changes or attach parent)

New X

Question History: Last NRC Exam N/A (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge _____

Comprehension / Analysis / **H**igher Cognitive X

10 CFR Part 55 Content: 55.41 (b)(7)

55.43 _____

Comments:

Examination Outline Cross-Reference: Q52

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	078 A3.03	
Importance Rating	2.9	

K/A Statement: Ability to monitor automatic features of the INSTRUMENT AIR SYSTEM, including: Air compressor loading/unloading

Proposed Question: Q52

52. The reactor is at 100% RTP with the following Instrument and Station Air Compressor (SAC) Configuration:
    - SAC-2 in Lead
    - SAC-1 in Standby
    - EIAC in Standby

    Subsequently, SAC 2 malfunctions and is running unloaded.

    If SAC 1 does NOT Auto Start, the EIAC should Auto Start when the EIAC Receiver air pressure lowers to _____.

    A. 100 psig

    B. 97 psig

    C. 95 psig

    D. 90 psig

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Proposed Answer: **C**

Explanation: Q52

A. Incorrect – EIAC Auto Starts at 95 psig. Plausible because SAC 1 Auto Starts at 100 psig.

B. Incorrect –The EIAC will NOT START until pressure in its receiver is 95 psig and lowering. Plausible since 9-3-E STA AIR HDR PRESS LO Alarm comes in at 97 psig.

C. Correct – Since the outlet piping of all 3 receivers is connected via normally OPEN valves, all 3 receivers would experience the drop in pressure when SAC 2 is running unloaded and SAC 1 fails to start. The EIAC will START when pressure in its receiver is 95 psig and lowering.

D. Incorrect – The EIAC will NOT START until pressure in its receiver is 95 psig and lowering. Plausible since SA valves begin to throttle close at 90 psig.

Technical Reference(s): DB-OP-02528 R28, LOSS OF INSTRUMENT AIR ABNORMAL OPERATING PROCEDURE (p125) Attachment 24, Background Information, standby SAC starts at 100psig, EIAC starts at 95 psig

SD-001, INSTRUMENT AIR SYSTEM DESCRIPTION; DB-OP-06251, STATION AND INSTRUMENT AIR SYSTEM OPERATING PROCEDURE; AND M-015A, C, & D, INSTRUMENT AIR PIPING DIAGRAMS

Reference(s) Provided: None

Learning Objective: OPS-SYS-602-08K

Question Source: Bank # _____

Modified Bank # _____ (note changes or attach parent)

New X

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge X

Comprehension / Analysis / **H**igher Cognitive _____

10 CFR Part 55 Content: 55.41 (b)(7)

55.43 _____

Comments:

Examination Outline Cross-Reference: Q53

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	078 A4.03	
Importance Rating	3.2	

K/A Statement: Ability to manually operate and/or monitor the INSTRUMENT AIR SYSTEM in the control room: Isolation/restoration of instrument air to isolated components/systems

Proposed Question: Q53

53. A reactor startup is in progress with power 35% RTP and all four Reactor Coolant Pumps (RCPs) in service.

Subsequently, a small earthquake (magnitude LESS THAN the Safe Shutdown Earthquake) causes leaks in the instrument air piping which results in a loss of ALL available air to the following components:

- MU3, LETDOWN ISOLATION VALVE
- MU32, MAKE-UP FLOW CONTROLLER
- MU38, RCP SEAL RETURN ISOLATION VALVE
- MU39, BATCH FLOW CONTROLLER
- SW1424, COMPONENT COOLING WATER (CCW) HEAT EXCHANGER 1-1 TEMPERATURE CONTROL VALVE

What is the expected component response to these malfunctions?

A. MU3 fails CLOSED isolating Letdown.

B. MU38 fails OPEN maintaining RCP Seal Return flow.

C. MU32 fails AS-IS maintaining current makeup flow to the RCS.

D. SW1424 fails AS-IS maintaining stable CCW system temperature.

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Proposed Answer: **A**

Explanation: Q53

A. Correct – MU3 fails CLOSED on a loss of instrument air isolating Letdown.

B. Incorrect – MU38 fails CLOSED on a loss of instrument air. Plausible as MU19 RCP Seal Injection Flow Controller fails open on a loss of instrument air.

C. Incorrect – MU32 fails OPEN on a loss of instrument air. Plausible since other valves lockup in their current position on certain failures.

D. Incorrect – SW1424 fails OPEN on a loss of instrument air. Plausible if the applicant does not remember failure mode of SW1424 and believes it is reasonable to maintain stable CCW system temperature.

Technical Reference(s): DB-OP-02528 R28, LOSS OF IA ABN OPERATING PROCEDURE. (p51) (p50 MU32, 51 MU3, 59 MU38,66 SW1424). OS-002 for MU&P valve failure positions.

SD-001, INSTRUMENT AIR, SD-018, SERVICE WATER.
SD-048, MU AND PURIFICATION SYSTEM DESCRIPTIONS.
RA-EP-02820, EARTHQUAKE EMERGENCY PLAN OFF NORMAL OCCURRENCE PROCEDURE.
DB-OP-06006, MU AND PURIFICATION SYSTEM PROCEDURE. DB-OP-06902, POWER OPERATIONS.
M-031A, B, & C, MAKE UP & PURIFICATION, M-041B, PRIMARY SERVICE WATER.
M-900I, IA PIPING DIAGRAMS PIPING DIAGRAMS

Reference(s) Provided: None

Learning Objective: OPS-SYS-602-08K

Question Source: Bank # _____

Modified Bank # _____ (note changes or attach parent)

New X

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge _____

Comprehension / Analysis / **H**igher Cognitive X

10 CFR Part 55 Content: 55.41 (b)(7)

55.43 _____

Comments:

Examination Outline Cross-Reference: Q54

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	103 K3.04	
Importance Rating	3.0	

K/A Statement: Knowledge of the effect that a loss or malfunction of the CONTAINMENT SYSTEM will have on the following systems or system parameters: Shield building vent system

Proposed Question: Q54

54. The reactor is at 100% RTP.

Fuel Handling is in progress.

The Zone 3 operator reports that one of the blowout panels in MPR 4 is damaged. Light can be seen through the damaged panel.

Which of the following ventilation system(s) is(are) impacted?

A. Station Emergency Ventilation System (EVS) Train 2 ONLY

B. Station EVS Trains 1 and 2

C. Aux Building Radwaste Ventilation System

D. Fuel Handling Ventilation System

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Proposed Answer: **B**

Explanation: Q54

A. Incorrect – Both trains of Station EVS draw on MPR 4. Plausible if the candidate believes the even numbered rooms go to the even numbered train of EVS.

B. Correct – The EVS trains are redundant to another. Each train has separate ductwork but both trains draw from Mechanical Penetration Room 4. Therefore, a leak on the blowout panel will impact the ability to draw the designed 0.25" vacuum between the annulus to atmosphere.

C. Incorrect – Aux Building Radwaste Ventilation does not draw a suction on MPR 4. Plausible since MPR 4 is considered a part of the auxiliary building.

D. Incorrect – Fuel Handling Ventilation does not draw off MPR 4. Plausible since fuel handling operations in the SFP are in progress and the SFP Area EVS system shares ductwork and fans with the Station EVS.

Technical Reference(s): OS-033D R16, EMERGENCY VENTILATION SYSTEM
SD-022C, EMERGENCY VENTILATION SYSTEM DESCRIPTION; Section 1.1, System Boundaries and Functions, 1.2.1 Design Requirements Section 2.1.2, pages 2-1 to 2-2, EVS Fan C30-1 started by RE8446 and EVS Fan C30-2 started by RE8447
DB-OP-00030 FUEL HANDLING OPERATIONS AND DB-OP-06904, SHUTDOWN OPERATIONS PROCEDURES; OS-034 SH.1&3, AUXILIARY BUILDING RADIOACTIVE HVAC SYSTEMS OPERATIONAL SCHEMATICS

Reference(s) Provided: None

Learning Objective:

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

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge ____
Comprehension / Analysis / **H**igher Cognitive X

10 CFR Part 55 Content: 55.41 (b)(7)
55.43 _____

Comments:

Examination Outline Cross-Reference: Q55

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	002 A4.03	
Importance Rating	4.0	

K/A Statement: Ability to manually operate and/or monitor the REACTOR COOLANT SYSTEM in the control room: Indications and controls necessary to recognize and correct saturation conditions

Proposed Question: Q55

55. A Reactor Trip has occurred from 100% RTP.

The following conditions are noted:
- SFAS Actuation - Levels 1-3
- Subcooling Margin indicates 0°F
- Incore Thermocouple temperatures are lowering and alternating between saturated and superheated conditions (approximately 10°F superheated)

All actions for DB-OP-02000, RPS, SFAS, SFRCS TRIP OR SG TUBE RUPTURE for these plant conditions have been completed.

Which of the following describes the primary mode of RCS cooling for these conditions?

A. Forced Circulation Cooling.

B. Single Phase Natural Circulation Cooling.

C. Conduction Cooling.

D. Boiler Condenser Cooling.

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Proposed Answer: **D**

Explanation: Q55

At Davis-Besse, the term Boiler Condenser is used vice Reflux. This knowledge is required to determine if the appropriate EOP Section to mitigate the event is Loss of Subcooling Margin or Inadequate Core Cooling. During Boiler Condenser Cooling, Incore thermocouples can trend slightly superheated, but that is not an indication that core cooling is inadequate. Direction to address the situation presented is provided by DB-OP-02000, Section 5, Loss of Subcooling Margin.

The following discussion of Boiler Condenser Cooling is provided by the Technical Bases Document Volume 3 (this is the generic EOP Bases for all B&W plants) "Boiler condenser cooling occurs when RC is boiled in the reactor core forming steam (removing core heat) which then flows through the hot leg piping to the SG where it condenses in the SG tubes. The condensed water then returns to the core by the cold leg piping. For the condensed water to flow back into the reactor core, the RC water level in the SG must be above the elevation of the RCP internal lip. This will provide the driving force to allow the water in the cold leg pipe to flow up and over the RCP discharge into the reactor vessel."

A. Incorrect – Plausible because the question does not provide the status of the RCPs. Forced circulation is the primary means of heat transfer following a reactor trip. Candidate should understand the EOP actions have been completed must know that RCPs are shutdown on a loss of SCM (SCM is less than 20°F subcooled), therefore Forced Circulation is incorrect.

B. Incorrect – Plausible because Single Phase Natural Circulation Cooling is the primary means of heat transfer following a reactor trip when RCPs are not available. Candidate should understand the EOP actions have been completed and must know that RCPs are shutdown on a loss of SCM (SCM is less than 20°F subcooled), therefore single-phase Natural Circulation is incorrect.

C. Incorrect – Plausible because conduction heat transfer from the RCS to Containment is occurring because the temperature of the RCS is higher than the temperature of Containment. Reflective insulation is installed on RCS piping to limit this heat transfer. While conductive heat transfer is occurring, it is not the primary mode of RCS cooling, therefore Conductive Cooling is incorrect.

D. Correct – With the RCS Saturated, Boiler condenser cooling occurs when RC is boiled in the reactor core forming steam (removing core heat) which then flows through the hot leg piping to the SG where it condenses in the SG tubes. The condensed water then returns to the core by the cold leg piping. For the conditions provided, boiling in the core is occurring (Subcooling Margin indicates 0°F). Lowering incore temperatures indicate cooling is occurring. For these plant conditions, that cooling would be boiler condenser cooling, therefore this response is the only correct response.

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Technical Reference(s):	DB-OP-02000 Bases and Deviation Document R22, Step 5.6. & 9.0 B&W Technical Bases Document Volume 3 R12, Page III.B.12 (page 168 of 862 on pdf viewer)		
Reference(s) Provided:	None		
Learning Objective:	OPS-GOP-300-09K		
Question Source:	Bank #	X	
	Modified Bank #		(note changes or attach parent)
	New		
Question History:	Last NRC Exam	2022 NRC Q4	(if applicable)
Question Cognitive Level:	Memory / **F**undamental Knowledge		
	Comprehension / Analysis / **H**igher Cognitive	X	
10 CFR Part 55 Content:	55.41	(b)(7)	
	55.43		

Comments:

Examination Outline Cross-Reference: Q56

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	014 K2.04	
Importance Rating	3.0	

K/A Statement: Knowledge of electrical power supplies to the following: ROD POSITION INDICATION SYSTEM: Rod position main control room display panel

Proposed Question: Q56

56. Reactor Power was operating at 100% RTP.

A loss of Off-Site power occurs causing the reactor to trip.

The CRD Position Indication Panel ____(1)____ indicate Relative Rod Position.

The individual 0 Percent lights will be ____(2)____ for Groups 1-7.

A. (1) will
   (2) ON

B. (1) will
   (2) OFF

C. (1) will NOT
   (2) ON

D. (1) will NOT
   (2) OFF

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Proposed Answer: **C**

Explanation: Q56

A. Incorrect – Part 1 is wrong. E2 and F2 are depowered when off-site power is lost. Plausible since Relative position now changes to 0 when the reactor is tripped. Part 2 is correct. YBU provides backup power to the 0% LED lights.

B. Incorrect – Part 1 is wrong. E2 and F2 are depowered when off-site power is lost. Plausible since Relative position now changes to 0 when the reactor is tripped. Part 2 is wrong. YBU provides backup power to the 0% LED lights. Plausible since redundant indications should come from different power supplies.

C. Correct – E2 and F2 are depowered when off-site power is lost. Relative position indication will be lost. YBU provides backup power to the 0% LED lights.

D. Incorrect – Plausible since Part 1 is correct, and the applicants are trained to determine the Immediate Action status of the reactor using Nuclear Instruments.

Technical Reference(s): DB-OP-02542 R17, LOSS OF YBU (p11)

Reference(s) Provided: None

Learning Objective:

Question Source: Bank # _____

Modified Bank # _____ (note changes or attach parent)

New X

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge _____

Comprehension / Analysis / **H**igher Cognitive X

10 CFR Part 55 Content: 55.41 (b)(7)

55.43 _____

Comments:

Examination Outline Cross-Reference: Q57

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	015 G2.2.2	
Importance Rating	4.6	

K/A Statement: NUCLEAR INSTRUMENTATION SYSTEM EQUIPMENT CONTROL: Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels

Proposed Question: Q57

57. The crew is performing DB-OP-06901, PLANT STARTUP and DB-OP-06912, APPROACH TO CRITICALITY.

While the Reactor Operator is withdrawing Group 6, the following plant parameters/conditions are observed:
- Rod Index is at 130 and rising
- Startup Rate rises to 1.0 decade per minute (DPM)
- Source Range Nuclear Instrument (NI) counts are 1400 and rising

The Reactor Operator STOPS withdrawing Regulating Group 6, and observes the following plant parameters/conditions:
- Rod Index is at 150 and steady
- Startup Rate is 0.8 decade per minute (DPM) and steady
- Source Range Nuclear Instrument (NI) counts are 1650 and rising

What action should the Reactor Operator take in response to the indicated plant parameters/conditions with NO Rod motion?

A. Immediately insert control rods to reduce the Startup Rate to less than 0.5 DPM.

B. Maintain current Rod Index until the Point of Adding Heat (POAH) is reached to stabilize power.

C. Continue WITHDRAWING Rods in small increments to re-establish a Startup Rate of 1.0 DPM.

D. TRIP the reactor because the Source Range NIs are not responding correctly.

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Proposed Answer: **A**

Explanation: Q57

A. Correct – The applicant must recognize that 0.8 DPM is too high and that rods must be inserted to lower DPM to 0.5 or less (per DB-OP-06912 precaution 2.1.2).

B. Incorrect – SUR is too high with no rod motion and that rods must be inserted to lower DPM to 0.5 or less (per DB-OP-06912 precaution 2.1.2). Plausible since the POAH is the next power level that the plant is raised to, after the Reactor becomes critical.

C. Incorrect – The maximum startup rate with no rod motion out is 0.5 decade per minute (DPM). The maximum startup rate during control rod withdrawal is 1.0 DPM. Plausible if the applicant forgets the SUR restrictions but believes rods must be withdrawn further to add reactivity to bring the reactor to the POAH.

D. Incorrect – The maximum startup rate with no rod motion out is 0.5 decade per minute (DPM). Plausible if the applicant believes the reactor should be tripped when the maximum SUR is exceeded.

Technical Reference(s): DB-OP-06912 R22, APPROACH TO CRITICALITY (p3) (Steps 2.1.2 and 2.1.3 p3)

SD-044, NUCLEAR INSTRUMENTATION/REACTOR PROTECTION SYSTEM DESCRIPTION; DB-OP-02505, NUCLEAR INSTRUMENTATION FAILURES; DB-OP-06403, REACTOR PROTECTION SYSTEM (RPS) AND NUCLEAR INSTRUMENTATION (NI) OPERATING PROCEDURE; DB-OP-06901, PLANT STARTUP;

Reference(s) Provided: None

Learning Objective:

Question Source: Bank # _____

Modified Bank # _____ (note changes or attach parent)

New X

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge _____

Comprehension / Analysis / **H**igher Cognitive X

10 CFR Part 55 Content: 55.41 (b)(6)

55.43 _____

Comments:

Examination Outline Cross-Reference: Q58

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	016 A1.01	
Importance Rating	3.2	

K/A Statement: Ability to predict and/or monitor changes in parameters associated with operation of the NON-NUCLEAR INSTRUMENTATION SYSTEM, including: Lights and alarms

Proposed Question: Q58

58. Steam Generator 1 Startup Range Level transmitter selected to ICS rapidly fails to 0 inches.

The following annunciators alarm:
- 14-4-E, ICS INPUT MISMATCH
- 14-4-F, ICS INPUT TRANSFER

The plant remains stable at 100% RTP.

The lights behind BOTH the “X” and “Y” pushbuttons should be ____(1)____.

Depressing the SASS Annunciator RESET pushbutton for the appropriate SASS module ____(2)____ , is required to reset Annunciator 14-4-E, ICS INPUT MISMATCH.

A. (1) ON
(2) AND the “Y” pushbutton

B. (1) ON
(2) ONLY

C. (1) OFF
(2) AND the “Y” pushbutton

D. (1) OFF
(2) ONLY

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Proposed Answer: **B**

Explanation: Q58

A. Incorrect – Plausible since part (1) is correct. The automatic transfer from the failed signal to the good signal is indicated to the operator both by the lights behind the pushbutton selector switches and by the station annunciator. Part (2) is wrong. Depressing just the SASS Annunciator RESET pushbutton for the appropriate SASS module will clear the Control Room Annunciator Panel. Part 2 is plausible since the Y instrument is now providing the input to ICS.

B. Correct – Part (1): The automatic transfer from the failed signal to the good signal is indicated to the operator both by the lights behind the pushbutton selector switches and by the station annunciator. Part (2): Depressing the SASS Annunciator RESET pushbutton for the appropriate SASS module will clear the Control Room Annunciator Panel Alarm.

C. Incorrect – Part (1) is wrong. It is plausible since the X instrument has failed and the Y light was initially off. Part (2) is wrong. See distractor "A" for explanation.

D. Incorrect - Part (1) is wrong. It is plausible since the X instrument has failed and the Y light was initially off. Part (2) is correct.

Technical Reference(s): DB-OP-02014 R17, MSR/ICS ALARM PANEL 14 ANNUNCIATORS (14-4-E, p45, 14-4-F 47), DB-OP-06407 R20, NON NUCLEAR INSTRUMENTATION SYSTEM OPERATING PROCEDURE (Attachment 10, pages 65-66)

Reference(s) Provided: None

Learning Objective:

Question Source: Bank # _____

Modified Bank # _____ (note changes or attach parent)

New X

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge ____

Comprehension / Analysis / **H**igher Cognitive X

10 CFR Part 55 Content: 55.41 (b)(5)

55.43 _____

Comments:

Examination Outline Cross-Reference: Q59

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	033 K5.06	
Importance Rating	3.7	

K/A Statement: Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the SPENT FUEL POOL COOLING SYSTEM: Shielding (water level)

Proposed Question: Q59

59. The plant has just completed a refueling outage and returned to 100% RTP.

Spent Fuel Pool (SFP) water level is at its normal level.

Subsequently, a significant non-isolable crack develops, resulting in a large leak on the SFP Cooling return line just prior to entering the SFP.

Assuming NO OPERATOR ACTION, how is the plant impacted?

SFP water level will stop lowering at approximately ____(1)____ feet.
Radiation levels on the SFP Bridge ____(2)____.

A. (1) 23
(2) DO NOT change

B. (1) 23
(2) will begin to rise

C. (1) 18
(2) DO NOT change

D. (1) 18
(2) will begin to rise

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Proposed Answer: **D**

Explanation: Q59

A. Incorrect – Plausible as the Technical Specifications require level to be at least 23 feet above the spent fuel assemblies. SFP water level WILL LOWER to 18 Feet (EL. 596'-0"). The SFP Pump will stop drawing a suction from the spent fuel pool when level lowers below 19 feet and the return line will stop contributing to the leak when level reaches 18 feet due to modification MOD 99-0051 which cut the return line at the 18 foot elevation to prevent a piping failure from draining the pool any further. Area radiation levels WILL RISE due to less water shielding. Plausible if the applicant believes that the design will maintain constant level in the SFP.

B. Incorrect – Plausible as the Technical Specifications require level to be at least 23 feet above the spent fuel assemblies. SFP water level WILL LOWER to 18 Feet (EL. 596'-0"). The SFP Pump will stop drawing a suction from the spent fuel pool when level lowers below 19 feet and the return line will stop contributing to the leak when level reaches 18 feet due to modification MOD 99-0051 which cut the return line at the 18 foot elevation to prevent a piping failure from draining the pool any further. Plausible because area radiation levels WILL RISE due to less water shielding and the applicant confuses the low water level alarm setpoint with the pump trip setpoint.

C. Incorrect – Plausible as Part (1) is correct. Plausible as due to 4'-6" less water above the fuel racks, area radiation levels WILL RISE. Area radiation levels WILL NOT rise to ARM alarm setpoints.

D. Correct – SFP water level WILL LOWER to 18 Feet (EL. 596'-0"). The SFP Pump will stop drawing a suction from the spent fuel pool when level lowers below 19 feet and the return line will stop contributing to the leak when level reaches 18 feet due to modification MOD 99-0051 which cut the return line at the 18 foot elevation to prevent a piping failure from draining the pool any further. Radiation levels WILL RISE due to less water shielding.

Technical Reference(s): DB-OP-02547 SPENT FUEL POOL COOLING MALFUNCTIONS; step 2.2.3, Attachment 1 SFP level schematic, M-035 SPENT FUEL POOL COOLING SYSTEM P&ID; OS-007 SPENT FUEL POOL COOLING SYSTEM OPERATIONAL SCHEMATIC; SD 024 SPENT FUEL POOL COOLING AND CLEANUP SYSTEM DESCRIPTION

Reference(s) Provided: None

Learning Objective:

Question Source: Bank # _____

Modified Bank # _____ (note changes or attach parent)

New X

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge _____

Comprehension / Analysis / **H**igher Cognitive X

10 CFR Part 55 Content: 55.41 (b)(5)

55.43 _____

Comments:

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Examination Outline Cross-Reference: Q60

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	034 K6.02	
Importance Rating	3.1	

K/A Statement: Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the FUEL HANDLING EQUIPMENT SYSTEM: RMS

Proposed Question: Q60

60. The plant is operating at 100% RTP.
    - Movement of fuel in the Spent Fuel Pool is in progress

    The following event occurs:
    - RE 8446, Fuel Handling Area Ventilation System Exhaust Radiation Monitor fails HIGH.

    How will the Fuel Handling Ventilation and the Emergency Ventilation Systems (EVS) respond to these conditions?

    Fuel Handling supply and exhaust fans ____(1)____.
    EVS Fan 1 ____(2)____.

    A. (1) stay running
       (2) starts automatically

    B. (1) TRIP
       (2) starts automatically

    C. (1) stay running
       (2) stays shutdown

    D. (1) TRIP
       (2) stays shutdown

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Proposed Answer: **B**

Explanation: Q60

A. Incorrect – Plausible since a trip of related RE 5403, Fuel Area Exhaust Monitor will not cause Fuel Handling Supply and Exhaust to shutdown.

B. Correct – IAW OS-0033D and OS-034 sh1 fuel handling supply and exhaust FANS TRIP and EVS starts automatically on a RE 8446 High radiation alarming condition.

C. Incorrect – Plausible since a trip of related RE 5403, Fuel Area Exhaust Monitor will not cause Fuel Handling Supply and Exhaust to shutdown and EVS would stay shutdown.

D. Incorrect – Plausible, in general process radiation monitors will cause realignment of the system and SU of the backup system. EVS is manually started for high radiation in other Aux building ventilation systems.

Technical Reference(s): DB-OP-02530 FUEL HANDLING ACCIDENT, Attachment 5, background information, page 2 of 3 last paragraph describes FH ventilation shutting down and EVS starting up. SD-022C, EMERGENCY VENTILATION SYSTEM DESCRIPTION Section 2.1.2, pages 2-1 to 2-2, EVS Fan C30-1 started by RE8446 and EVS Fan C30-2 started by RE8447.

Reference(s) Provided: None

Learning Objective:

Question Source: Bank # X

Modified Bank # _____ (note changes or attach parent)

New _____

Question History: Last NRC Exam NRC 2011 Q62 (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge X

Comprehension / Analysis / **H**igher Cognitive _____

10 CFR Part 55 Content: 55.41 (b)(7)

55.43 _____

Comments:

Davis-Besse 2024 NRC Written Exam

Examination Outline Cross-Reference: Q61

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	041 K1.09	
Importance Rating	3.0	

K/A Statement: Knowledge of the physical connections and/or cause and effect relationships between the STEAM DUMP/TURBINE BYPASS CONTROL SYSTEM and the following systems: Main Turbine / Generator system

Proposed Question: Q61

61. Initial Conditions:
    - The Reactor is operating at 22% RTP
    - Main Turbine Load is at 140 MWe
    - PIC ICS 12B (A), TURB BYPASS VALVES are in AUTO

    Subsequently the Main Turbine trips.
    - All systems respond as expected
    - Turbine header pressure is at 950 psig and lowering
    - SG pressures are at 970 psig and lowering

    Based on these conditions, the TBVs should _____

    A. remain CLOSED because 995 psig is the post reactor trip SG control pressure setpoint. The 115 psig bias is in effect.

    B. throttle OPEN because Turbine Header pressure is above its setpoint of 920 psig. The 50 psig bias is in effect.

    C. remain CLOSED because Header Pressure control will automatically transfer to the AVVs in this plant condition.

    D. throttle OPEN because Turbine Header pressure is above its setpoint of 870 psig. The 0 psig bias is in effect.

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Proposed Answer: **D**

Explanation: Q61

A. Incorrect – Plausible this would be true if the reactor and turbine tripped, the 115psi bias would be in effect so TBVs would be closed. With reactor power at 22% RTP, a turbine trip does not result in a reactor trip.

B. Incorrect – Plausible because with the reactor not tripped and the turbine NOT tripped > 92 MW of load a 50 psig bias is established so with header pressure greater than setpoint TBVs would be open. Incorrect since the turbine tripped.

C. Incorrect – Plausible since during some plant conditions pressure control will transfer from the TBVs to the AVVs

D. Correct – With power initially at 22% RTP, a turbine trip will not result in a reactor trip. With the reactor operating and the turbine tripped, the TBV bias goes to 0 psi so the TBVs will open to establish Turbine Header pressure at the setpoint of 870 psig.

Technical Reference(s): SD-012A, SYSTEM DESCRIPTION FOR MAIN STEAM SYSTEM.
DB-OP-06401 R31 Attachment 9 for PC ICS10 operation (p114)
Attachment 10 for TBV operation and bias.
Crib Notes for Main Steam System
DB-OP-06401 R31 Attachment 9 (p114)

Reference(s) Provided: None

Learning Objective:

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

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge _____
Comprehension / Analysis / **H**igher Cognitive X

10 CFR Part 55 Content: 55.41 (b)(2 to 9)
55.43 _____

Comments:

Examination Outline Cross-Reference: Q62

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	050 K6.03	
Importance Rating	2.7	

K/A Statement: Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the CONTROL ROOM VENTILATION: Plant pneumatic system

Proposed Question: Q62

62. The Reactor is operating at 100% RTP when the following annunciators alarm and indications are observed:
    - 9-3-E, STA AIR HDR PRESS LO is LIT
    - 9-1-F, INSTR AIR HDR PRESS LO is LIT
    - PI 810, INSTRUMENT AIR HEADER PRESSURE indication rapidly lowering
    - PI 811, STATION AIR HEADER PRESSURE indication rapidly lowering

    The crew entered DB-OP-02528, INSTRUMENT AIR SYSTEM MALFUNCTIONS due to a severe loss of Instrument Air.

    Based on these conditions, the Control Room **Normal** Ventilation system will _____(1)_____.

    If a subsequent small break LOCA occurs and containment pressure reaches 19.0 psia, with NO operator action the Control Room **Emergency** Ventilation system (CREVS) will ___(2)___.

A. (1) shutdown
   (2) remain OFF

B. (1) remain ON
   (2) remain OFF

C. (1) shutdown
   (2) AUTO start

D. (1) remain ON
   (2) AUTO start

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Proposed Answer: **A**

Explanation: Q62

A. Correct – The CTRM Normal Ventilation (CRNV) system will isolate and stop operating when instrument air is lost. Specifically, Control Room isolation dampers 5301A-H and 5311A-H will fail closed which will result in the CRNV fans tripping. The CREV system requires operators to manually start and align. The dampers associated with the CREV are motor operated and would not be affected by the loss of instrument air. The high containment pressure signal caused by the LOCA would give the CRNV an isolation signal (which has already occurred due to the loss of instrument air) but does not automatically start the CREV system.

B. Incorrect – Plausible as the CREV system dampers are motor operated, but the CRNV are pneumatic and fail closed on a loss of instrument air. Part 2 is correct.

C. Incorrect – Plausible as Part 1 is correct and an SFAS Level 3 signal (containment pressure > 18.7 psia) results in numerous automatic actions including an isolation of the CRNV but it does not auto align and start CREV which is unaffected by the loss of instrument air.

D. Incorrect – Plausible as the CREV system dampers are motor operated, but the CRNV are pneumatic and fail closed on a loss of instrument air. Part 2 is plausible as an SFAS Level 3 signal (containment pressure > 18.7 psia) results in numerous automatic actions including an isolation of the CRNV but it does not auto align and start CREV which is unaffected by the loss of instrument air.

Technical Reference(s): SD 029A, SYSTEM DESCRIPTION FOR CONTROL ROOM NORMAL VENTILATION; SD 029B, SYSTEM DESCRIPTION FOR CONTROL ROOM EMERGENCY VENTILATION; DB-OP-02528, INSTRUMENT AIR SYSTEM MALFUNCTIONS Attachment 15, HVAC system actions, Note 1, all non-safety ventilation shuts down on loss of IA; TECHNICAL SPECIFICATIONS BASES B3.7.10, CONTROL ROOM EMERGENCY VENTILATION SYSTEM (CREVS) states that CREVS is manually started. P&ID 32A CL 2 & 3

Reference(s) Provided: None

Learning Objective:

Question Source: Bank # _____

Modified Bank # _____ (note changes or attach parent)

New X

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge _____

Comprehension / Analysis / **H**igher Cognitive X

10 CFR Part 55 Content: 55.41 (b)(7)

55.43 _____

Comments:

Examination Outline Cross-Reference: Q63

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	055 K4.01	
Importance Rating	3.3	

K/A Statement: Knowledge of CONDENSER AIR REMOVAL SYSTEM design features and/or interlocks that provide for the following: Draw main condenser vacuum

Proposed Question: Q63

63. The Reactor is operating at 75% RTP when the following conditions occur:
    - Air intrusion results in condenser pressure rising to 5” HgA.
    - The crew has entered and is performing the actions of DB-OP-02518, HIGH CONDENSER PRESSURE.

    Based on these conditions, the Condenser Vacuum system will AUTOMATICALLY _____.

    A. open AS1933, STEAM SUPPLY STOP VALVE TO STEAM HOGGER to place the Steam Hogger on-line

    B. open the steam supply, suction, and discharge valves for SJAE #2

    C. start the Mechanical Hogger and its cooling water support system

    D. open pressure control valve VS1061, VACUUM CONTROL

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Proposed Answer: **C**

Explanation: Q63

A. Incorrect – Plausible as the steam hogger is the preferred system for initially drawing condenser vacuum during a plant start. It is not used on-line. AS1933 has an auto closure signal when the station vent VS926 and VS926A are closed or if exhaust vent pressure > 4 psig but no auto open signal.

B. Incorrect – Plausible as a supplemental manual action of DB-OP-02518 is to reconfigure the SJAE lineup to support improving vacuum conditions.

C. Correct – With the Mechanical Hogger in standby (control switch in neutral) it receives an auto start signal along with its cooling water system when condenser pressure rises to 4.5" HgA.

D. Incorrect – Plausible as PCV VS1061 helps control condenser pressure in Auto by opening to allow air intrusion if pressure is too low which could cause condenser tube vibrations.

Technical Reference(s): SD 026B, SYSTEM DESCRIPTION FOR CONDENSER VACUUM SYSTEM; DB-OP-02518, HIGH CONDENSER PRESSURE Step 4.3 is to verify the mechanical hogger starts, Attachment 1 manually checks SJAE and closes PCV1061; DB-OP-06231, VACUUM SYSTEM, Crib Notes for Vacuum System

Reference(s) Provided: None

Learning Objective:

Question Source: Bank # _____

Modified Bank # _____ (note changes or attach parent)

New X

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge ____

Comprehension / Analysis / **H**igher Cognitive X

10 CFR Part 55 Content: 55.41 (b)(7)

55.43 _____

Comments:

Examination Outline Cross-Reference: Q64

Level	RO	SRO
Tier #	3	
Group #		
K/A #	G2.1.17	
Importance Rating	3.9	

K/A Statement: CONDUCT OF OPERATIONS: Ability to make accurate, clear, and concise verbal reports

Proposed Question: Q64

64. You have just performed a turnover assuming the At-the-Controls Operator (ATC) role.

In accordance with NOP-OP-1002, CONDUCT OF OPERATIONS the _____ MUST be verbally informed.

A. Zone 3 Equipment Operator

B. Balance of Plant (BOP) Reactor Operator

C. Command SRO

D. Shift Manager

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Proposed Answer: **C**

Explanation: Q64

A. Incorrect – Plausible as the RO ATC gives specific verbal commands to Equipment Operators during normal shift operations.

B. Incorrect – Plausible as the RO ATC and BOP share the control room as a watch station and regularly communicate with each other.

C. Correct – NOP-OP-1002, Step 4.1.8.4 states, “The Command SRO is verbally informed that the RO ATC function has been assumed.

D. Incorrect – Plausible as numerous reports are required to be made to the Shift Manager by members of the operating crew.

Technical Reference(s): NOP-OP-1002, CONDUCT OF OPERATIONS 4.1.4.6

Reference(s) Provided: None

Learning Objective:

Question Source: Bank # _____

Modified Bank # _____ (note changes or attach parent)

New X

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge X

Comprehension / Analysis / **H**igher Cognitive ____

10 CFR Part 55 Content: 55.41 (b)(10)

55.43 _____

Comments:

Examination Outline Cross-Reference: Q65

Level	RO	SRO
Tier #	3	
Group #		
K/A #	G2.1.15	
Importance Rating	2.7	

K/A Statement: CONDUCT OF OPERATIONS: Knowledge of administrative requirements for temporary management direction, such as standing orders, night orders, or operations memoranda.

Proposed Question: Q65

65. In accordance with NOP-OP-1002, CONDUCT OF OPERATIONS, Night Orders are valid for a MAXIMUM of _____ day(s).

A. 1

B. 14

C. 30

D. 90

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Proposed Answer: **B**

Explanation: Q65

A. Incorrect – Plausible if the term Night Orders means that the information is only valid until the next day.

B. Correct – IAW NOP-OP-1002, Night Orders are for short-term communication only and are only valid for 14 days.

C. Incorrect – Plausible because NOP-OP-1002 discusses material deficiency tags which are required to be audited monthly.

D. Incorrect – Plausible because Standing Orders are reviewed on a Quarterly basis.

Technical Reference(s): NOP-OP-1002, CONDUCT OF OPERATIONS Step 4.15.2.2

Reference(s) Provided: None

Learning Objective:

Question Source: Bank # X

Modified Bank # _____ (note changes or attach parent)

New _____

Question History: Last NRC Exam DB 2018 Q67 (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge X

Comprehension / Analysis / **H**igher Cognitive _____

10 CFR Part 55 Content: 55.41 (b)(10)

55.43 _____

Comments:

Examination Outline Cross-Reference: Q66

Level	RO	SRO
Tier #	3	
Group #		
K/A #	G2.2.6	
Importance Rating	3.0	

K/A Statement: Knowledge of the process for making changes to procedures.

Proposed Question: Q66

66. IAW the guidance provided in NOP-SS-3001, PROCEDURE REVIEW AND APPROVAL, which of the procedure changes, listed below, MAY be made by using the Procedure Correction method?

(1) Changes to setpoints
(2) Changing component location
(3) Changes to equipment position
(4) Correcting Table of Contents
(5) Correcting typographical errors
(6) Changing the purpose of the procedure

A. 1, 2, & 4

B. 2, 4, & 5

C. 1, 3, & 6

D. 3, 5, & 6

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Proposed Answer: **B**

Explanation: Q66

A. Incorrect –
Item 1: Changes to setpoints and equipment position shall not be made using the Procedure Correction method. Plausible as Items 2 & 4 are allowed.

B. Correct –
Item 2: Changing component location,
Item 4; Correcting Table of Contents, and
Item 5: Correcting editorial errors such as typographical errors are all allowed.

C. Incorrect –
Items 1 & 3 Changes to setpoints and equipment position shall not be made using the Procedure Correction method.
Plausible as
Item 6, Changing the purpose of the procedure, is a "significant change" as defined by NOP-SS-3001 but may be confused with changing the title of a procedure which is allowed by the Procedure Correction process.

D. Incorrect –
Item 3: Changes to equipment position shall not be made using the Procedure Correction method.
Plausible as
Item 6, Changing the purpose of the procedure, is a "significant change" as defined by NOP-SS-3001 but may be confused with changing the title of a procedure which is allowed by the Procedure Correction process, and
Item 5: Correcting editorial errors such as typographical errors is allowed.

Technical Reference(s): NOP-SS-3001, PROCEDURE REVIEW AND APPROVAL Step 4.2.1.3, Procedure Correction.

Reference(s) Provided: None

Learning Objective:

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

Question History: Last NRC Exam DB 2011 Q68 (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge X
Comprehension / Analysis / **H**igher Cognitive _____

10 CFR Part 55 Content: 55.41 (b)(10)
55.43 _____

Comments:

Examination Outline Cross-Reference: Q67

Level	RO	SRO
Tier #	3	
Group #		
K/A #	G2.2.43	
Importance Rating	3.0	

K/A Statement: EQUIPMENT CONTROL: Knowledge of the process used to track inoperable alarms

Proposed Question: Q67

67. The Reactor is in MODE 1 with the following condition:
    - 3-1-C, BWST LVL LO annunciator is repeatedly alarming and clearing.
    - Maintenance has informed operations that the annunciator is repeatedly alarming due to a circuit fault.
    - Due to the nature of the fault, maintenance has reported that it will be an extended period of time until repairs can be made.

    In accordance with NOP-OP-1002, CONDUCT OF OPERATIONS annunciator 3-1-C ____(1)____ be disabled.

    Compensatory measures and increased monitoring for nuisance alarm annunciator 3-1-C shall be documented in the ____(2)____.

    A. (1) should
       (2) Unit Narrative Log

    B. (1) should NOT
       (2) Unit Narrative Log

    C. (1) should
       (2) Inoperable Equipment Tracking Log

    D. (1) should NOT
       (2) Inoperable Equipment Tracking Log

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Proposed Answer: **A**

Explanation: Q67

A. Correct – Per NOP-OP-1002, this annunciator should be considered a nuisance alarm as it is being caused by a faulted circuit and not an actual condition. Since the nuisance alarm will take an extended period to correct it SHOULD be disabled. The additional monitoring and compensatory measures taken because the nuisance alarm exists whether it is disabled or not are to be recorded in the Control Room Narrative Logs.

B. Incorrect – Plausible because this would be true if the nuisance alarm were expected to be repaired in a short period of time the alarm SHOULD NOT be disabled. Part (2) is correct.

C. Incorrect – Plausible as Part (1) is correct. Part (2) is plausible as this is the log called out in nop-op-1002 and DB-OP-00018, INOPERBALE EQUIPMENT TRACKING LOG to track equipment required to be operable or functional to meet the requirements of the Technical Specifications, TRM, and ODCM.

D. Incorrect – Plausible because this would be true if the nuisance alarm were expected to be repaired in a short period of time the alarm SHOULD NOT be disabled. Part (2) is plausible as this is the log called out in nop-op-1002 and DB-OP-00018, INOPERBALE EQUIPMENT TRACKING LOG to track equipment required to be operable or functional to meet the requirements of the Technical Specifications, TRM, and ODCM.

Technical Reference(s): NOP-OP-1002, CONDUCT OF OPERATIONS Step 4.9.2.14 for definition of nuisance alarms; 4.9.2.18 for documenting in the Narrative log; DB-OP-00018, INOPERBALE EQUIPMENT TRACKING LOG

Reference(s) Provided: None

Learning Objective:

Question Source: Bank # _____

Modified Bank # _____ (note changes or attach parent)

New X

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge X

Comprehension / Analysis / **H**igher Cognitive ____

10 CFR Part 55 Content: 55.41 (b)(10)

55.43 _____

Comments:

Examination Outline Cross-Reference: Q68

Level	RO	SRO
Tier #	3	
Group #		
K/A #	G2.3.5	
Importance Rating	2.9	

K/A Statement: RADIATION CONTROL: Ability to use RMSs, such as fixed radiation monitors and alarms or personnel monitoring equipment

Proposed Question: Q68

68. The Reactor is at 50% RTP.

The Reactor Operator selects the GROSS mode on RE609, Main Steam Line 1 Radiation Monitor.

Operating the radiation monitor in the GROSS mode will result in displayed count rate _____.

A. rising due to the extended band of isotopes the RE would detect

B. rising due to the detector saturating in the elevated radiation field

C. lowering due to lower detector sensitivity to N-16 gammas

D. lowering since GROSS mode only indicates when the reactor is shutdown

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Proposed Answer: **A**

Explanation: Q68

A. Correct – In the analyze mode the detector is calibrated for N16 gammas. The Gross mode allows a greater band of isotopes to be detected therefore the displayed output will be greater.
B. Incorrect – Plausible since the detector could saturate but this would cause the count rate to lower.
C. Incorrect – Plausible since N16 gammas are what the analyze mode is calibrated for.
D. Incorrect – Plausible since this is when the detector is procedurally placed in the GROSS mode.

Technical Reference(s): BASES AND DEVIATION DOCUMENT FOR DB-OP-02000, Rev 22. Steps 4.12 and 4.13 p36, verify RE609 in GROSS.

Reference(s) Provided: None

Learning Objective:

Question Source: Bank # X

Modified Bank # _____ (note changes or attach parent)

New _____

Question History: Last NRC Exam DB 2022 Q50 (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge X

Comprehension / Analysis / **H**igher Cognitive ____

10 CFR Part 55 Content: 55.41 (b)(11) & (12)

55.43 _____

Comments:

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Examination Outline Cross-Reference: Q69

Level	RO	SRO
Tier #	3	
Group #		
K/A #	G2.4.26	
Importance Rating	3.1	

K/A Statement: Knowledge of facility protection requirements, including fire brigade and portable firefighting equipment usage.

Proposed Question: Q69

69. Reactor power is at 75% RTP.

All 3 Condensate Pumps are running.

Per DB-OP-02529, FIRE PROCEDURE, which of the following conditions requires IMMEDIATE activation of the Fire Brigade?

A. A small brush fire near the switchyard fence.

B. Simultaneous fire alarms received in physically separate regions of the plant.

C. A trip of Condensate Pump 2 and a report of a slight electrical odor in High Voltage Switchgear Room 2.

D. A single Fire Detection Zone alarm and a simultaneous start of a Fire Pump.

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Proposed Answer: **D**

Explanation: Q69

A. Incorrect – Plausible as a fire outside the Protected Area which affects plant equipment is an entry condition for DB-OP-02529. In addition, the Fire Brigade could be activated in this instance, but it would require Shift Manager judgement and direction which was not indicated in the question stem or answer choice.

B. Incorrect – Plausible as the fire brigade must be immediately activated if multiple fire alarms are received in ADJACENT plant locations.

C. Incorrect – Plausible as the fire brigade must be activated if there is a fire in an area that has safe shutdown equipment that results in Spurious actuation of safe shutdown equipment. Incorrect since B Bus nor Condensate Pump 2 are safe shutdown equipment.

D. Correct – DB-OP-02529, FIRE PROCEDURE, step 2.2 directs activating the fire brigade IF a single Fire Detection Zone or Fire Suppression Area alarm is received AND a simultaneous start of a Fire Pump occurs.

Technical Reference(s): DB-OP-02529, FIRE PROCEDURE Step 2.2

Reference(s) Provided: None

Learning Objective:

Question Source: Bank # X

Modified Bank # _____ (note changes or attach parent)

New _____

Question History: Last NRC Exam DB 2016 Q75 (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge X

Comprehension / Analysis / **H**igher Cognitive _____

10 CFR Part 55 Content: 55.41 (b)(10)

55.43 _____

Comments:

K/A Change.

Examination Outline Cross-Reference: Q70

Level	RO	SRO
Tier #	4	
Group #		
K/A #	192006 K1.13	
Importance Rating		3.0

K/A Statement: FISSION PRODUCT POISONS: Plot the curve and explain the reasoning for the reactivity insertion by xenon-135 versus time for the following: reactor shutdown

Proposed Question: Q70

70. The reactor is shutdown after sustained power operations at 100% RTP.

Explain the shape and reason for the plot of Xe^{135} concentration in the reactor core at:

- 4 hours after shutdown
- 24 hours after shutdown

A. INCREASING in BOTH instances due to the decay of I^{135}.

B. DECREASING in BOTH instances since Xe^{135} production from fission has stopped.

C. INCREASING at 4 hours due to the decay of I^{135} and DECREASING at 24 hours since Xe^{135} production from fission has stopped.

D. DECREASING at 4 hours since Xe^{135} production from fission has stopped and INCREASING at 24 hours due to the decay of I^{135}.

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Proposed Answer: **C**

Explanation: Q70

A. Incorrect – Xe135 concentration will rise sharply for the first approximately 10 hours after a shutdown from 100% power as gains due to I135 decay will greatly outweigh Xe135 losses since I135 decay to Xe135 has a shorter half-life than Xe135 decay and since neutron absorption by Xe135 has greatly lowered with neutron flux greatly lowered post shutdown. After 10 hours Xe^{135} concentration will lower as production from fission of I135 has stopped.

B. Incorrect – Plausible as Xe^{135} concentration will decrease for this reason after 10 hours following shutdown.

C. Correct – Xe135 concentration will rise sharply for the first approximately 10 hours after a shutdown from 100% power as gains due to I135 decay will greatly outweigh Xe135 losses since I135 decay to Xe135 has a shorter half-life than Xe135 decay and since neutron absorption by Xe135 has greatly lowered with neutron flux greatly lowered post shutdown. After 10 hours Xe135 concentration will lower as production from fission of I135 has stopped.

D. Incorrect – Plausible as concentration will eventually decrease due to loss of production from fission, but not until concentration peaks at 10 hours.

Technical Reference(s): GFES

Reference(s) Provided: None

Learning Objective:

Question Source: Bank # _____

Modified Bank # X (note changes or attach parent)

New _____

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge X

Comprehension / Analysis / **H**igher Cognitive _____

10 CFR Part 55 Content: 55.41 (b)(1)

55.43 _____

Comments:

Modified from PWR GFES Exam Bank Question P59.

Examination Outline Cross-Reference: Q71

Level	RO	SRO
Tier #	4	
Group #		
K/A #	192007 K1.05	
Importance Rating	3.2	

K/A Statement: FUEL DEPLETION AND BURNABLE POISONS: Describe the effects of boration/dilution on reactivity during forced-flow and natural circulation conditions

Proposed Question: Q71

71. The Reactor is in MODE 4. The crew is performing DB-OP-06001, BORON CONCENTRATION CONTROL.

Compared to adding boric acid to the reactor coolant system (RCS) during forced circulation, adding boric acid during natural circulation requires ____(1)____ time to achieve complete mixing in the RCS.

After complete mixing occurs, a 1 ppm increase in RCS boron concentration during natural circulation will cause a/an ____(2)____ change in reactivity for a given reactor coolant temperature.

A. (1) more
(2) smaller

B. (1) more
(2) equal

C. (1) less
(2) smaller

D. (1) less
(2) equal

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Proposed Answer: **B**

Explanation: Q71

A. Incorrect – Plausible as part 1 of the question is correct. Part 2 is plausible if the applicant confuses the effect of a lower RCS flow rate on heat transfer with impact on reactivity.

B. Correct – With at least 1 RCP running or DHR pump operating in shutdown cooling mode of operation boron that is added to the RCS will more quickly achieve complete mixing as compared to the lower flow rates achieved when in natural circulation. Once complete mixing has occurred whether in natural circulation or in forced convection mode the reactivity effect of adding boron is only affected by moderator temperature coefficient and fuel doppler coefficient of reactivity which are a product of time in core life (fuel and burnable poison depletion).

C. Incorrect – Part 1 is plausible if the applicant believes that the laminar flow characteristics of natural circulation will more quickly achieve complete mixing of the added boric acid as compared to the turbulent flow of forced convection. Part 2 is plausible if the applicant confuses the effect of a lower RCS flow rate on heat transfer with impact on reactivity.

D. Incorrect – Part 1 is plausible if the applicant believes that the laminar flow characteristics of natural circulation will more quickly achieve complete mixing of the added boric acid as compared to the turbulent flow of forced convection. Plausible as part 2 of the question is correct.

Technical Reference(s): DB-OP-06001, BORON CONCENTRATION CONTROL, GFES

Reference(s) Provided: None

Learning Objective:

Question Source: Bank # X

Modified Bank # ____ (note changes or attach parent)

New ____

Question History: Last NRC Exam ____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge ____

Comprehension / Analysis / **H**igher Cognitive X

10 CFR Part 55 Content: 55.41 (b)(1)

55.43 ____

Comments:

From GFES Bank November 2020 PWR Exam Question ID P3364.

Examination Outline Cross-Reference: Q72

Level	RO	SRO
Tier #	4	
Group #		
K/A #	192008 K1.03	
Importance Rating	4.0	

K/A Statement: REACTOR OPERATIONAL PHYSICS: (STARTUP AND APPROACH TO CRITICALITY) Describe count rate and instrument response that should be observed for rod withdrawal during the approach to criticality

Proposed Question: Q72

72. A reactor startup is in progress in accordance with DB-OP-06901, PLANT STARTUP with the crew taking actions informed by DB-OP-06912, APPROACH TO CRITICALITY.

The reactor has an estimated initial K_{eff} of 0.8 with a stable source range count rate of 100 cps as indicated on NI5874A.

If control rods are withdrawn until K_{eff} equals 0.95, at what value will the count rate stabilize on NI5874A?

A. 150 cps

B. 200 cps

C. 250 cps

D. 400 cps

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Proposed Answer: **D**

Explanation: Q72

A. Incorrect – Plausible as K_{eff} changed by 0.15 which the applicant could mistake for counts now reading 150 cps.

B. Incorrect – Plausible as the applicant will be attempting to assess count rate doubling during the approach to criticality.

C. Incorrect – Plausible since K_{eff} changed by .15 which could be assessed as counts increasing by 150% to 250 cps.

D. Correct – The count rate for a subcritical reactor is determined by the inverse of 1-k_{eff}. The count rate ratio is equivalent to the inverse ratio of 1-k_{eff} which leads to a new stable value of 400 cps.

Technical Reference(s): DB-OP-06901, PLANT STARTUP; DB-OP-06912, APPROACH TO CRITICALITY

Reference(s) Provided: None

Learning Objective

Question Source: Bank # X

Modified Bank # _____ (note changes or attach parent)

New _____

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge X

Comprehension / Analysis / **H**igher Cognitive _____

10 CFR Part 55 Content: 55.41 (b)(1)

55.43 _____

Comments:

From PWR GFES bank question QID P848.

Examination Outline Cross-Reference: Q73

Level	RO	SRO
Tier #	4	
Group #		
K/A #	193003 K1.08	
Importance Rating	2.8	

K/A Statement: STEAM: Define the following terms: saturated liquid

Proposed Question: Q73

73. The pressurizer is operating in a saturated condition at 656°F.

If a sudden 10 percent liquid out surge occurs,

Pressurizer pressure will ____(1)____; and
Pressurizer temperature will ____(2)____.

A. (1) remain the same
(2) lower

B. (1) remain the same
(2) remain the same

C. (1) lower
(2) lower

D. (1) lower
(2) remain the same

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Proposed Answer: **C**

Explanation: Q73

A. Incorrect – Part (1) is plausible if the applicant believes the heaters and current saturation conditions will maintain pressure the same. Part (2) is correct.

B. Incorrect – Part (1) is plausible if the applicant believes the heaters and current saturation conditions will maintain pressure the same. Part (2) is plausible since pressure and temperature should follow each other.

C. Correct –Pressure will lower when the out surge occurs, since the PZR is a saturated system temperature will lower also.

D. Incorrect – Part (1) is correct. Part (2) is plausible if the applicant believes the heaters will maintain PZR Temp during the out surge.

Technical Reference(s): SD-0039A, RC System

Reference(s) Provided: **None**

Learning Objective:

Question Source: Bank # X

Modified Bank # ____ (note changes or attach parent)

New ____

Question History: Last NRC Exam ____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge ____

Comprehension / Analysis / **H**igher Cognitive X

10 CFR Part 55 Content: 55.41 (b)(14)

55.43 ____

Comments: From GFES PWR Bank QID P774

Examination Outline Cross-Reference: Q74

Level	RO	SRO
Tier #	4	
Group #		
K/A #	193004 K1.15	
Importance Rating	2.8	

K/A Statement: THERMODYNAMIC PROCESS: (THROTTLING AND THE THROTTLING PROCESS) Determine the exit conditions for a throttling process based on the use of steam and/or water

Proposed Question: Q74

74. Plant Conditions:
- RCS pressure is 2185 psig
- Pressurizer Temperature is 650°F
- Pressurizer Quench Tank pressure is 10 psig
- Pressurizer PORV, RC2A is leaking

Based on these conditions, what do you expect the PORV leak detection temperature instrument TE RC12-1 to be reading?

A. 140°F

B. 193°F

C. 240°F

D. 300°F

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Proposed Answer: **C**

Explanation: Q74

A. Incorrect – Plausible if the applicant determines that the conditions downstream of the PORV remain constant as a saturated vapor at 1122 Btu/lbm. Therefore, pressure would be 4 psia and Tsat at 140F.

B. Incorrect – Plausible if the applicant uses 10 psia when assessing the steam tables instead of 10 psig when determining saturation conditions downstream of the PORV.

C. Correct – Throttling is an isenthalpic process. The applicant should determine the enthalpy of saturated vapor at 2185 psig (2200 psia) which is 1122 Btu/lbm. The pressure in the Quench Tank is 10 psig (25 psia). Since the enthalpy remains constant it can be seen that at 10 psig 1122 is in between the enthalpy of saturated liquid and saturated vapor therefore saturation conditions exist. Therefore, the temperature would be 240ºF.

D. Incorrect – Plausible if the applicant believes temperature in the line will be between saturation conditions in the pressurizer and in the quench tank.

Technical Reference(s): SD 039A, SYSTEM DESCRIPTION FOR REACTOR COOLANT SYSTEM; SD 039B, SYSTEM DESCRIPTION FOR REACTOR COOLANT SYSTEM ACCESSORY SYSTEMS

Reference(s) Provided: **STEAM TABLES**

Learning Objective:

Question Source: Bank # X

Modified Bank # _____ (note changes or attach parent)

New _____

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge _____

Comprehension / Analysis / **H**igher Cognitive X

10 CFR Part 55 Content: 55.41 (b)(14)

55.43 _____

Comments: GFES PWR Exam Bank P76.

Examination Outline Cross-Reference: Q75

Level	RO	SRO
Tier #	4	
Group #		
K/A #	193010 K1.06	
Importance Rating	3.8	

K/A Statement: BRITTLE FRACTURE AND VESSEL THERMAL STRESS: Define PTS

Proposed Question: Q75

75. The plant was operating at 100% RTP when a loss of Off-Site power occurred.

Which of the following events would be MOST likely to cause a pressurized thermal shock (PTS) to the reactor vessel?

A. Starting an RCP in an idle RCS loop with the associated SG temperature less than the loop temperature after Off-Site power is restored.

B. Exceeding the 50°F/hr heat up rate limits post trip due to a loss of heat transfer event.

C. Maintaining full MU/HPI flow during and after a complete and unisolable rupture of a reactor vessel hot leg nozzle.

D. Maintaining full MU/HPI flow during and after a complete and unisolable Main Steam line rupture in CTMT.

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Proposed Answer: **D**

Explanation: Q75

A. Incorrect – Plausible as starting an RCP raises loop pressure.

B. Incorrect – Per DB-OP-02000, Specific Rule 5 PTS is a concern post trip when RCS temperature is < 360F and cooldown rate exceeds the PTLR limits. This is plausible as heatup rate is exceeded in this event.

C. Incorrect – Plausible as a Large LOCA will cause an RCS cooldown, but the RCS will be depressurized in this instance.

D. Correct – An overcooling event has occurred which will significantly cool down the RCS. Continuous ECCS injection will repressurize the RCS. PTS is likely to occur with cooler RCS temperatures and higher pressures.

Technical Reference(s): DB-OP-02000 Basis and Deviation Document R22 (SR5 p455)

DB-OP-06900, PLANT HEATUP; DB-OP-02000, RPS, SFAS, SFRCS TRIP, OR SG TUBE RUPTURE SR5

Reference(s) Provided: None.

Learning Objective:

Question Source: Bank # X

Modified Bank # ____ (note changes or attach parent)

New ____

Question History: Last NRC Exam ____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge X

Comprehension / Analysis / **H**igher Cognitive ____

10 CFR Part 55 Content: 55.41 (b)(14)

55.43 ____

Comments:

PWR GFES Bank QID P2800.

K/A Change.

Examination Outline Cross-Reference: Q76

Level	RO	SRO
Tier #		1
Group #		1
K/A #	000009 G2.4.22	
Importance Rating		4.4

K/A Statement: Small Break LOCA / EMERGENCY PROCEDURES/PLAN: Knowledge of the bases for prioritizing safety functions during abnormal and emergency operations

Proposed Question: Q76

76. A small break LOCA resulted in a loss of Subcooling Margin (SCM). The Crew has completed all Immediate Actions and have routed to Section 5, LACK OF ADEQUATE SUBCOOLING MARGIN.

The following plant conditions CURRENTLY exist:
- RCS pressure is 1300 psig
- RCS T-cold is 545°F and decreasing.
- RCS average incore temperature is 542°F and decreasing.
- Pressurizer level is 45" and increasing.
- Steam generator levels are being maintained per Specific Rule 4.

Given the above conditions, the Command SRO should direct the crew to ___(1)___ and route to Section ___(2)___ WITH SGS REMOVING HEAT COOLDOWN.

A. (1) cooldown at a limit of 50°F/hr
(2) 11.0 RCS SATURATED

B. (1) cooldown at a limit of 100°F/hr
(2) 11.0 RCS SATURATED

C. (1) cooldown at a limit of 50°F/hr
(2) 13.0 RCS SUBCOOLED

D. (1) cooldown at a limit of 100°F/hr
(2) 13.0 RCS SUBCOOLED

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Proposed Answer: **C**

Explanation: Q76

A. Incorrect – Plausible because Specific Rule 5 (PTS) is invoked which requires cooldown at less than 50°F/hr. SCM is adequate in this scenario, therefore the crew should route to Section 13, RCS Subcooled.

B. Incorrect – Plausible because cooldown limit is normally up to 100°F/hr. Cooldown is limited to less than 50°F/hr if possible, by Specific Rule 5 (PTS). SCM is currently adequate in this scenario, therefore the crew should route to Section 13, RCS Subcooled.

C. Correct – PTS is invoked (No RCPs running and HPI injecting due to initial LOSCM) which requires minimizing cooldown rate, if possible, to less than 50°F/hr. SCM is adequate in this scenario, therefore the crew should route to Section 13, RCS Subcooled.

D. Incorrect – Plausible because the cooldown limit is normally up to 100°F/hr. Cooldown is limited to less than 50°F/hr if possible, by Specific Rule 5 (PTS). SCM is adequate in this scenario, therefore the crew should route to Section 13, RCS Subcooled.

Technical Reference(s): DB-OP-02000 RPS, SFAS, SFRCS TRIP, OR SG TUBE RUPTURE – Specific Rules 3, 5, Section 5.

Reference(s) Provided: **STEAM TABLES**

Learning Objective: OPS-GOP-301-05S

Question Source: Bank # _____

Modified Bank # X DB Exam Bank #288094

New _____

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge _____

Comprehension / Analysis / **H**igher Cognitive X

10 CFR Part 55 Content: 55.41 _____

55.43 (b)(5)

Comments:

This question meets Tier 1 criteria as it tests the applicant's knowledge of actions found in the Emergency Operating Procedures.

Examination Outline Cross-Reference: Q77

Level	RO	SRO
Tier #		1
Group #		1
K/A #	000026 AA2.06	
Importance Rating		3.3

K/A Statement: Ability to determine and/or interpret the following as they apply to LOSS OF COMPONENT COOLING WATER: The length of time after the loss of CCW flow to a component before that component may be damaged

Proposed Question: Q77

77. The Reactor is operating at 100% RTP.
- A large leak was identified on the CCW Loop 2 Essential Header
- CCW Pump 2 control switch is placed in LOCKOUT and the close power fuses have been pulled
- The crew enters the applicable conditions for LCO 3.7.7 Component Cooling Water (CCW) System and LCO 3.8.1 AC SOURCES – Operating
- A Safety Function Determination for the Supported System(s) have been completed

The REQUIRED ACTIONS for LCO 3.5.2 ECCS – Operating ____(1)____ required to be performed.

If an emergency event were to subsequently occur, the Train 2 HPI and LPI pumps ___(2)____.

A. (1) are NOT
(2) may be operated for up to one hour

B. (1) are NOT
(2) must be tripped immediately

C. (1) are
(2) may be operated for up to one hour

D. (1) are
(2) must be tripped immediately

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Proposed Answer: **A**

Explanation: Q77

A. Correct – Part (1) LCO 3.0.6 and 3.7.7 require entry into LCO 3.7.7 Condition A and LCO 3.8.1 Condition B for the plant conditions indicated in the question stem. LCO 3.0.6 along with direction from 3.7.7 do not require the supported HPI/LPI ECCS equipment LCO 3.5.2 Actions to be performed. The LCO is still not met. Part (2) DB-OP-02523 Attachment 2 indicates that MU, HPI, and LPI pumps on the effected loop may be operated for up to 1 hour with no/degraded CCW cooling.

B. Incorrect – Plausible as Part (1) is correct. Part (2) is plausible since some equipment (EDG for example) is required to be tripped immediately when its cooling has been lost.

C. Incorrect – Plausible as HPI/LPI ECCS equipment is supported by CCW. Part (2) is correct.

D. Incorrect – Plausible as Part (1) is correct. Part (2) is plausible since some equipment (EDG for example) is required to be tripped immediately when its cooling has been lost.

Technical Reference(s): DB-OP-02523 R13, COMPONENT COOLING WATER SYSTEM MALFUNCTIONS – Attachment 2 (p60)

Reference(s) Provided: None

Learning Objective:

Question Source: Bank # _____

Modified Bank # X (note changes or attach parent)

New _____

Question History: Last NRC Exam 2008 Q45 (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge X

Comprehension / Analysis / **H**igher Cognitive _____

10 CFR Part 55 Content: 55.41 _____

55.43 (b)(5)

Comments:

This question was considered significantly modified as the question stem was changed to ask about LCO action statements to enter (SRO ONLY) and the distractors and answer were all changed to address change in question stem.

This question meets Tier 1 criteria as it tests the applicant's knowledge of actions found in the Abnormal Operating Procedures.

Examination Outline Cross-Reference: Q78

Level	RO	SRO
Tier #		1
Group #		1
K/A #	000056 AA2.81	
Importance Rating		3.7

K/A Statement: Ability to determine and/or interpret the following as they apply to LOSS OF OFFSITE POWER: S/G level and pressure

Proposed Question: Q78

78. Reactor power was initially at 100% RTP.

An event occurs which causes a loss of offsite power.
- All plant systems have responded properly
- The crew entered DB-OP-02000, RPS, SFAS, SFRCS TRIP, OR SG TUBE RUPTURE and completed all Immediate Actions

(1) Which of the following describes the expected plant response to this event?
(2) What action should the Command SRO direct NEXT to mitigate the transient?

A. (1) Each AFW Train will maintain its respective SG at 49 inches.
(2) Establish Pressurizer Level at 100 inches.

B. (1) Each AFW Train will maintain the opposite SG at 55 inches.
(2) Establish Pressurizer Level at 100 inches.

C. (1) Each AFW Train will maintain its respective SG at 49 inches.
(2) Restore instrument air to the Atmospheric Vent Valves.

D. (1) Each AFW Train will maintain the opposite SG at 55 inches.
(2) Restore instrument air to the Atmospheric Vent Valves.

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Proposed Answer: **C**

Explanation: Q78

A. Incorrect – Plausible as Part (1) is correct. Part (2) is plausible as this action will be required but will be done per Attachment 1 after IA is restored to the AVVs per SR 4.

B. Incorrect – Plausible because each AFW train monitors both SG conditions such as level and is able to supply the cross-train SG under some conditions. For example, if a single SG isolated on low SG pressure, both AFW Trains will align to feed the good SG with the cross-train pump attempting the control level at 55 inches while the onside AFW train controls at 49 inches to prevent hunting. This response is incorrect because it does not describe the expected plant response to this event. Part (2) is plausible as this action will be required but will be done per Attachment after IA is restored to the AVVs per SR 4.

C. Correct – SFRCS will actuate on the loss of all RCPs when the LOOP occurs, causing both AFW Pumps to start and control level their respective SG at 49". The LOOP will result in a loss of IA to the AVVs which must be restored to avoid opening Main Steam Safety Valves.

D. Incorrect – Plausible because each AFW train monitors both SG conditions such as level and is able to supply the cross-train SG under some conditions. For example, if a single SG isolated on low SG pressure, both AFW Trains will align to feed the good SG with the cross-train pump attempting the control level at 55 inches while the onside AFW train controls at 49 inches to prevent hunting. This response is incorrect because it does not describe the expected plant response to this event. Part (2) is correct.

Technical Reference(s): DB-OP-02000, RPS, SFAS, SFRCS TRIP, OR SG TUBE RUPTURE; (2) DB-OP-02521, LOSS OF AC BUS POWER SOURCES

Reference(s) Provided: None

Learning Objective:

Question Source: Bank # ____

Modified Bank # X (note changes or attach parent)

New ____

Question History: Last NRC Exam 2022 Q13 (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge ____

Comprehension / Analysis / **H**igher Cognitive X

10 CFR Part 55 Content: 55.41 ____

55.43 (b)(5)

Comments:

This question was considered significantly modified as the question stem was changed to ask about procedure decision points (SRO ONLY) and the distractors and answer were all changed to address changes in the question stem.

This question meets Tier 1 criteria as it tests the applicant's knowledge of actions found in the Emergency and Abnormal Operating Procedures.

Examination Outline Cross-Reference: Q79

Level	RO	SRO
Tier #		1
Group #		1
K/A #	000057 G2.4.06	
Importance Rating		4.7

K/A Statement: Loss of Vital AC Instrument Bus: Knowledge of emergency and abnormal operating procedures major action categories

Proposed Question: Q79

79. The following plant conditions have occurred from 80% RTP:
- ICS runback in progress
- Deaerator #1 level indicates 7.8 feet
- Deaerator #2 level indicates 0 feet
- #1 MFPT speed indicates 4950 rpm
- #2 MFPT speed indicates 0 rpm
- NNI-X ABT transferred to alternate power supply
- EIAC indicating lights are out
- Instrument Air Header pressure indicates 0 psig

The following annunciators (among others) are LIT:
- 1-1-K, EDG 2 TRBL
- 3-1-C, BWST LVL LO
- 3-3-B, SFP TEMP HI
- 14-2-D, ICS/NNI 118V AC PWR TRBL

Based on these conditions, the Command SRO will direct the crew to _____.

A. trip the reactor IAW DB-OP-02542, LOSS OF YBU

B. stabilize the plant by placing ICS Hand/Auto stations in HAND IAW DB-OP-02532, LOSS OF NNI/ICS POWER

C. place BOTH ICS FW Loop Hand/Auto Stations in HAND to stabilize RCS T_{AVE} IAW DB-OP-02526, PRIMARY TO SECONDARY HEAT TRANSFER UPSET

D. verify SAC 1 and 2 are running IAW DB-OP-02528, INSTRUMENT AIR MALFUNCTIONS

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Proposed Answer: **A**

Explanation: Q79

A. Correct – A loss of instrumentation bus YBU gives all of the indications and alarms provided in the stem.

B. Incorrect – Plausible as an instrument bus power supply has failed as is indicated by the diverse number of indications and alarms which have suddenly occurred. One of these is the NNI-X ABT switching to the alternate power supply.

C. Incorrect – Plausible as a loss of YBU will cause the #2 MFPT speed to indicate 0 in the main control room. The pump will not have actually tripped.

D. Incorrect – Plausible as the loss of YBU will cause the EIAC to trip and will cause all of the control instrument air header pressure indicators to read zero. The SACs are not affected by the loss of YBU.

Technical Reference(s): DB-OP-02542, LOSS OF YBU.

Reference(s) Provided: None

Learning Objective:

Question Source: Bank # _____

Modified Bank # X (note changes or attach parent)

New _____

Question History: Last NRC Exam 2005 Q79 (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge _____

Comprehension / Analysis / **H**igher Cognitive X

10 CFR Part 55 Content: 55.41 _____

55.43 (b)(5)

Comments:

This question was considered significantly modified as the question stem was changed to ask about a different instrumentation bus failure and 2 distractors were changed.

This question meets Tier 1 criteria as it tests the applicant's knowledge of actions found in the Abnormal Operating Procedures.

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Examination Outline Cross-Reference: Q80

Level	RO	SRO
Tier #		1
Group #		1
K/A #	000058 G2.1.08	
Importance Rating		4.1

K/A Statement: Loss of DC Power CONDUCT OF OPERATIONS: Ability to coordinate personnel activities outside the control room

Proposed Question: Q80

80. The Reactor was operating at 45% RTP, during power ascension following a refueling outage, when the following alarms were received:
    - 1-5-F, DC PANEL VOLTAGE LO
    - 1-6-G, DC BUS 2 TRBL
    - 1-6-K, YV2-YV4 TRBL

    An equipment operator sent to investigate reports that +125V DC PNL D2P reads zero volts.

    Based on these indications the Command SRO will need to direct an EO to _____.

    A. open MU38, RCP SEAL RETURN within 30 minutes and serve as a dedicated operator.

    B. throttle open CD81, CNDS PUMPS MINIMUM RECIRC BYPASS when condensate flow drops to < 1.0 mpph.

    C. place the DC control power knife switch in the ALTERNATE DBP07 position at Bus D2.

    D. throttle closed FW6398, MDFP ISOLATION TO SG1 to prevent overfill of SG1.

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Proposed Answer: **A**

Explanation: Q80

A. Correct – The event described is a failure of DC Bus D2P. Per DB-OP-02538, LOSS OF D2P AND DBP Attachment 3, MU38 will fail closed and must be locally reopened within 30 minutes with RCPs running. In addition, a local, dedicated operator must be stationed at MU38 to close it if a subsequent SFAS Level 3 signal is received.

B. Incorrect – Plausible as this is a locally required action required for a loss of DC Bus D1P per DB-OP-02537, LOSS OF D1P AND DAP.

C. Incorrect – Plausible as this is a locally required action required for a loss of DC Bus D2N per DB-OP-02540, LOSS OF D2N AND DBN.

D. Incorrect – Plausible as this is a locally required action required for a loss of DC Bus D1N resulting in a reactor trip where the MDFP is used to maintain SG level per DB-OP-02539, LOSS OF D1N AND DAN.

Technical Reference(s): DB-OP-02538, LOSS OF D2P AND DBP

Reference(s) Provided: None

Learning Objective:

Question Source: Bank # _____

Modified Bank # _____ (note changes or attach parent)

New X

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge X

Comprehension / Analysis / **H**igher Cognitive _____

10 CFR Part 55 Content: 55.41 _____

55.43 (b)(1)&(5)

Comments:

This question meets Tier 1 criteria as it tests the applicant's knowledge of actions found in the Abnormal Operating Procedures.

Examination Outline Cross-Reference: Q81

Level	RO	SRO
Tier #		1
Group #		1
K/A #	000077 AA2.06	
Importance Rating		3.4

K/A Statement: Ability to determine and/or interpret the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: Generator frequency

Proposed Question: Q81

81. A grid disturbance is occurring. The crew enters DB-OP-02546, DEGRADED GRID.

The following plant conditions exist:
- Generator Output = 850 MWe and (+) 450 MVARs
- Generator Hydrogen Pressure = 60 psig
- Generator Output Voltage = 24.5 KV

In accordance with DB-PF-06703, CC9.5 ESTIMATED CAPABILITY CURVES – LEAD-LAG, the Generator output ____(1)____ within the limits of the Generator Capability Curve.

Subsequently:
- Switchyard Alarm 25-3-C, SYS FREQUENCY LOW is in alarm
- The BOP Reactor Operator reports Generator Frequency is at 57.0 Hz
- Turbine vibrations are slightly elevated and rising

The Command SRO ____(2)____ required to direct the ATC Reactor Operator to trip the Reactor.

REFERENCE PROVIDED

A. (1) was
(2) is

B. (1) was NOT
(2) is

C. (1) was
(2) is NOT

D. (1) was NOT
(2) is NOT

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Proposed Answer: **A**

Explanation: Q81

A. Correct – Part (1) is correct as the generator output is within the limits of the Generator Capability Curve (CC9.5). Part (2) is correct as an automatic generator trip on low frequency should have occurred at 57.7 Hz (relays 81U1 and 81 U2) and since it did not occur (based on stem conditions) it must be manually tripped. At power levels >40% the SRO will direct the Reactor to be tripped.

B. Incorrect – Part (1) is plausible since it would be correct if power factor were leading or if Gen H2 pressure were lower. Part (2) is correct.

C. Incorrect – Plausible as Part (1) is correct. Part (2) is plausible as there is not a specific procedure step in DB-OP-02546 to trip the main turbine if generator frequency is too low.

D. Incorrect – Part (1) is plausible since it would be correct if power factor were leading or if Gen H2 pressure were lower. Part (2) is plausible as there is not a specific procedure step in DB-OP-02546 to trip the main turbine if generator frequency is too low.

Technical Reference(s): DB-OP-02546, DEGRADED GRID; DB-PF-06703, CC9.5 ESTIMATED CAPABILITY CURVES – LEAD-LAG; DB-OP-06301, GENERATOR AND EXCITER OPERATING PROCEDURE; SD-005, SYSTEM DESCRIPTION FOR MAIN GENERATOR AND AUXILIARIES

Reference(s) Provided: **DB-PF-06703, CC9.5 ESTIMATED CAPABILITY CURVES – LEAD-LAG**

Learning Objective:

Question Source: Bank # X

Modified Bank # _____ (note changes or attach parent)

New _____

Question History: Last NRC Exam Oconee 2015 Q81 (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge _____

Comprehension / Analysis / **H**igher Cognitive X

10 CFR Part 55 Content: 55.41 _____

55.43 (b)(5)

Comments:

This question meets Tier 1 criteria as it tests the applicant's knowledge of actions found in the Abnormal Operating Procedures.

Examination Outline Cross-Reference: Q82

Level	RO	SRO
Tier #		1
Group #		2
K/A #	000024 AA2.05	
Importance Rating		3.9

K/A Statement: Ability to determine and/or interpret the following as they apply to EMERGENCY BORATION: Amount of boron to add to achieve required SDM

Proposed Question: Q82

82. An event has occurred which leaves rod index in the Unacceptable region at 230 Rod Index.

The Core Operating Limits Report shows rods entering the:
- Restricted region at 240 rod index
- Acceptable region at 255 rod index

Reactor Operators Guidance:
- 2 gallons boric acid per rod index

With power remaining constant:

(1) What is the MINIMUM required acid which is REQUIRED to be added to meet shutdown margin requirements?

AND

(2) What is the Technical Specification basis for this addition?

A. (1) 20 gallons
(2) regulating rods may be inserted too far to provide sufficient negative reactivity insertion following a reactor trip

B. (1) 50 gallons
(2) regulating rods may be inserted too far to provide sufficient negative reactivity insertion following a reactor trip

C. (1) 20 gallons
(2) precludes long term depletion with abnormal group insertions or configurations and limits the potential for an adverse xenon redistribution

D. (1) 50 gallons
(2) precludes long term depletion with abnormal group insertions or configurations and limits the potential for an adverse xenon redistribution

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Proposed Answer: **A**

Explanation: Q82

A. Correct – Part (1) 20 gallons will move rods to the restricted region. SDM is met in the restricted region Part (2) is correct per TS bases 3.2.1.D.1.

B. Incorrect – Plausible since (1) 50 gallons will place rods in the acceptable region which is not required to meet SDM. Part (2) is correct.

C. Incorrect – Plausible since (1) 20 gallons is correct, and Part (2) is the bases for going from the restricted to acceptable region

D. Incorrect – Plausible since (1) 50 gallons will place rods in the acceptable region and Part (2) is bases for going from restricted to acceptable region.

Technical Reference(s) TS 3.2.1 REGULATING ROD INSERTION LIMITS – Bases.

Reference(s) Provided: None

Learning Objective:

Question Source: Bank # X

Modified Bank # _____ (note changes or attach parent)

New _____

Question History: Last NRC Exam DB 2016 Q83 (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge X

Comprehension / Analysis / **H**igher Cognitive ____

10 CFR Part 55 Content: 55.41 _____

55.43 (b)(5)

Comments:

This question meets Tier 1 criteria as it tests the applicant's knowledge of actions found in the Annunciator Response and Abnormal Operating Procedures.

Examination Outline Cross-Reference: Q83

Level	RO	SRO
Tier #		1
Group #		2
K/A #	000061 G2.1.25	
Importance Rating		4.2

K/A Statement: Area Radiation Monitoring System Alarms CONDUCT OF OPERATIONS: Ability to interpret reference materials, such as graphs, curves, and tables (reference potential)

Proposed Question: Q83

83. At 0800, a LOCA occurred inside of containment with the reactor operating at 100% RTP.

The crew entered and performed the immediate actions of DB-OP-02000, RPS, SFAS, SFRCS TRIP, OR SG TUBE RUPTURE.

At 0815, the Shift Manager declared an ALERT based on a loss of the RCS Barrier associated with UNISOLABLE RCS leakage.

At 0905 the following conditions exist:
- Containment Area Rad Monitor RE 4596A reads 2600 R/hr
- Containment Area Rad Monitor RE 4596B reads 2500 R/hr
- Containment pressure is 41 psia
- CTMT Spray Pump 1 and 2 are running

Based on these conditions, the Shift Manager MUST _____.

REFERENCE PROVIDED

A. determine a concurrent EAL ALERT entry condition exists due to RE 4596A and RE 4596B exceeding the RCS Barrier Loss value

B. declare a SITE AREA EMERGENCY due to RE 4596A exceeding the RCS Barrier Loss and FC Barrier Loss values

C. declare a SITE AREA EMERGENCY due to a loss of the RCS Barrier and a potential loss of the Containment Barrier due to pressure exceeding 40 psia

D. declare a GENERAL EMERGENCY due to a loss of the RCS and FC Barriers due to RE 4596A exceeding loss values and a potential loss of the Containment Barrier due to pressure exceeding 40 psia

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Proposed Answer: **B**

Explanation: Q83

A. Incorrect – Plausible if it had been less than 1 hour since reactor had been shut down. Since it has been between 1-2 hours since the event occurred, RE 4596A has exceeded the new FC loss threshold as well as a loss of the RCS barrier has occurred.

B. Correct – With RE 4596A exceeding 2560 R/hr and it being greater than 1 hour since the reactor was shut down, Table F-2 of the Hot EAL Chart indicate a loss of both the Fuel Cladding and RCS Barriers. Therefore, EAL FS1 must be declared due to a loss of 2 fission product barriers.

C. Incorrect – Plausible as the RCS Barrier is lost per information in the stem. Also, a potential loss of the containment barrier is considered to have occurred if containment pressure > 40 psia (which it is) with < 1 full train of containment cooling running. According to Table F-3 of the Hot EAL Chart a full train of containment cooling is 2 containment spray pumps running.

D. Incorrect – Plausible as the loss of the FC and RCS barriers has occurred due to RE 4596A reading, but a potential loss of the containment barrier has not occurred since at least 1 train of containment cooling is running with containment pressure > 40 psia.

Technical Reference(s): Davis-Besse EAL Wall Board – Hot EALs RCS Barrier – Loss (provided in stem, RCS Loss A.1). added Fuel Clad Loss, C.1 in FPB matrix, FS1, loss of any 2 barriers.

Reference(s) Provided: **Davis-Besse EAL Wall Boards**

Learning Objective:

Question Source: Bank # _____

Modified Bank # _____ (note changes or attach parent)

New X

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge _____

Comprehension / Analysis / **H**igher Cognitive X

10 CFR Part 55 Content: 55.41 _____

55.43 (b)(5)

Comments:

This question meets Tier 1 criteria as it tests the applicant's knowledge of actions found in the Emergency Procedures.

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Examination Outline Cross-Reference: Q84

Level	RO	SRO
Tier #		1
Group #		2
K/A #	000078 G2.4.38	
Importance Rating		4.4

K/A Statement: RCS Leak / EMERGENCY PROCEDURES/PLAN: Ability to take actions required by the facility emergency plan implementing procedures, including supporting or acting as emergency coordinator

Proposed Question: Q84

84. The reactor is in MODE 1 when a significant RCS Leak occurs requiring the crew to manually trip the reactor, align maximum makeup flow, and start high pressure injection.

The Shift Manager declares an ALERT under EAL FA1.

The Shift Manager MUST direct _____.

A. that there be NO eating, drinking, smoking, or chewing

B. evacuation of all non-essential personnel within the protected area

C. performance of Assembly/Accountability

D. activation of the Emergency Response Organization

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Proposed Answer: **D**

Explanation: Q84

A. Incorrect – Plausible as this is an action required for a SITE AREA EMERGENCY.

B. Incorrect – Plausible as this is an action required for SITE AREA EMERGENCY.

C. Incorrect – Plausible as this is an action required for SITE AREA EMERGENCY or General Emergency or at the discretion of the CR EC.

D. Correct – activation of the Emergency Response Organization is required upon the declaration of an ALERT.

Technical Reference(s): NOP-LP-5502

Reference(s) Provided: None

Learning Objective:

Question Source: Bank # _____

Modified Bank # _____ (note changes or attach parent)

New X

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge X

Comprehension / Analysis / **H**igher Cognitive _____

10 CFR Part 55 Content: 55.41 _____

55.43 (b)(5)

Comments:

This question meets Tier 1 criteria as it tests the applicant's knowledge of actions found in the Emergency Procedures.

Examination Outline Cross-Reference: Q85

Level	RO	SRO
Tier #		1
Group #		2
K/A #	BW A05 AA2.16	
Importance Rating		3.0

K/A Statement: Ability to determine and/or interpret the following as they apply to EMERGENCY DIESEL ACTUATION: Makeup pump, tank, valves, and/or flow

Proposed Question: Q85

85. The Reactor is at 100% RTP.
- C1 Alternate Source Breaker, ABDC1 is removed from its cubicle for maintenance.

Subsequently:
- Breaker AC110, BUS TIE FROM C2 BUS fails open.
- EDG 1 starts and is powering Bus C1

While EDG 1 is operating in this configuration:
- 1-4-A, EDG 1 DAY TK LVL LO is LIT
- An Equipment Operator reports that the Emergency Fuel Oil Transfer Pump 1 breaker tripped on overcurrent

Assuming ABDC1 and AC110 can NOT be restored in a timely manner, the Command SRO should direct _____.

A. gravity filling EDG 1 Day Tank from the EDG Fuel Storage (Week) Tank 1

B. the SBODG be placed in service to supply Bus C1

C. filling EDG 1 Day Tank from the Diesel Oil Storage Tank (DOST)

D. ordering a fuel truck to directly fuel EDG 1 Day Tank

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Proposed Answer: **C**

Explanation: Q85

A. Incorrect – Plausible as this is a method used for filling the Diesel Fire Pump Day Tank located in DB-OP-06273.

B. Incorrect – Plausible since the SBODG can supply C1 during other scenarios, but not directly. In this scenario the breakers to line-up the SBODG to C1 are unavailable.

C. Correct – DB-OP-06273, Diesel Fuel Oil Transfer provides for this method of emergency filling either EDG Day Tank. This method does not make EDG 2 inoperable.

D. Incorrect – Plausible as this is an infrequent but proceduralized method of filling the Diesel Fire Pump Day Tank in DB-OP-06273. Fuel trucks are also used for the SBODG Fuel Oil Tank, but NOT the EDG Day Tanks.

Technical Reference(s): DB-OP-06273, DIESEL FUEL OIL TRANSFER

Reference(s) Provided: None

Learning Objective:

Question Source: Bank # _____

Modified Bank # _____ (note changes or attach parent)

New X

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge _____

Comprehension / Analysis / **H**igher Cognitive X

10 CFR Part 55 Content: 55.41 _____

55.43 (b)(5)

Comments:

This question meets Tier 1 criteria as it tests the applicant's knowledge of actions found in the Abnormal Operating Procedures.

Examination Outline Cross-Reference: Q86

Level	RO	SRO
Tier #		2
Group #		1
K/A #	005 G2.4.46	
Importance Rating		4.2

K/A Statement: RESIDUAL HEAT REMOVAL SYSTEM, EMERGENCY PROCEDURES/PLAN: Ability to verify that the alarms are consistent with the plant conditions

Proposed Question: Q86

86. Initial Plant Conditions:
- The Reactor is shutdown in MODE 5
- RCS Cooldown and depressurization are in progress
- DHR Loop 1 aligned for Standby DHR operation
- DHR Loop 2 in service
- Conditions have been established to stop all Reactor Coolant Pumps
- RCS Temperature is 160°F
- RCS Pressure is 210 psig
- Pressurizer Level is 85 inches

When the last RCP is stopped, an event occurs. The following conditions are noted:
- 3-1-I, LP INJ 2 FLOW LO is LIT
- Incore thermocouple temperatures are rising
- No flow on FYI DH2A, LPI/DH PUMP 2 OUTLET FLOW
- DHR Pump 1 Breaker Status Light is Green
- DHR Pump 2 Breaker Status Light is Red
- DH1A, DH PUMP 2 DISCHARGE TO RCS is closed
- Pressurizer Level is slowly rising

Which of the following procedures should the Command SRO direct the crew to perform based on this event?

A. DB-OP-02000, RPS, SFAS, SFRCS TRIP OR SG TUBE RUPTURE

B. DB-OP-02527, LOSS OF DECAY HEAT REMOVAL, Section 4.1, Loss of DHR Pump

C. DB-OP-02527, LOSS OF DECAY HEAT REMOVAL, Section 4.2, Loss of DHR Flowpath

D. DB-OP-02527, LOSS OF DECAY HEAT REMOVAL, Section 4.3, Loss of DHR Inventory.

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Proposed Answer: **C**

Explanation: Q86

A. Incorrect – Plausible because a lack of heat transfer event is occurring. DB-OP-02000, Section 6, Lack of Heat Transfer provides direction to respond to an overheating event, however once the plant is on decay heat removal, DB-OP-02527, Loss of Decay Heat Removal becomes the controlling procedure, not DB-OP-02000. A candidate may select this response if they do not know that DB-OP-02527 would be the controlling procedure for this event.

B. Incorrect – Plausible because this is the correct procedure to respond to loss of Decay Heat Removal Event, however the #2 DHR pump remains in operation as noted by the breaker status light. The DHR pump has not been lost. This would be the correct procedure section if the DHR Pump was lost. The candidate may select this response if the more likely scenario, loss of DHR pump had occurred.

C. Correct – Based on the indications provided, a loss of Decay Heat Removal Flowpath has occurred. This can be determined by the pump still operating in conjunction with the low flow alarm and rising pressure level and RCS temperatures as core begins to heat up due to a loss of cooling.

D. Incorrect – Plausible because on a loss of inventory from the DHR system, the red status light for the DHR breaker would still be on and the Low Flow Alarm would be possible if the DHR Pump was cavitating or a system break upstream of the flow indicator had occurred. Stopping an RCP does cause a hydraulic transient as system flow coasts down which makes a leak possible. The candidate may select this response if they think a loss of inventory had occurred. Incorrect due to pressurizer level rising.

Technical Reference(s): DB-OP-02527 R20, Loss of DHR, Section 4.2, Loss of DHR Flowpath

Reference(s) Provided: None

Learning Objective:

Question Source: Bank # X

Modified Bank # ____ (note changes or attach parent)

New ____

Question History: Last NRC Exam 2022 Q87 (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge ____

Comprehension / Analysis / **H**igher Cognitive X

10 CFR Part 55 Content: 55.41 ____

55.43 (b)(5)

Comments:

Examination Outline Cross-Reference: Q87

Level	RO	SRO
Tier #		2
Group #		1
K/A #	006 A2.15	
Importance Rating		3.5

K/A Statement: Ability to (a) predict the impacts of the following on the EMERGENCY CORE COOLING SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Vortex/cavitation

Proposed Question: Q87

87. A Large Break LOCA has occurred, and the following indications exist:
- Containment pressure is 42 psia
- RCS pressure is 150 psig
- The HPI and MU Pumps have been stopped

DB-OP-02000, Attachment 7, TRANSFERRING LPI SUCTION TO THE EMERGENCY SUMP, is in progress.
- BWST level is at 9 feet and slowly lowering
- DH9B, CTMT EMERGENCY SUMP ISO VALVE has been opened using HISDH9B
- DH9A, CTMT EMERGENCY SUMP ISO VALVE did NOT open using HISDH9A
- DH7A, BWST OUTLET LINE 2 remains open

In accordance with DB-OP-02000, Attachment 7, TRANSFERRING LPI SUCTION TO THE EMERGENCY SUMP, the Command SRO must direct the crew to:

Block and Stop LPI Pump 2 and CTMT Spray Pump 2 ____(1)____.

And will also direct an ____(2)____.

A. (1) IMMEDIATELY
(2) EO to locally deenergize and then manually open DH9A

B. (1) IF AT ANY TIME cavitation is noted
(2) RO to manually trip Train 2 SFAS Level 5 Output Modules and then open DH9A using HISDH9A

C. (1) IMMEDIATELY
(2) RO to manually trip Train 2 SFAS Level 5 Output Modules and then open DH9A using HISDH9A

D. (1) IF AT ANY TIME cavitation is noted
(2) EO to locally deenergize and then manually open DH9A

Proposed Answer: **B**

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Examination Outline Cross-Reference: Q87

Explanation: Q87

A. Incorrect – Plausible as the applicant could determine that all Train 2 pumps need to be stopped with DH9A closed and BWST at 9ft. In addition, when an MOV does not reposition, there are instances where the CTRM will direct the valve breaker opened and operated locally from the field to reposition the valve.

B. Correct –DB-OP-02000, Attachment 7 directs operators to Block and Stop pumps experiencing fluctuating flow and motor amps. To open DH9A, the crew will need to manually trip SFAS Level 5 for BWST level and then open DH9A.

C. Incorrect – Plausible as the applicant could determine that all Train 2 pumps need to be stopped with DH9A closed and BWST at 9ft. Part 2 is correct.

D. Incorrect – Plausible Part 1 is correct. When an MOV does not reposition, there are instances where the CTRM will direct the valve breaker opened and operated locally from the field to reposition the valve.

Technical Reference(s): DB-OP-02000, Attachment 7, TRANSFERRING LPI SUCTION TO THE EMERGENCY SUMP

Reference(s) Provided: None

Learning Objective:

Question Source: Bank # _____

Modified Bank # _____ (note changes or attach parent)

New X

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge _____

Comprehension / Analysis / **H**igher Cognitive X

10 CFR Part 55 Content: 55.41 _____

55.43 (b)(5)

Comments:

Examination Outline Cross-Reference: Q88

Level	RO	SRO
Tier #		2
Group #		1
K/A #	012 G2.1.6	
Importance Rating		4.8

K/A Statement: REACTOR PROTECTION SYSTEM, CONDUCT OF OPERATIONS: Ability to manage the control room crew during plant transients (SRO Only)

Proposed Question: Q88

88. Initial plant conditions:
- Reactor power is at 100% RTP
- RPS Ch 3 is in Manual Bypass for I&C testing

The following event occurs:
- The following Annunciators alarm:
  4-4-B, HOT LEG PRESS HI
  4-4-C, HOT LEG PRESS LO
  4-1-D, PZR RLF VLV OPEN
  5-1-G, RPS CH 1 TRIP
- The PORV indicates OPEN
- PZR Heaters are OFF

The ATC RO completes all Immediate Actions from memory and reports the following:
- RCS Pressure is at 2010 psig and stable
- RCS Temperature is at 582°F and stable

Which of the following should the Command SRO direct NEXT?

A. Trip the Reactor and GOTO DB-OP-02000, RPS, SFAS, SFRCS TRIP, OR SG TUBE RUPTURE.

B. Pull DB-OP-02004, REACTOR COOLANT ALARM PANEL 4 ANNUNCIATORS and complete required actions IAW Annunciator 4-4-B, HOT LEG PRESS HI.

C. Provide an RCS Pressure control band using the PZR Heaters IAW NOP-OP-1002, CONDUCT OF OPERATIONS.

D. Open RC11, PORV BLOCK VALVE to restore PORV operability IAW DB-OP-02513, PRESSURIZER SYSTEM ABNORMAL OPERATION.

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Proposed Answer: **C**

Explanation: Q88

A. Incorrect - Plausible since RPS Ch 3 is in Bypass and RPS CH 1 trips. Incorrect since the stem puts RPS in a 2 out of 3 trip logic and only one channel trips therefore the reactor does not trip.

B. Incorrect – These conditions require entry into an abnormal procedure that has Immediate Actions. Verifying IAs is a higher priority than reviewing an annunciator procedure. The ARP is reviewed for plant response as time permits. Plausible since ROs perform Annunciator reviews for unexpected conditions.

C. Correct – Setting a limit (control band) is required when manual actions are required IAW NOP-OP-1002, Step 4.10.3.

D. Incorrect – While important to restore operability to each component, the conditions in the stem give a stuck open PORV, in which TS 3.4.11, Condition A requires closing of the PORV block valve to maintain barrier integrity.

Technical Reference(s): DB-OP-01003, Operations Procedure Use Instructions, 6.5.1

Reference(s) Provided: None

Learning Objective:

Question Source: Bank # ____

Modified Bank # ____ (note changes or attach parent)

New X

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge ____

Comprehension / Analysis / **H**igher Cognitive X

10 CFR Part 55 Content: 55.41 (b)()

55.43 _____

Comments:

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Examination Outline Cross-Reference: Q89

Level	RO	SRO
Tier #		2
Group #		1
K/A #	064 G2.2.45	
Importance Rating		4.4

K/A Statement: EMERGENCY DIESEL GENERATOR SYSTEM, EQUIPMENT CONTROL Ability to determine and/or interpret TS with action statements of greater than 1 hour (SRO Only)

Proposed Question: Q89

89. Initial Plant Conditions:
- The Reactor is at 100% RTP
- No other safety equipment is removed from service

At 0800 on January 7th:
- During a routine surveillance, EDG 1 failed to reach rated speed and was shutdown and declared inoperable

At 0845 on January 7th:
- Surveillance Requirement 3.8.1.1 for OPERABLE offsite circuit(s) was completed

At 1100 on January 7th:
- Troubleshooting determined the EDG 1 governor was failed

Based on the above conditions and in accordance with Technical Specification 3.8.1 AC Sources - Operating, which REQUIRED ACTION and COMPLETION TIME is correct?

A. Restore EDG 1 to OPERABLE status by 0800 on January 10th.

B. Declare required features supported by EDG 1 inoperable by 1200 on January 7th.

C. Determine EDG 2 is not inoperable due to common cause failure by 1100 on January 8th.

D. Perform Surveillance Requirement 3.8.1.1 for OPERABLE offsite circuit(s) by 1645 on January 7th.

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Proposed Answer: **D**

Explanation: Q89

A. Incorrect – This is a 7-day action NOT 72 hours. Plausible since there are 72 hr actions like restoring HPI which is supplied by the EDG when its normal supply is lost.

B. Incorrect – Plausible as this would be a possible required action per TS 3.8.1, Condition B.2. However, only when its redundant required feature(s) is inoperable.

C. Incorrect – Plausible as this would be a possible required action per TS 3.8.1, Condition B, Required Action B.3.1 if EDG 1 remained inoperable for greater than 24 hours.

D. Correct – Perform SR 3.8.1.1 for OPERABLE offsite circuit(s) 1 hour AND Once per 8 hours thereafter.

Technical Reference(s): TS Bases R32, 3.8.1 AC Sources – Operating. (p B 3.8.1-8)

Reference(s) Provided: None

Learning Objective:

Question Source: Bank # _____

Modified Bank # _____ (note changes or attach parent)

New X

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge _____

Comprehension / Analysis / **H**igher Cognitive X

10 CFR Part 55 Content: 55.41 _____

55.43 (b)(3) & (5)

Comments:

Examination Outline Cross-Reference: Q90

Level	RO	SRO
Tier #		2
Group #		1
K/A #	103 A2.04	
Importance Rating		3.4

K/A Statement: Ability to (a) predict the impacts of the following on the CONTAINMENT SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those operations Conditions requiring containment evacuation (including recognition of the alarm)

Proposed Question: Q90

90. Initial Conditions:
- RCS drain has commenced for refueling outage
- The plant is in reduced inventory
- Decay Heat Removal loop 1 is in Operation
- Decay Heat Removal loop 2 is in Standby

The following occurs:
- CTMT Normal Sump Level is rising
- DH Pump 1 motor amps becomes erratic
- DH Pump 1 discharge pressure becomes erratic
- DH Pump 1 flow becomes erratic
- DB-OP-02527, LOSS OF DECAY HEAT REMOVAL is entered

Based on the above plant conditions, complete the following statements.

In accordance with DB-OP-02527, the NEXT action related to the Decay Heat Removal Pumps will be to ____(1)_____. Containment evacuation____(2)____required.

A. (1) start DH Pump 2
(2) is

B. (1) stop DH Pump 1
(2) is

C. (1) start DH Pump 2
(2) is NOT

D. (1) stop DH Pump 1
(2) is NOT

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Proposed Answer: **B**

Explanation: Q90

A. Incorrect: 1st part is incorrect; the standby decay heat pump is not started prior to securing the pump that is cavitating. 1st part is plausible because starting a standby DH pump first could maintain DH removal. 2nd part is correct because with reduced RCS inventory (RCS draining), containment evacuation is required

B. Correct: 1st part is correct the cavitating decay heat pump will be secured prior to starting the standby pump. 2nd part is correct with the RCS draining into containment and it resulting in the cavitation of the running DH pump Shutdown cooling is not able to be restored as the low RCS level does not provide adequate NPSH to run DH pumps in shutdown cooling mode.

C. Incorrect: 1st part is incorrect (see A) 2nd part is incorrect because the containment evacuation alarm is required. Plausible because there is no mention of elevated dose rates in containment.

D. Incorrect: 1st part is correct (see B). 2nd part is incorrect (see C).

Technical Reference(s): DB-OP-02527, LOSS OF DECAY HEAT REMOVAL; RA-EP-02864, CONTAINMENT EVACUATION

Reference(s) Provided: None

Learning Objective:

Question Source: Bank # X

Modified Bank # _____ (note changes or attach parent)

New _____

Question History: Last NRC Exam 2018 Q77 (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge _____

Comprehension / Analysis / **H**igher Cognitive X

10 CFR Part 55 Content: 55.41 _____

55.43 (b)(5)

Comments:

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Examination Outline Cross-Reference: Q91

Level	RO	SRO
Tier #		2
Group #		2
K/A #	035 G2.1.45	
Importance Rating		4.3

K/A Statement: SG SYSTEM CONDUCT OF OPERATIONS: Ability to identify and interpret diverse indications to validate the response of another indication.

Proposed Question: Q91

91. The plant is operating at 100% RTP.

An event occurs which causes the following indications:

- 14-4-E, ICS INPUT MISMATCH is LIT
- 14-5-E, ICS SG 1 ON LO LVL LIMIT is LIT
- 12-3-A, SG 1 OPERATE LVL HI is LIT
- 12-4-A, SG 1 LVL LO is LIT
- Computer point L884, SG 1 LVL (NNI-X) on Plant Computer Group 61 is at 30 inches and slowly lowering
- Computer point L883, SG 1 LVL (NNI-Y) on Plant Computer Group 61 is at 240 inches and slowly rising

The Command SRO should direct the operators to establish manual control of feedwater flow and establish a proper demand signal in accordance with ____(1)____ to mitigate this event.

The BASIS for the Technical Specification LCO challenged (LCO 3.7.18, Steam Generator Level) by this event is to ensure Steam Generator (SG) water inventory is maintained ____(2)____.

A. (1) NORM-OP-1002, CONDUCT OF OPERATIONS HANDBOOK
(2) high enough to provide adequate primary to secondary heat transfer

B. (1) NORM-OP-1002, CONDUCT OF OPERATIONS HANDBOOK
(2) below initial SG inventory assumptions used in the main steam line break (MSLB) accident analysis

C. (1) DB-OP-02532, LOSS OF NNI/ICS POWER
(2) high enough to provide adequate primary to secondary heat transfer

D. (1) DB-OP-02532, LOSS OF NNI/ICS POWER
(2) below initial SG inventory assumptions used in the main steam line break (MSLB) accident analysis

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Proposed Answer: **B**

Explanation: Q91

A. Incorrect – Plausible as part (1) is correct. Part (2) is plausible as it is the TS Basis for the low S/G Level Limit.

B. Correct – Malfunction is failure of controlling Startup (SU) level low which results in SG overfeed. See M-533-00171 R10. If the faulted instrument or component is apparent, action should be taken by the operator to establish manual control of the faulted component and a proper demand signal established IAW NORM-OP-1002, Conduct of Operations Handbook. DB-OP-02526 will be used if the cause is NOT apparent. See step 2.1.4 and 2.1.5. TS Bases for high S/G water level is listed in B 3.7.18 page B 3.7.18-1.

C. Incorrect – Part (1) is plausible if it is determined the cause of the event is a loss of ICS or NNI power. Part (2) is plausible because this is the TS basis for the low-level limit.

D. Incorrect – Part (1) is plausible if it is determined the cause of the event is a loss of ICS or NNI power. Part (2) is correct.

Technical Reference(s): NORM-OP-1002, CONDUCT OF OPERATIONS HANDBOOK; TS Bases for TS 3.7.18.

Reference(s) Provided: None

Learning Objective:

Question Source: Bank # ____

Modified Bank # X (note changes or attach parent)

New ____

Question History: Last NRC Exam 2015 Q93 (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge ____

Comprehension / Analysis / **H**igher Cognitive X

10 CFR Part 55 Content: 55.41 ____

55.43 (b)(5)

Comments:
Modified question stem to indicate level lowering. All distractors and answers were changed.

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Examination Outline Cross-Reference: Q92

Level	RO	SRO
Tier #		2
Group #		2
K/A #	045 G2.2.12	
Importance Rating		4.1

K/A Statement: MAIN TURBINE GENERATOR SYSTEM, EQUIPMENT CONTROL: Knowledge of surveillance procedures

Proposed Question: Q92

92. DB-SS-04150, MAIN TURBINE STOP VALVE TEST and DB-SS-04159, ONLINE ELECTRICAL TRIP DEVICE TEST are BOTH scheduled to be performed during the shift.

During the performance of ____(1)____ the affected ARTS TURBINE-GEN Bypass Switch is required to be placed in the BYPASS position.

When the ARTS TURBINE-GEN Bypass Switch is placed in the BYPASS position, LCO 3.3.16, Anticipatory Reactor Trip System (ARTS) Instrumentation Condition A ____(2)____ met.

A. (1) DB-SS-04159, ONLINE ELECTRICAL TRIP DEVICE TEST
(2) is still

B. (1) DB-SS-04159, ONLINE ELECTRICAL TRIP DEVICE TEST
(2) is NOT

C. (1) DB-SS-04150, MAIN TURBINE STOP VALVE TEST
(2) is still

D. (1) DB-SS-04150, MAIN TURBINE STOP VALVE TEST
(2) is NOT

Proposed Answer: **C**

Explanation: Q92

A. Incorrect – DB-SS-04159 does not require the ARTS TURBINE-GEN Bypass Switch placed in the BYPASS position. Plausible since a failed ETD could cause EHC pressure to lower to the ARTS Trip setpoint.
Part (2) is correct as according to LCO 3.3.16 Table 3.3.16-1 only 3 channels are required to be operable when >45% RTP therefore
LCO 3.3.16 is met.

B. Incorrect – DB-SS-04159 does not require the ARTS TURBINE-GEN Bypass Switch placed in the BYPASS position. Plausible since a failed ETD could cause EHC pressure to lower to the ARTS Trip setpoint. According to LCO 3.3.16 Table 3.3.16-1 only 3 channels are required to be operable when >45% RTP therefore LCO 3.3.16 is still being met. Plausible since 4 channels of RPS and SFAS are required to be operable in Mode 1.

C. Correct – During the performance of DB-SS-4150 the ARTS TURBINE-GEN Bypass Switch is required to be placed in the BYPASS position. According to LCO 3.3.16 Table 3.3.16-1 only 3 channels are required to be operable when >45% RTP therefore LCO 3.3.16 is met.

D. Incorrect – Plausible as during the performance of DB-SS-4150 the ARTS TURBINE-GEN Bypass Switch is required to be placed in the BYPASS position. According to LCO 3.3.16 Table 3.3.16-1 only 3 channels are required to be operable when >45% RTP therefore
LCO 3.3.16 is still being met. Plausible since 4 channels of RPS and SFAS are required to be operable in Mode 1.

Technical Reference(s): LCO 3.3.16, Table 3.3.16-1 ANTICIPATORY REACTOR TRIP SYSTEM INSTRUMENTATION; DB-SS—4150, MAIN TURBINE STOP VALVE TEST; DB-SS-04159, ONLINE ELECTRICAL TRIP DEVICE TEST

Reference(s) Provided: None

Learning Objective:

Question Source: Bank # X

Modified Bank # _____ (note changes or attach parent)

New _____

Question History: Last NRC Exam 2021 SRO Q18 (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge X

Comprehension / Analysis / **H**igher Cognitive _____

10 CFR Part 55 Content: 55.41 _____

55.43 (b)(2)

Comments:

Examination Outline Cross-Reference: Q93

Level	RO	SRO
Tier #		2
Group #		2
K/A #	068 A2.04	
Importance Rating		3.8

K/A Statement: Ability to (a) predict the impacts of the following on the LIQUID RADWASTE SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Failure of automatic isolation

Proposed Question: Q93

93. The crew is performing a Miscellaneous Waste Monitor Tank (MWMT) release in accordance with section 4.3 of DB-OP-03011, RADIOACTIVE LIQUID BATCH RELEASE.

The following release parameters exist:
- Actual Maximum Release Rate (MRR) is 100 gpm
- HS 1877 is in the DISCHARGE B position
- RE1878B is functional and aligned
- RE1878A is NON-Functional, all required actions have been completed to start the release

During the release, 7-1-D, MISC WST SYS OUT RAD HI alarms, and the following conditions are received:
- WM1876, MISC WST DISCH ISO VALVE, indicates OPEN
- WM1877A, MISC WST DISCH LOW FLOW CONTROL VLV, indicates CLOSED
- WM1877B, MISC WST DISCH HIGH FLOW CONTROL VLV, indicates OPEN
- RE1878B is observed to be failed high
- MWMT Transfer pump is running

Based on the above conditions:
- WM1876 and WM1877B should have CLOSED **AND** the MMWT Transfer pump will ____(1)____.
- The Command SRO will direct the crew to manually position components which should have repositioned and then, in accordance with DB-OP-03011, ____(2)____.

A. (1) continue to RUN
(2) direct Chemistry to recalculate the MRR and change high alarm setpoints proportionally per Attachment 22, RESPONSE TO A RE WARN OR HIGH ALARM, then proceed with the release

B. (1) trip OFF
(2) direct Chemistry to recalculate the MRR and change high alarm setpoints proportionally per Attachment 22, RESPONSE TO A RE WARN OR HIGH ALARM, then proceed with the release

C. (1) continue to RUN
(2) restore the system to a normal line-up, complete the procedure for record close out

Examination Outline Cross-Reference: Q93

and initiate a new release permit to complete the release

D. (1) trip OFF
(2) restore the system to a normal line-up, complete the procedure for record close out and initiate a new release permit to complete the release

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Proposed Answer: **C**

Explanation: Q93

A. Incorrect – Plausible as Part (1) is correct. Part (2) is wrong since this is a failure of the RE's to automatically close the discharge valves. They will be declared non-functional. The release cannot proceed unless ODCM actions are taken. L&P 2.1.7 states, "If for any reason, any liquid release must be stopped prior to its completion, careful consideration should be given to system restoration to a normal lineup, i.e. pumps shutdown, valve lineups restored."

B. Incorrect – Plausible as in other instances when isolation valves are shut the supporting pump will receive a trip signal. Part (2) is wrong. See A explanation.

C. Correct – The MWMT Transfer pump does not trip OFF. The release will be terminated due to a failure of RE1878B to close the discharge valves. ODCM actions will be required to start a new release if the RE's are not fixed.

D. Incorrect – Plausible as in other instances when isolation valves are shut the supporting pump will receive a trip signal and Part (2) is correct.

Technical Reference(s): DB-OP-03011, RADIOACTIVE LIQUID BATCH RELEASE; DB-OP-06111, MISCELLANEOUS LIQUID WASTE SYSTEM; 7-1-D MISC WST SYS OUT RAD HI; OS-29, OPERATIONAL SCHEMATIC MISCELLANEOUS LIQUID RADWASTE SYSTEM, Sheets 1 and 2. ODCM.

Reference(s) Provided: None

Learning Objective:

Question Source: Bank # _____

Modified Bank # _____ (note changes or attach parent)

New X

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge ___

Comprehension / Analysis / **H**igher Cognitive X

10 CFR Part 55 Content: 55.41 _____

55.43 (b)(5)

Comments:

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Examination Outline Cross-Reference: Q94

Level	RO	SRO
Tier #		3
Group #		
K/A #	G2.1.36	
Importance Rating		4.1

K/A Statement: Knowledge of procedures and limitations involved in core alterations

Proposed Question: Q94

94. You are the Fuel Handling Director on the Main Fuel Handling Bridge over the Reactor Cavity.

A spent fuel assembly is in the main mast being transferred over the reactor core to its new location when a report is received that a Steam Generator nozzle dam has failed and refueling cavity level is lowering.

In accordance with DB-OP-00030, FUEL HANDLING OPERATIONS you MUST direct the Main Bridge Operator to place the fuel into which one of the following locations?

A. Refueling canal racks.

B. Transfer mechanism basket in the vertical position.

C. Spent fuel pool (SFP) and then install the SFP gate.

D. Back into the core.

Davis-Besse 2024 NRC Written Exam

Proposed Answer: **D**

Explanation: Q94

A. Incorrect – Plausible as this is a location in the refuel cavity, but DB-OP-00030 ATT 1 requires all fuel assemblies be removed from the refuel canal racks under these conditions.

B. Incorrect – Plausible as the transfer mechanism basket in the lowered or horizontal position is a permitted location for the fuel assembly.

C. Incorrect – Plausible as installing the SFP gate is an immediate action under these conditions but moving the fuel assembly to the SFP first is an unnecessary delay.

D. Correct – Lowering the spent fuel back into the reactor core ensure adequate shielding as level lowers in the reactor cavity due to the failure of the SG nozzle dam.

Technical Reference(s): DB-OP-00030, FUEL HANDLING OPERATIONS, Attachment 1.

Reference(s) Provided: None

Learning Objective:

Question Source: Bank # _____

Modified Bank # _____ (note changes or attach parent)

New X

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge X

Comprehension / Analysis / **H**igher Cognitive _____

10 CFR Part 55 Content: 55.41 _____

55.43 (b)(6)

Comments:

Davis-Besse 2024 NRC Written Exam

Examination Outline Cross-Reference: Q95

Level	RO	SRO
Tier #		3
Group #		
K/A #	G2.1.39	
Importance Rating		4.3

K/A Statement: CONDUCT OF OPERATIONS: Knowledge of conservative decision-making practices

Proposed Question: Q95

95. A severe thunderstorm WATCH has been issued for Ottawa County. The Shift Manager SHALL _____.

A. stop ALL work until a YELLOW Risk Management Plan is in place

B. stop work on equipment Danger Tagged OOS, that ONLY impacts PRA Risk

C. post protected equipment barriers on offsite and onsite power equipment

D. determine on a case-by-case basis if any Risk Management Actions will be taken

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Proposed Answer: **D**

Explanation: Q95

A. Incorrect – Plausible as this would be required for a Severe Thunderstorm WARNING.

B. Incorrect – PRA Risk activities should typically be completed. This is assuming that completing this work will improve the Risk margin by making the equipment Available.

C. Incorrect – Plausible because protected equipment barriers on onsite and offsite equipment are commonly used as a risk mitigating action for a severe weather WARNING.

D. Correct – In accordance with NOP-OP-1007, Section 4.2.5.1 the Shift Manger must determine on a case-by-case basis any Risk Management Actions that will be taken during severe weather WATCHES.

Technical Reference(s): NOP-OP-1007, RISK MANAGEMENT, Step 4.2.5.1
NOP-OP-1002 Step 4.11.2.4

Reference(s) Provided: None

Learning Objective:

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

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge X
Comprehension / Analysis / **H**igher Cognitive _____

10 CFR Part 55 Content: 55.41 _____
55.43 (b)(5)

Comments:

Examination Outline Cross-Reference: Q96

Level	RO	SRO
Tier #		3
Group #		
K/A #	G2.2.23	
Importance Rating		4.6

K/A Statement: EQUIPMENT CONTROL: Ability to track TS limiting conditions for operation

Proposed Question: Q96

96. The Reactor is at 100% RTP.
- At 0345 on February 8, HPI Pump 1 is declared inoperable.
- At 0700 on February 8, HPI Pump 2 is declared inoperable.
- At 0730 on February 8, HPI Pump 1 is restored to operable status.

Including any extensions that are permitted by Technical Specifications, what is the LATEST time and date to restore HPI Subsystem 2 to operable status without requiring a unit shutdown to commence?

REFERENCE PROVIDED

A. 0345 on February 11

B. 0700 on February 11

C. 0345 on February 12

D. 0700 on February 12

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Proposed Answer: **B**

Explanation: Q96

A. Incorrect – plausible since this is the completion time required to be tracked with one pump inoperable.

B. Correct – Tech Spec section 1.3 allows a 24 hour extension from the first pump inoperable completion time but not to exceed the stated completion time as measured from the discovery of the subsequent operability. The total Completion Time allowed for completing a Required Action to address the subsequent inoperability shall be limited to the ***more restrictive*** of either: The stated Completion Time, as measured from the initial entry into the Condition, plus an additional 24 hours; or the stated Completion Time as measured from discovery of the subsequent inoperability.

C. Incorrect – plausible since this would be the time if a 24 hour extension was allowed from the first pump.

D. Incorrect – plausible since this would be the time if a 24 hour extension was allowed from the second pump.

Technical Reference(s): Davis-Besse Improved Technical Specifications, Section 1.3 – COMPLETION TIMES; TS 3.5.2 – ECCS, OPERATING

Reference(s) Provided: **TS Section 1.3 Completion Times**

Learning Objective:

Question Source: Bank # X

Modified Bank # _____ (note changes or attach parent)

New _____

Question History: Last NRC Exam DB 2016 Q97 (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge _____

Comprehension / Analysis / **H**igher Cognitive X

10 CFR Part 55 Content: 55.41 _____

55.43 (b)(2)

Comments:

Examination Outline Cross-Reference: Q97

Level	RO	SRO
Tier #		3
Group #		
K/A #	G2.2.25	
Importance Rating		4.2

K/A Statement: Knowledge of the bases in TS for limiting conditions for operation and safety limits (SRO Only)

Proposed Question: Q97

97. AXIAL POWER IMBALANCE limits as described in the COLR are established to ensure compliance with T.S. Safety Limit (SL) 2.1.1.1 and Limiting Condition for Operation (LCO) 3.2.3.

SL 2.1.1.1: In MODES 1 and 2, the combination of reactor THERMAL POWER and AXIAL POWER IMBALANCE shall not exceed the protective limit shown in the COLR for the various combinations of three and four reactor coolant pump operation.

LCO 3.2.3: AXIAL POWER IMBALANCE shall be maintained within the limits specified in the COLR.

The bases for the limits for Safety Limit 2.1.1.1 ensure ____(1)____ .
The bases for LCO 3.2.3 ____(2)___ .

A. (1) fuel centerline temperature and DNBR are within limits
(2) limit the core power distribution based on accident initial condition criteria

B. (1) reactivity accidents do not result in damage to the Reactor Coolant Pressure Boundary
(2) limit the core power distribution based on accident initial condition criteria

C. (1) fuel centerline temperature and DNBR are within limits
(2) ensure limits defined by the Shut Down Margin (SDM) limits are NOT violated

D. (1) reactivity accidents do not result in damage to the Reactor Coolant Pressure Boundary
(2) ensure limits defined by the Shut Down Margin (SDM) limits are NOT violated

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Proposed Answer: **A**

Explanation: Q97

A. Correct – The fuel centerline melt and DNBR fuel design limits are not directly monitored by installed plant instrumentation. Instead, monitoring the process variable AXIAL POWER IMBALANCE ensures that the core operates within the fuel design criteria. With AXIAL POWER IMBALANCE within the protective limits of the SAFETY LIMITS, fuel centerline temperature and DNBR are also within limits. The AXIAL POWER IMBALANCE operating limits in LCO 3.2.3, also specified in the COLR, limit the core power distribution based on accident initial condition criteria but are not reactor core SLs.

B. Incorrect – Plausible as Part (1) discusses reactivity accidents. However, this is the bases for SAFETY LIMIT 2.1.2.
Part (2) is correct.

C. Incorrect – Plausible as Part (1) is correct.
Part (2) is plausible as it is the bases for LCO 3.2.1 ROD INSERTION LIMITS.

D. Incorrect – Plausible as Part (1) discusses reactivity accidents. However, this is the bases for SAFETY LIMIT 2.1.2.
Part (2) is plausible as it is the bases for LCO 3.2.1 ROD INSERTION LIMITS.

Technical Reference(s): Improved Technical Specifications Bases Section 2.1 - SAFETY LIMITS; LCO 3.2.3 - AXIAL POWER IMBALANCE.

Reference(s) Provided: None

Learning Objective:

Question Source: Bank # _____

Modified Bank # _____ (note changes or attach parent)

New X

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge X

Comprehension / Analysis / **H**igher Cognitive _____

10 CFR Part 55 Content: 55.41 _____

55.43 (b)(2)

Comments:

Davis-Besse 2024 NRC Written Exam

Examination Outline Cross-Reference: Q98

Level	RO	SRO
Tier #		3
Group #		
K/A #	G2.3.06	
Importance Rating		3.8

K/A Statement: RADIATION CONTROL: Ability to approve liquid or gaseous release permits

Proposed Question: Q98

98. The crew is preparing to perform a liquid release of Clean Waste Monitor Tank 1 (CWMT 1).

Which condition would require the Shift Manager to NOT permit the release to occur?

A. A release of the Miscellaneous Waste Monitor Tank is already in progress.

B. Both radiation monitors RE1770A and RE1770B are NON-functional.

C. Both CWMT release rate flow meters FI1700A and FQI1700A are NON-functional.

D. Total dilution flow computer point F201 is NON-functional.

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Proposed Answer: **A**

Explanation: Q98

A. Correct – Per step 4.9.1 for approval of a CWMT1 release is to verify that NO other radioactive liquid release is in progress.

B. Incorrect – Plausible as radiation monitors RE1770A and B are vital to performing a controlled safe liquid release of the CWMT1, but the ODCM and DB-OP-03011 allows a release to occur if both channels are non-functional as along as two independent tank samples are performed, two independent release rate calculations are performed, and two independent discharge valve lineups are performed.

C. Incorrect – Plausible as every effort should be made to restore at least one of the flow instruments before a release is performed, but per the ODCM and DB-OP-03011, a release may take place if flow rate is estimated every 4 hours based on CWMT1 volume changes.

D. Incorrect – Plausible as every effort should be made to restore F201 prior to a liquid release, but the ODCM and DB-OP-03011 permit a release if F201 is OOS as long as flow to collection box is estimated every four hours using known collection box input data.

Technical Reference(s): DB-OP-03011, RADIOACTIVE LIQUID BATCH RELEASE, Step 4.9.1, for performing CWMT1 release, verify NO other radioactive liquid release is being performed.

Reference(s) Provided None

Learning Objective:

Question Source: Bank # _____

Modified Bank # _____ (note changes or attach parent)

New X

Question History: Last NRC Exam _____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge X

Comprehension / Analysis / **H**igher Cognitive _____

10 CFR Part 55 Content: 55.41 _____

55.43 (b)(4)

Comments:

Examination Outline Cross-Reference: Q99

Level	RO	SRO
Tier #		3
Group #		
K/A #	G2.4.05	
Importance Rating		4.3

K/A Statement: EMERGENCY PROCEDURES/PLAN: Knowledge of the organization of the operating procedures network for normal, abnormal, and emergency evolutions

Proposed Question: Q99

99. The Plant is operating at 100% RTP when the following occurs:
    - Multiple Fire Alarms are received from Room 603, CONTROL ROOM AC EQUIPMENT ROOM, Fire Area HH
    - The Fire Brigade is dispatched in accordance with DB-OP-02529, FIRE PROCEDURE
    - The Fire Brigade Captain reports a significant fire is in progress and requests off-site assistance
    - The ATC Reactor Operator reports High Pressure Injection Pump 2 and Containment Spray Pump 2 have spuriously started

    Subsequently, the Reactor trips.
    - NO other effects of the fire are indicated at this time

    Which of the following procedures should the Command SRO transition to NEXT?

    A. DB-OP-02000, RPS, SFAS, SFRCS TRIP OR SG TUBE RUPTURE

    B. DB-OP-02501, SERIOUS STATION FIRE

    C. DB-OP-02508, CONTROL ROOM EVACUATION

    D. DB-OP-02519, SERIOUS CONTROL ROOM FIRE

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Proposed Answer: **B**

Explanation: Q99

A. Incorrect – Plausible because in general, the correct procedure to implement following a Reactor Trip is DB-OP-02000.

B. Correct – Spurious operation of safety related equipment requires implementation of DB-OP-02501 which takes priority over DB-OP-02000. See DB-OP-01003, OPERATIONS PROCEDURE USE INSTRUCTIONS step 6.5.2.a.

C. Incorrect – Plausible because DB-OP-02519, Serious Station Fire Attachment 20 for Fire Area HH directs use of DB-OP-02508, CONTROL ROOM EVACUATION if the fire in area HH affects Control Room Habitability. In addition, a fire in the Control Room AC area could introduce smoke into the Control Room.

D. Incorrect – plausible because a fire in the Control Room AC area could introduce smoke into the Control Room, however the Control Room circuits would not be involved in the fire which would require use of DB-OP-02519, SERIOUS CONTROL ROOM FIRE.

Technical Reference(s): DB-OP-01003, OPERATIONS PROCEDURE USE INSTRUCTIONS, Step 6.5.2.a. DB-OP-02501 Step 2.1.1 symptoms for Serious Station Fire.

Reference(s) Provided None

Learning Objective:

Question Source: Bank # X

Modified Bank # _____ (note changes or attach parent)

New _____

Question History: Last NRC Exam DB 2015 Q99 (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge _____

Comprehension / Analysis / **H**igher Cognitive X

10 CFR Part 55 Content: 55.41 _____

55.43 (b)(5)

Comments:

Davis-Besse 2024 NRC Written Exam

Examination Outline Cross-Reference: Q100

Level	RO	SRO
Tier #		3
Group #		
K/A #	G2.4.40	
Importance Rating		4.5

K/A Statement: Knowledge of SRO responsibilities in emergency plan implementing procedures (SRO Only)

Proposed Question: Q100

100. You are the licensed senior operator in the Control Room during an Emergency Plan activation that resulted from a tornado striking several locations on site.

The Shift Manager (Emergency Director) has directed you to perform an off-site dose assessment.

- Davis-Besse Unified RASCAL Interface (URI) Dose Assessment Software is NOT available.

As a result of the tornado strike, the following Meteorological Tower data is NOT functioning:

- Upper delta temperature
- Lower delta temperature
- All 10-meter wind direction indications

Given this set of circumstances, Stability Class _____ MUST be used during calculations of off-site dose.

A. F – Moderately stable

B. D – Neutral

C. B – Moderately unstable

D. A – Extremely unstable

Davis-Besse 2024 NRC Written Exam

Proposed Answer: **A**

Explanation: Q100

A. Correct – Classes range from A to G. Class F is directed to be used by RA-EP-02240 if no indications are available.

B. Incorrect – Plausible since Class D is neutral (middle of scale).

C. Incorrect – Plausible since Class B and Class F (the answer) both use moderately in their description.

D. Incorrect – Plausible since Class A is most unstable. The candidate may think this is the most conservative since it's the only distractor with "Extremely" in its description.

Technical Reference(s): RA-EP-02240 R4, Offsite Dose Assessment (p40), Attachment 7, Meteorological Data, 4.d

Reference(s) Provided None

Learning Objective:

Question Source: Bank # X

Modified Bank # ____ (note changes or attach parent)

New ____

Question History: Last NRC Exam ____ (if applicable)

Question Cognitive Level: Memory / **F**undamental Knowledge X

Comprehension / Analysis / **H**igher Cognitive ____

10 CFR Part 55 Content: 55.41 ____

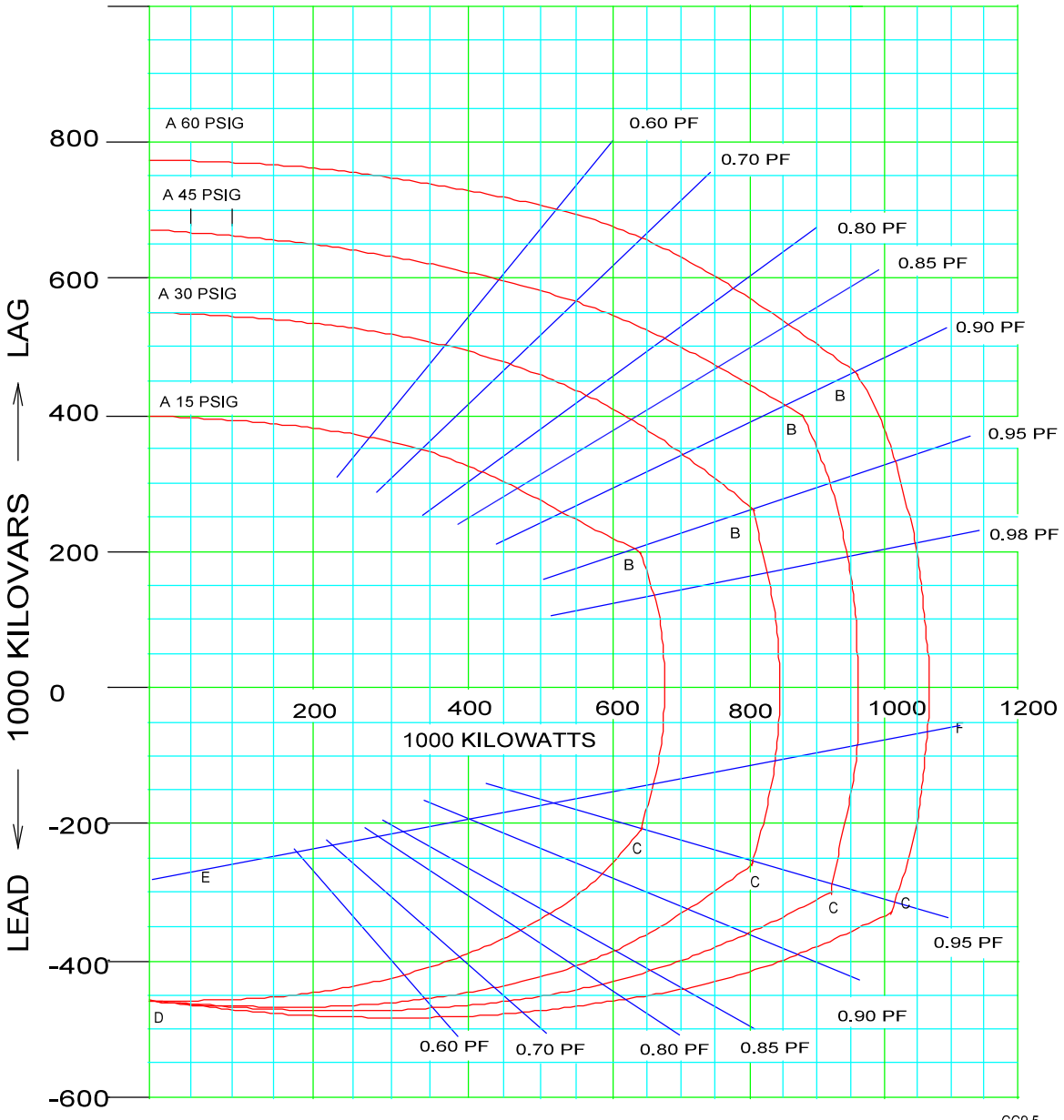
55.43 (b)(5)

Comments:

DB EXAM BANK ID: 290429

ESTIMATED CAPABILITY CURVES - LEAD-LAG

ATB 4 POLE, 1,068,794 KVA, 1800 RPM,
25,000 VOLTS, 0.90 PF, 0.58 SCR
60 PSIG HYDROGEN PRESSURE, 500 VOLTS EXCITATION



CURVE AB LIMITED BY FIELD HEATING
CURVE BC LIMITED BY ARMATURE HEATING
CURVE CD LIMITED BY ARMATURE CORE END HEATING
CURVE EF LIMITED BY GENERATOR AUTOMATIC VOLTAGE
REGULATOR (RFA 91-0285)

H — Hazards

1 Security

GENERAL EMERGENCY: None

SITE AREA EMERGENCY

HS1 HOSTILE ACTION within the PROTECTED AREA
1 2 3 4 5 6 DEF

HS1.1
A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervisor.

ALERT

HA1 HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes
1 2 3 4 5 6 DEF

HA1.1
A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the Security Shift Supervisor.

HA1.2
A validated notification from NRC of an aircraft attack threat within 30 min. of the site.

UNUSUAL EVENT

HU1 Confirmed SECURITY CONDITION or threat
1 2 3 4 5 6 DEF

HU1.1
A SECURITY CONDITION that does **not** involve a HOSTILE ACTION as reported by the Security Shift Supervisor.

HU1.2
Notification of a credible security threat directed at the site.

HU1.3
A validated notification from the NRC providing information of an aircraft threat.

2 Seismic Event

GENERAL EMERGENCY: None

SITE AREA EMERGENCY: None

ALERT: None

UNUSUAL EVENT

HU2 Seismic event greater than OBE levels
1 2 3 4 5 6 DEF

HU2.1
Seismic event > OBE as indicated by OBE alarm on seismic alarm panel C5764A.

3 Natural or Tech. Hazard

GENERAL EMERGENCY: None

SITE AREA EMERGENCY: None

ALERT: None

UNUSUAL EVENT

HU3 Hazardous event
1 2 3 4 5 6 DEF

HU3.1
A tornado strike within the PROTECTED AREA.

HU3.2
Internal room or area FLOODING of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component needed for the current operating mode.

HU3.3
Movement of personnel within the PROTECTED AREA is IMPEDED due to an offsite event involving hazardous materials (e.g., an offsite chemical spill or toxic gas release).

HU3.4
A hazardous event that results in on-site conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles (Note 7).

Notes

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Note 3: If the effluent flow past an effluent monitor is known to have stopped, indicating that the release path is isolated, the effluent monitor reading is no longer VALID for classification purposes.

Note 4: The pre-calculated effluent monitor values presented in EALs RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Note 5: If the equipment in the listed room or area was already inoperable or out-of-service before the event occurred, then no emergency classification is warranted.

Note 6: If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, declaration of a General Emergency is not required.

Note 7: This EAL does not apply to routine traffic impediments such as fog, snow, ice, or vehicle breakdowns or accidents.

Note 8: A manual trip action is ANY operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and **does not** include manually driving in control rods or implementation of boron injection strategies.

Note 9: During a main steam line break in containment or LOCA with temperature >170F, there is a potential to induce transient errors into the output of RE4596A and B during the peak rate of temperature change. Consult alternate indications. If the main steam line break is accompanied by core damage this error is insignificant.

Note 10: In the absence of reliable RCS temperature indication caused by the loss of decay heat removal capability, classification should be based on the RCS pressure increase criteria when the RCS is intact in Mode 5 or based on time to boil data when in Mode 6 or the RCS is not intact in Mode 5.

4 Fire

GENERAL EMERGENCY: None

SITE AREA EMERGENCY: None

ALERT

Table H-1 Safe Shutdown Fire Areas

- Containment
- Control Room
- Auxiliary Building
- Intake Structure
- Borated Water Storage Tank

UNUSUAL EVENT

HU4 FIRE potentially degrading the level of safety of the plant
1 2 3 4 5 6 DEF

HU4.1
A FIRE is **not** extinguished within 15 min. of **ANY** of the following FIRE detection indications (Note 1):
- Report from the field (i.e., visual observation)
- Receipt of multiple (more than 1) fire alarms or indications
- Field verification of a single fire alarm

AND
The FIRE is located within **ANY** Table H-1 area.

HU4.2
Receipt of a single fire alarm (i.e., no other indications of a FIRE).
AND
The fire alarm is indicating a FIRE within **ANY** Table H-1 area.
AND
The existence of a FIRE is **not** verified within 30 min. of alarm receipt (Note 1).

HU4.3
A FIRE within the plant PROTECTED AREA **not** extinguished within 60 min. of the initial report, alarm or indication (Note 1).

HU4.4
A FIRE within the plant PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish.

5 Hazardous Gas

GENERAL EMERGENCY: None

SITE AREA EMERGENCY: None

ALERT

HA5 Gaseous release IMPEDING access to equipment necessary for normal plant operations, cooldown or shutdown
1 2 3

HA5.1
Release of a toxic, corrosive, asphyxiant or flammable gas into **ANY** Table H-2 Safe Shutdown Rooms or Areas.
AND
Entry into the room or area is prohibited or IMPEDED. (Note 5)

UNUSUAL EVENT

Table H-2 Safe Shutdown Rooms/Areas

Room/Area	Mode Applicability
Aux Bldg. 565' ele. Room 236 #2 Mechanical Penetration Room	1, 2, 3
Aux Bldg. 585' ele. Room 304 corridor outside #3 Mechanical Penetration Room	1, 2, 3
Aux Bldg. 603' ele. Room 427 - #2 Electrical Penetration Room	1, 2, 3

6 Control Room Evacuation

GENERAL EMERGENCY: None

SITE AREA EMERGENCY

HS6 Inability to control a key safety function from outside the Control Room
1 2 3 4 5 6

HS6.1
An event has resulted in plant control being transferred from the Control Room to the Auxiliary Shutdown Panel.
AND
Control of **ANY** of the following key safety functions is not reestablished within 15 min. (Note 1):
- Reactivity (modes 1, 2 and 3 **only**)
- Core cooling (RCS inventory)
- RCS heat removal (ability to maintain heat sink)

ALERT

HA6 Control Room evacuation resulting in transfer of plant control to alternate locations
1 2 3 4 5 6 DEF

HA6.1
An event has resulted in plant control being transferred from the Control Room to the Auxiliary Shutdown Panel.

UNUSUAL EVENT

7 Emergency Coordinator Judgment

GENERAL EMERGENCY

HG7 Other conditions exist which in the judgment of the Emergency Coordinator warrant declaration of a General Emergency
1 2 3 4 5 6 DEF

HG7.1
Other conditions exist which in the judgment of the Emergency Coordinator indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

SITE AREA EMERGENCY

HS7 Other conditions exist which in the judgment of the Emergency Coordinator warrant declaration of a Site Area Emergency
1 2 3 4 5 6 DEF

HS7.1
Other conditions exist which in the judgment of the Emergency Coordinator indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts, (1) toward site personnel or equipment that could lead to the likely failure of or, (2) that prevent effective access to equipment needed for the protection of the public. **ANY** releases are **not** expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the SITE BOUNDARY.

ALERT

HA7 Other conditions exist which in the judgment of the Emergency Coordinator warrant declaration of an Alert
1 2 3 4 5 6 DEF

HA7.1
Other conditions exist which, in the judgment of the Emergency Coordinator, indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. **ANY** releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

UNUSUAL EVENT

HU7 Other conditions exist which in the judgment of the Emergency Coordinator warrant declaration of a UE
1 2 3 4 5 6 DEF

HU7.1
Other conditions exist which in the judgment of the Emergency Coordinator indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. **No** releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of SAFETY SYSTEMS occurs.

R — Abnormal Rad Release / Rad Effluent

1 Rad Effluent

GENERAL EMERGENCY

RG1 Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE
1 2 3 4 5 6 DEF

RG1.1
Station Vent Channel 1 Noble Gas (RE 4598 AB/BB) > 8.4E +00 µCi/cc for 15 minutes or longer. (Notes 1, 2, 3, 4)

RG1.2
Dose assessment using actual meteorology indicates doses > 1000 mrem TEDE or 5000 mrem thyroid CDE at or beyond the SITE BOUNDARY. (Note 4)

RG1.3
Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY:
- Closed window dose rates > 1000 mR/hr expected to continue for ≥ 60 min.
- Analyses of field survey samples indicate thyroid CDE > 5000 mrem for 60 min. of inhalation.

(Notes 1, 2)

SITE AREA EMERGENCY

RS1 Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE
1 2 3 4 5 6 DEF

RS1.1
Station Vent Channel 1 Noble Gas (RE 4598 AB/BB) > 8.4E -01 µCi/cc for 15 minutes or longer. (Notes 1, 2, 3, 4)

RS1.2
Dose assessment using actual meteorology indicates doses > 100 mrem TEDE or 500 mrem thyroid CDE at or beyond the SITE BOUNDARY. (Note 4)

RS1.3
Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY:
- Closed window dose rates > 100 mR/hr expected to continue for ≥ 60 min.
- Analyses of field survey samples indicate thyroid CDE > 500 mrem for 60 min. of inhalation.

(Notes 1, 2)

ALERT

RA1 Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE
1 2 3 4 5 6 DEF

RA1.1
Station Vent Channel 1 Noble Gas (RE 4598 AB/BB) > 8.4E -02 µCi/cc for 15 minutes or longer. (Notes 1, 2, 3, 4)

RA1.2
Dose assessment using actual meteorology indicates doses > 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the SITE BOUNDARY. (Note 4)

RA1.3
Analysis of a liquid effluent sample indicates a concentration or release rate that would result in doses > 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the SITE BOUNDARY for 60 min. of exposure. (Notes 1, 2)

RA1.4
Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY:
- Closed window dose rates > 10 mR/hr expected to continue for ≥ 60 min.
- Analyses of field survey samples indicate thyroid CDE > 50 mrem for 60 min. of inhalation.

(Notes 1, 2)

UNUSUAL EVENT

RU1 Release of gaseous or liquid radioactivity > 2 times the ODCM limits for 60 minutes or longer
1 2 3 4 5 6 DEF

RU1.1
Station Vent Channel 1 Noble Gas (RE 4598AA/BA) > 5.72E-03 µCi/cc for 60 minutes or longer.

OR

ANY of the following effluent monitors > 2 times the high alarm setpoint established by a current radioactivity discharge permit for 60 minutes or longer:
- Waste Gas System Outlet (RE 1822A or B).
- Clean Waste System Outlet (RE 1770A or B).
- Miscellaneous Waste System Outlet (RE 1878A or B).
- Discharge permit specified monitor.

(Notes 1, 2, 3)

RU1.2
Sample analyses for a gaseous or liquid release indicates concentration or release rate > 2 x ODCM limits for ≥ 60 min. (Notes 1, 2)

2 Irradiated Fuel Event

GENERAL EMERGENCY

RG2 Spent fuel pool level cannot be restored to at least the top of the fuel racks for 60 minutes or longer
1 2 3 4 5 6 DEF

RG2.1
Spent fuel pool level **cannot** be restored to at least 1 ft. for ≥ 60 min. (Note 1)

SITE AREA EMERGENCY

RS2 Spent fuel pool level at the top of the fuel racks
1 2 3 4 5 6 DEF

RS2.1
Lowering of spent fuel pool level to 1 ft.

ALERT

RA2 Significant lowering of water level above, or damage to, irradiated fuel
1 2 3 4 5 6 DEF

RA2.1
Uncovery of irradiated fuel in the REFUELING PATHWAY

RA2.2
Damage to irradiated fuel resulting in a release of radioactivity as indicated by a high radiation alarm on **ANY** of the following radiation monitor indications:
- RE 8426 SFP Area
- RE 8427 SFP Area
- RE 8417 Fuel Handling Area
- RE 8418 Fuel Handling Area
- RE 8425 Equipment Hatch Area
- RE 8446/8447 Fuel Handling Exhaust
- RE 4598AA/BA Station Vent

RA2.3
Lowering of spent fuel pool level to 10 ft.

UNUSUAL EVENT

RU2 UNPLANNED loss of water level above irradiated fuel
1 2 3 4 5 6 DEF

RU2.1
UNPLANNED water level drop in the REFUELING PATHWAY as indicated by **ANY** of the following:
- SFP level alarm or indication (LI 1600)
- SFP level indication (LI 4801 A/B)
- Refueling Canal low water level alarm or indication (LI 1627)
- Visual observation

AND
UNPLANNED rise in corresponding area radiation levels as indicated by **ANY** of the following radiation monitors:
- RE 8426 SFP Area
- RE 8427 SFP Area
- RE 8417 Fuel Handling Area
- RE 8418 Fuel Handling Area
- RE 8425 Equipment Hatch Area

3 Area Radiation Levels

GENERAL EMERGENCY: None

SITE AREA EMERGENCY: None

Table R-2 Safe Shutdown Rooms/Areas

Room/Area	Mode Applicability
Aux Bldg. 565' ele. Room 236 #2 Mechanical Penetration Room	1, 2, 3
Aux Bldg. 585' ele. Room 304 corridor outside #3 Mechanical Penetration Room	1, 2, 3
Aux Bldg. 603' ele. Room 427 - #2 Electrical Penetration Room	1, 2, 3

ALERT

RA3 Radiation levels that IMPEDE access to equipment necessary for normal plant operations, cooldown or shutdown

RA3.1 1 2 3 4 5 6 DEF
Dose rate > 15 mR/hr in **EITHER** of the following areas:
- Control Room (RE 8430 or 8431)
- Central Alarm Station (RE 8435 or 8436)

RA3.2 1 2 3
An UNPLANNED event results in radiation levels that prohibit or IMPEDE access to **ANY** Table R-2 Safe Shutdown Rooms or Areas. (Note 5)

UNUSUAL EVENT: None

E — DFSF

1 Confinement Boundary

GENERAL EMERGENCY: None

SITE AREA EMERGENCY: None

ALERT: None

UNUSUAL EVENT

EU1 Damage to a loaded cask CONFINEMENT BOUNDARY
1 2 3 4 5 6 DEF

EU1.1
Damage to a loaded canister CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading on the surface of a loaded spent fuel HSM cask > **EITHER** of the following:
- 100 mrem/hr (neutron + gamma) on the HSM cask wall or roof
- 100 mrem/hr (neutron + gamma) on the center of the HSM cask door

S — System Malfunct.

1 Loss of Essential AC Power

GENERAL EMERGENCY

SG1a Prolonged loss of **ALL** offsite and **ALL** onsite AC power to essential buses
1 2 3 4

SG1.1
Loss of **ALL** offsite and **ALL** onsite AC power capability to essential 4160V buses C1 and D1

AND EITHER:
- Restoration of at least one essential bus in < 4 hours is **not** likely (Note 1)
- Calculated Clad Temperature in **Region 3** (DB-OP-02000 Figure 2)

SG1b Loss of **ALL** essential AC and DC power sources for 15 minutes or longer
1 2 3 4

SG1.2
Loss of **ALL** offsite and **ALL** onsite AC power capability to essential 4160V buses C1 and D1 for ≥ 15 min.
AND
Loss of **ALL** 125 VDC power based on battery bus voltage indications < 105 VDC on **ALL** essential DC distribution panels D1P, D1N, D2P and D2N for ≥ 15 min.
(Note 1)

SITE AREA EMERGENCY

SS1 Loss of **ALL** offsite power and **ALL** onsite AC power to essential buses for 15 minutes or longer
1 2 3 4

SS1.1
Loss of **ALL** offsite and **ALL** onsite AC power capability to essential 4160V buses C1 and D1 for ≥ 15 min. (Note 1)

ALERT

SA1 Loss of **ALL** but one AC power source to essential buses for 15 minutes or longer
1 2 3 4

SA1.1
AC power capability, Table S-1, to essential 4160V buses C1 and D1 reduced to a single power source for ≥ 15 min. (Note 1)
AND
ANY additional single power source failure will result in loss of **ALL** AC power to SAFETY SYSTEMS

UNUSUAL EVENT

SU1 Loss of **ALL** offsite AC power capability to essential buses for 15 minutes or longer
1 2 3 4

SU1.1
Loss of **ALL** offsite AC power capability, Table S-1, to essential 4160V buses C1 and D1 for ≥ 15 min. (Note 1)

Table S-1 Offsite/Onsite AC Power Sources

Offsite:
- X11
- X11 (back-fed via Main Transformer if already aligned)
- X01
- X02

Onsite:
- EDG1
- EDG2
- SBODG

2 Loss of Essential DC Power

GENERAL EMERGENCY: None

SITE AREA EMERGENCY

SS2 Loss of **ALL** essential DC power for 15 minutes or longer
1 2 3 4

SS2.1
Loss of **ALL** 125 VDC power based on battery bus voltage indications < 105 VDC on **ALL** essential DC distribution panels D1P, D1N, D2P and D2N for ≥ 15 min. (Note 1)

ALERT: None

UNUSUAL EVENT: None

3 Loss of Control Room Indications

GENERAL EMERGENCY: None

SITE AREA EMERGENCY

Table S-2 Safety System Parameters

- Reactor power
- Pressurizer level
- RCS pressure
- In-core T/C temperature
- Level in at least one S/G
- Auxiliary or emergency feed flow

ALERT

SA3 UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress
1 2 3 4

SA3.1
An UNPLANNED event results in the inability to monitor one or more Table S-2 SAFETY SYSTEM parameters from within the Control Room for ≥ 15 min. (Note 1)
AND
ANY Significant Transient is in progress, Table S-3

UNUSUAL EVENT

SU3 UNPLANNED loss of Control Room indications for 15 minutes or longer
1 2 3 4

SU3.1
An UNPLANNED event results in the inability to monitor one or more Table S-2 SAFETY SYSTEM parameters from within the Control Room for ≥ 15 min. (Note 1)

4 RCS Activity

GENERAL EMERGENCY: None

SITE AREA EMERGENCY

Table S-3 Significant Transients

- Reactor trip
- Runback > 25% thermal power
- Electrical load rejection > 25% electrical load
- Safety injection actuation

ALERT: None

UNUSUAL EVENT

SU4 Reactor coolant activity greater than Technical Specification allowable limits
1 2 3 4

SU4.1
RCS activity > Technical Specification LCO 3.4.16 as indicated by **ANY** of the following:
- Dose equivalent I-131 in the unacceptable region of Figure 3.4.16-1
- > 1 µCi/gm dose equivalent I-131 for > 48 hrs.
- > 100/Ē µCi/gm gross specific coolant activity

5 RCS Leakage

GENERAL EMERGENCY: None

SITE AREA EMERGENCY: None

ALERT: None

UNUSUAL EVENT

SU5 RCS leakage for 15 minutes or longer
1 2 3 4

SU5.1
RCS unidentified or pressure boundary leakage > 10 gpm for ≥ 15 min.
OR
RCS identified leakage > 25 gpm for ≥ 15 min.
OR
Leakage from the RCS to a location outside containment > 25 gpm for ≥ 15 min.
(Note 1)

6 RPS Failure

GENERAL EMERGENCY: None

SITE AREA EMERGENCY

SS6 Inability to shut down the reactor causing a challenge to core cooling or RCS heat removal
1

SS6.1
An automatic or manual trip fails to shut down the reactor as indicated by reactor power ≥ 5%
AND
ALL actions to shut down the reactor are **not** successful as indicated by reactor power ≥ 5%
AND EITHER:
- Calculated Clad Temperature in **Region 3** (DB-OP-02000 Figure 2)
- MFW, AFW and MU-HPI PORV Cooling are **all** unavailable

ALERT

SA6 Automatic or manual trip fails to shut down the reactor **AND** subsequent manual actions taken at the Controls Area are **not** successful in shutting down the reactor
1

SA6.1
An automatic or manual trip fails to shut down the reactor as indicated by reactor power ≥ 5%
AND
Manual trip actions taken at the Controls Area (manual RPS trip pushbuttons and de-energizing E2 and F2) are **not** successful in shutting down the reactor as indicated by reactor power ≥ 5% (Note 8).

UNUSUAL EVENT

SU6 Automatic or manual trip fails to shut down the reactor
1

SU6.1
An automatic trip did **not** shut down the reactor as indicated by reactor power ≥ 5% after **ANY** RPS setpoint is exceeded
AND
A subsequent manual trip action taken at the Controls Area (manual RPS trip pushbuttons or de-energizing E2 and F2) is successful in shutting down the reactor as indicated by reactor power < 5% (Note 8).

SU6.2
A manual trip did **not** shut down the reactor as indicated by reactor power ≥ 5% after **ANY** manual trip action was initiated
AND
A subsequent automatic trip or manual trip action taken at the Controls Area (manual RPS trip pushbuttons or de-energizing E2 and F2) is successful in shutting down the reactor as indicated by reactor power < 5% (Note 8).

7 Loss of Comm.

GENERAL EMERGENCY: None

SITE AREA EMERGENCY: None

Table S-4 Communication Methods

System	Onsite	ORO	NRC
Public Address (Gaitronics)	X		
Onsite Radios	X		
Plant Telephones	X	X	X
Commercial Telephones	X	X	X
4-Way Ringdown Circuit		X	
NRC Emergency Telephone System			X

ALERT: None

UNUSUAL EVENT

SU7 Loss of **ALL** onsite or offsite communications capabilities
1 2 3 4

SU7.1
Loss of **ALL** Table S-4 onsite communication methods.

SU7.2
Loss of **ALL** Table S-4 ORO communication methods.

SU7.3
Loss of **ALL** Table S-4 NRC communication methods.

8 CMT Failure

GENERAL EMERGENCY: None

SITE AREA EMERGENCY: None

ALERT

Table S-6 Containment Cooling Full Train

CT Spray Pumps	CT Cooling Fans
2	0
1	1
0	2

UNUSUAL EVENT

SU8 Failure to isolate containment or loss of containment pressure control
1 2 3 4

SU8.1
ANY penetration is not closed within 15 min. of a VALID containment isolation signal (Note 1)

SU8.2
Containment pressure > 40 psia with < one full train of containment cooling, Table S-6, operating per design for > 15 min. (Note 1)

9 Hazardous Event Affecting Safety Systems

GENERAL EMERGENCY: None

SITE AREA EMERGENCY

Table S-5 Hazardous Events

- Seismic event (earthquake)
- Internal or external FLOODING event
- High winds or tornado strike
- FIRE
- EXPLOSION
- Other events with similar hazard characteristics as determined by the Emergency Coordinator

ALERT

SA9 Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode
1 2 3 4

SA9.1
The occurrence of **ANY** Table S-5 Hazardous Event
AND EITHER:
- Event has caused indications of degraded performance in at least one train of a SAFETY SYSTEM required for the current operating mode
- The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure required for the current operating mode

UNUSUAL EVENT: None

F — Fission Product Barrier Degradation

GENERAL EMERGENCY

FG1 1 2 3 4
Loss of **ANY** two barriers
AND
Loss or Potential Loss of third barrier (Table F-1)

SITE AREA EMERGENCY

FS1 1 2 3 4
Loss or Potential Loss of **ANY** two barriers (Table F-1)

ALERT

FA1 1 2 3 4
ANY Loss or **ANY** Potential Loss of **EITHER** Fuel Clad or RCS (Table F-1)

UNUSUAL EVENT: None

Table F-1 Fission Product Barrier Matrix

Category	Fuel Clad (FC) Barrier: Loss	Fuel Clad (FC) Barrier: Potential Loss	Reactor Coolant System (RCS) Barrier: Loss	Reactor Coolant System (RCS) Barrier: Potential Loss	Containment (CT) Barrier: Loss	Containment (CT) Barrier: Potential Loss
A RCS or SG Tube Leakage	None	None	1. An automatic or manual ECCS (SFAS) actuation required by **EITHER:** • UNISOLABLE RCS leakage • SG tube RUPTURE	1. Operation of a standby Makeup Pump (>250 gpm) is required by **EITHER:** • UNISOLABLE RCS leakage • SG tube leakage **OR** 2. PTS requirements invoked (SR5)	1. A leaking or RUPTURED SG is FAULTED outside of containment	None
B Inadequate Heat Removal	1. Calculated Clad Temperature in **Region 3** or higher (DB-OP-02000 Figure 2)	1. Calculated Clad Temperature in **Region 2** or higher (DB-OP-02000 Figure 2) **OR** 2. Loss of **ALL** feedwater **AND** SG cooling is required (In combination with RCS Potential Loss B.1, meeting this threshold would result in a Site Area Emergency)	None	1. Loss of **ALL** feedwater **AND** SG cooling is required (Meeting this threshold results in a Site Area Emergency because this threshold is identical to Fuel Clad Barrier Potential Loss threshold B.2; both will be met)	None	1. Calculated Clad Temperature in **Region 3** or higher (DB-OP-02000 Figure 2) **AND** Restoration procedures **not** effective within 15 min. (Note1)
C CT Radiation / RCS Activity	1. RE 4596A or B > Table F-2 column "FC Loss" (Note 9) **OR** 2. Dose equivalent I-131 coolant activity > 300 µCi/gm	None	1. RE 4596A or B > Table F-2 column "RCS Loss" (Note 9)	None	None	1. RE 4596A or B > Table F-2 column "CT Potential Loss" (Note 9)
D CT Integrity or Bypass	None	None	None	None	1. Containment isolation is required **AND EITHER:** • Containment integrity has been lost based on Emergency Coordinator judgment • UNISOLABLE pathway from Containment to the environment exists **OR** 2. Indications of RCS leakage outside of containment	1. Containment pressure > 50.4 psia **OR** 2. Containment hydrogen concentration > 4% **OR** 3. Containment pressure > 40 psia with < one full train, Table F-3, of containment cooling operating per design for ≥ 15 min. (Note 1)
E Emergency Coordinator Judgment	1. **ANY** condition in the opinion of the Emergency Coordinator that indicates Loss of the Fuel Clad Barrier	1. **ANY** condition in the opinion of the Emergency Coordinator that indicates Potential Loss of the Fuel Clad Barrier	1. **ANY** condition in the opinion of the Emergency Coordinator that indicates Loss of the RCS Barrier	1. **ANY** condition in the opinion of the Emergency Coordinator that indicates Potential Loss of the RCS Barrier	1. **ANY** condition in the opinion of the Emergency Coordinator that indicates Loss of the Containment Barrier	1. **ANY** condition in the opinion of the Emergency Coordinator indicates Potential Loss of the Containment Barrier

Table F-2 Containment Radiation R/hr (RE 4596A or B)

Time After S/D (Hrs.)	RCS Loss	FC Loss	CT Pot. Loss
0-1	1.50E+01	3.03E+03	1.40E+04
1-2	1.50E+01	2.56E+03	1.18E+04
2-8	1.50E+01	1.61E+03	7.46E+03
8-16	1.50E+01	1.14E+03	5.28E+03
16-24	1.50E+01	8.66E+02	4.00E+03
>24	1.50E+01	3.94E+02	1.82E+03

Table F-3 Containment Cooling Full Train

Spray	Coolers
2	0
1	1
0	2

Modes:

1	2	3	4	5	6	DEF
Power Operation	Startup	Hot Standby	Hot Shutdown	Cold Shutdown	Refueling	Defueled

Davis-Besse Nuclear Power Station

DBRM-EMER-1500B, Rev. 3
Davis-Besse EAL Wallboards

HOT CONDITIONS RCS > 200°F)

H — Hazards

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
1 Security	None	**HS1** HOSTILE ACTION within the PROTECTED AREA (1 2 3 4 5 6 DEF) **HS1.1** A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervisor.	**HA1** HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes (1 2 3 4 5 6 DEF) **HA1.1** A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the Security Shift Supervisor. **HA1.2** A validated notification from NRC of an aircraft attack threat within 30 min. of the site.	**HU1** Confirmed SECURITY CONDITION or threat (1 2 3 4 5 6 DEF) **HU1.1** A SECURITY CONDITION that does **not** involve a HOSTILE ACTION as reported by the Security Shift Supervisor. **HU1.2** Notification of a credible security threat directed at the site. **HU1.3** A validated notification from the NRC providing information of an aircraft threat.
2 Seismic Event	None	None	None	**HU2** Seismic event greater than OBE levels (1 2 3 4 5 6 DEF) **HU2.1** Seismic event > OBE as indicated by OBE alarm on seismic alarm panel C5764A.
3 Natural or Tech. Hazard	See Notes	None	None	**HU3** Hazardous event (1 2 3 4 5 6 DEF) **HU3.1** A tornado strike within the PROTECTED AREA. **HU3.2** Internal room or area FLOODING of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component needed for the current operating mode. **HU3.3** Movement of personnel within the PROTECTED AREA is IMPEDED due to an offsite event involving hazardous materials (e.g., an offsite chemical spill or toxic gas release). **HU3.4** A hazardous event that results in on-site conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles (Note 7).
4 Fire		None	**Table H-1 Safe Shutdown Fire Areas** • Containment • Control Room • Auxiliary Building • Intake Structure • Borated Water Storage Tank	**HU4** FIRE potentially degrading the level of safety of the plant (1 2 3 4 5 6 DEF) **HU4.1** A FIRE is **not** extinguished within 15 min. of **ANY** of the following FIRE detection indications (Note 1): • Report from the field (i.e., visual observation) • Receipt of multiple (more than 1) fire alarms or indications • Field verification of a single fire alarm **AND** The FIRE is located within **ANY** Table H-1 area. **HU4.2** Receipt of a single fire alarm (i.e., no other indications of a FIRE). **AND** The fire alarm is indicating a FIRE within **ANY** Table H-1 area. **AND** The existence of a FIRE is **not** verified within 30 min. of alarm receipt (Note 1). **HU4.3** A FIRE within the plant PROTECTED AREA **not** extinguished within 60 min. of the initial report, alarm or indication (Note 1). **HU4.4** A FIRE within the plant PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish.
5 Hazardous Gas	None	None	**HA5** Gaseous release IMPEDING access to equipment necessary for normal plant operations, cooldown or shutdown (1 2 3) **HA5.1** Release of a toxic, corrosive, asphyxiant or flammable gas into **ANY** Table H-2 Safe Shutdown Rooms or Areas. **AND** Entry into the room or area is prohibited or IMPEDED. (Note 5)	**Table H-2 Safe Shutdown Rooms/Areas** Area — Mode Aux Bldg. 565' ele. Room 236 #2 Mechanical Penetration Room — 1, 2, 3 Aux Bldg. 585' ele. Room 304 corridor outside #3 Mechanical Penetration Room — 1, 2, 3 Aux Bldg. 603' ele. Room 427 - #2 Electrical Penetration Room — 1, 2, 3
6 Control Room Evacuation	None	**HS6** Inability to control a key safety function from outside the Control Room (1 2 3 4 5 6) **HS6.1** An event has resulted in plant control being transferred from the Control Room to the Auxiliary Shutdown Panel. **AND** Control of **ANY** of the following key safety functions is not reestablished within 15 min. (Note 1): • Reactivity (modes 1, 2 and 3 **only**) • Core cooling (RCS inventory) • RCS heat removal (ability to maintain heat sink)	**HA6** Control Room evacuation resulting in transfer of plant control to alternate locations (1 2 3 4 5 6 DEF) **HA6.1** An event has resulted in plant control being transferred from the Control Room to the Auxiliary Shutdown Panel.	
7 Emergency Coordinator Judgment	**HG7** Other conditions exist which in the judgment of the Emergency Coordinator warrant declaration of a General Emergency (1 2 3 4 5 6 DEF) **HG7.1** Other conditions exist which in the judgment of the Emergency Coordinator indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.	**HS7** Other conditions exist which in the judgment of the Emergency Coordinator warrant declaration of a Site Area Emergency (1 2 3 4 5 6 DEF) **HS7.1** Other conditions exist which in the judgment of the Emergency Coordinator indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts, (1) toward site personnel or equipment that could lead to the likely failure of or, (2) that prevent effective access to equipment needed for the protection of the public. **ANY** releases are **not** expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the SITE BOUNDARY.	**HA7** Other conditions exist which in the judgment of the Emergency Coordinator warrant declaration of an Alert (1 2 3 4 5 6 DEF) **HA7.1** Other conditions exist which, in the judgment of the Emergency Coordinator, indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. **ANY** releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.	**HU7** Other conditions exist which in the judgment of the Emergency Coordinator warrant declaration of a UE (1 2 3 4 5 6 DEF) **HU7.1** Other conditions exist which in the judgment of the Emergency Coordinator indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. **No** releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of SAFETY SYSTEMS occurs.

Notes

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Note 3: If the effluent flow past an effluent monitor is known to have stopped, indicating that the release path is isolated, the effluent monitor reading is no longer VALID for classification purposes.

Note 4: The pre-calculated effluent monitor values presented in EALs RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Note 5: If the equipment in the listed room or area was already inoperable or out-of-service before the event occurred, then no emergency classification is warranted.

Note 6: If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, declaration of a General Emergency is not required.

Note 7: This EAL does not apply to routine traffic impediments such as fog, snow, ice, or vehicle breakdowns or accidents.

Note 8: A manual trip action is ANY operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and **does not** include manually driving in control rods or implementation of boron injection strategies.

Note 9: During a main steam line break in containment or LOCA with temperature >170F, there is a potential to induce transient errors into the output of RE4596A and B during the peak rate of temperature change. Consult alternate indications. If the main steam line break is accompanied by core damage this error is insignificant.

Note 10: In the absence of reliable RCS temperature indication caused by the loss of decay heat removal capability, classification should be based on the RCS pressure increase criteria when the RCS is intact in Mode 5 or based on time to boil data when in Mode 6 or the RCS is not intact in Mode 5.

R — Abnormal Rad Release / Rad Effluent

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
1 Rad Effluent	**RG1** Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE (1 2 3 4 5 6 DEF) **RG1.1** Station Vent Channel 1 Noble Gas (RE 4598 AB/BB) > 8.4E +00 µCi/cc for 15 minutes or longer. (Notes 1, 2, 3, 4) **RG1.2** Dose assessment using actual meteorology indicates doses > 1000 mrem TEDE or 5000 mrem thyroid CDE at or beyond the SITE BOUNDARY. (Note 4) **RG1.3** Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY: • Closed window dose rates > 1000 mR/hr expected to continue for ≥ 60 min. • Analyses of field survey samples indicate thyroid CDE > 5000 mrem for 60 min. of inhalation. (Notes 1, 2)	**RS1** Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE (1 2 3 4 5 6 DEF) **RS1.1** Station Vent Channel 1 Noble Gas (RE 4598 AB/BB) > 8.4E -01 µCi/cc for 15 minutes or longer. (Notes 1, 2, 3, 4) **RS1.2** Dose assessment using actual meteorology indicates doses > 100 mrem TEDE or 500 mrem thyroid CDE at or beyond the SITE BOUNDARY. (Note 4) **RS1.3** Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY: • Closed window dose rates > 100 mR/hr expected to continue for ≥ 60 min. • Analyses of field survey samples indicate thyroid CDE > 500 mrem for 60 min. of inhalation. (Notes 1, 2)	**RA1** Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE (1 2 3 4 5 6 DEF) **RA1.1** Station Vent Channel 1 Noble Gas (RE 4598 AB/BB) > 8.4E -02 µCi/cc for 15 minutes or longer. (Notes 1, 2, 3, 4) **RA1.2** Dose assessment using actual meteorology indicates doses > 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the SITE BOUNDARY. (Note 4) **RA1.3** Analysis of a liquid effluent sample indicates a concentration or release rate that would result in doses > 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the SITE BOUNDARY for 60 min. of exposure. (Notes 1, 2) **RA1.4** Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY: • Closed window dose rates > 10 mR/hr expected to continue for ≥ 60 min. • Analyses of field survey samples indicate thyroid CDE > 50 mrem for 60 min. of inhalation. (Notes 1, 2)	**RU1** Release of gaseous or liquid radioactivity > 2 times the ODCM limits for 60 minutes or longer (1 2 3 4 5 6 DEF) **RU1.1** Station Vent Channel 1 Noble Gas (RE 4598AA/BA) > 5.72E-03 µCi/cc for 60 minutes or longer. **OR** **ANY** of the following effluent monitors > 2 times the high alarm setpoint established by a current radioactivity discharge permit for 60 minutes or longer: • Waste Gas System Outlet (RE 1822A or B). • Clean Waste System Outlet (RE 1770A or B). • Miscellaneous Waste System Outlet (RE 1878A or B). • Discharge permit specified monitor. (Notes 1, 2, 3) **RU1.2** Sample analyses for a gaseous or liquid release indicates concentration or release rate > 2 x ODCM limits for ≥ 60 min. (Notes 1, 2)
2 Irradiated Fuel Event	**RG2** Spent fuel pool level cannot be restored to at least the top of the fuel racks for 60 minutes or longer (1 2 3 4 5 6 DEF) **RG2.1** Spent fuel pool level **cannot** be restored to at least 1 ft. for ≥ 60 min. (Note 1)	**RS2** Spent fuel pool level at the top of the fuel racks (1 2 3 4 5 6 DEF) **RS2.1** Lowering of spent fuel pool level to 1 ft.	**RA2** Significant lowering of water level above, or damage to, irradiated fuel (1 2 3 4 5 6 DEF) **RA2.1** Uncovery of irradiated fuel in the REFUELING PATHWAY **RA2.2** Damage to irradiated fuel resulting in a release of radioactivity as indicated by a high radiation alarm on **ANY** of the following radiation monitor indications: • RE 8426 SFP Area • RE 8427 SFP Area • RE 8417 Fuel Handling Area • RE 8418 Fuel Handling Area • RE 8425 Equipment Hatch Area • RE 8446/8447 Fuel Handling Exhaust • RE 4598AA/BA Station Vent **RA2.3** Lowering of spent fuel pool level to 10 ft.	**RU2** UNPLANNED loss of water level above irradiated fuel (1 2 3 4 5 6 DEF) **RU2.1** UNPLANNED water level drop in the REFUELING PATHWAY as indicated by **ANY** of the following: • SFP level alarm or indication (LI 1600) • SFP level indication (LI 4801 A/B) • Refueling Canal low water level alarm or indication (LI 1627) • Visual observation **AND** UNPLANNED rise in corresponding area radiation levels as indicated by **ANY** of the following radiation monitors: • RE 8426 SFP Area • RE 8427 SFP Area • RE 8417 Fuel Handling Area • RE 8418 Fuel Handling Area • RE 8425 Equipment Hatch Area
3 Area Radiation Levels	None	None **Table R-2 Safe Shutdown Rooms/Areas** Area — Mode Aux Bldg. 565' ele. Room 236 #2 Mechanical Penetration Room — 1, 2, 3 Aux Bldg. 585' ele. Room 304 corridor outside #3 Mechanical Penetration Room — 1, 2, 3 Aux Bldg. 603' ele. Room 427 - #2 Electrical Penetration Room — 1, 2, 3	**RA3** Radiation levels that IMPEDE access to equipment necessary for normal plant operations, cooldown or shutdown **RA3.1** (1 2 3 4 5 6 DEF) Dose rate > 15 mR/hr in **EITHER** of the following areas: • Control Room (RE 8430 or 8431) • Central Alarm Station (RE 8435 or 8436) **RA3.2** (1 2 3) An UNPLANNED event results in radiation levels that prohibit or IMPEDE access to **ANY** Table R-2 Safe Shutdown Rooms or Areas. (Note 5)	None

E — DFSF

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
1 Confinement Boundary	None	None	None	**EU1** Damage to a loaded cask CONFINEMENT BOUNDARY (1 2 3 4 5 6 DEF) **EU1.1** Damage to a loaded canister CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading on the surface of a loaded spent fuel HSM cask > **EITHER** of the following: • 100 mrem/hr (neutron + gamma) on the HSM cask wall or roof • 100 mrem/hr (neutron + gamma) on the center of the HSM cask door

C — Cold SD/ Refueling System Malfunct.

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
1 RCS Level	**CG1** Loss of RCS inventory affecting fuel clad integrity with containment challenged (5 6) **CG1.1** RCS level **cannot** be monitored for ≥ 30 min. (Note 1) **AND** Core uncovery is indicated by **ANY** of the following: • UNPLANNED increase in Containment Sumps, Auxiliary Building Sumps, BWST or RCDT levels of sufficient magnitude to indicate core uncovery • Containment Radiation Monitor (RE 4596A or B) reading > 16 R/hr • Refueling Bridge Portable Area Radiation Monitor reading > 30 R/hr (when installed) • Erratic Source Range Monitor indication **AND** **ANY** Containment Challenge indication, Table C-1 **Table C-1 Containment Challenge Indications** • CONTAINMENT CLOSURE **not** established (Note 6) • Containment Hydrogen concentration > 4% • UNPLANNED rise in Containment pressure	**CS1** Loss of RCS inventory affecting core decay heat removal capability (5 6) **CS1.1** RCS level **cannot** be monitored for ≥ 30 min. (Note 1) **AND** Core uncovery is indicated by **ANY** of the following: • UNPLANNED increase in Containment Sumps, Auxiliary Building Sumps, BWST or RCDT levels of sufficient magnitude to indicate core uncovery • Containment Radiation Monitor (RE 4596A or B) reading > 16 R/hr • Refueling Bridge Portable Area Radiation Monitor reading > 30 R/hr (when installed) • Erratic Source Range Monitor indication	**CA1** Loss of RCS inventory (5 6) **CA1.1** Loss of RCS inventory as indicated by RCS level ≤ 0.4 ft. (LI 10596) **CA1.2** RCS level cannot be monitored for ≥ 15 min. (Note 1) **AND EITHER** • UNPLANNED increase in Containment Sumps, Auxiliary Building Sumps, BWST or RCDT due to a loss of RCS inventory • Visual observation of UNISOLABLE RCS leakage	**CU1** UNPLANNED loss of RCS inventory for 15 minutes or longer (5 6) **CU1.1** UNPLANNED loss of reactor coolant results in RCS level less than a required lower limit for ≥ 15 min. (Note 1) **CU1.2** RCS level **cannot** be monitored **AND EITHER:** • UNPLANNED increase in Containment Sumps, Auxiliary Building Sumps, BWST or RCDT due to a loss of RCS inventory • Visual observation of UNISOLABLE RCS leakage
2 Loss of Essential AC Power	None	**Table C-2 Offsite/Onsite AC Power Sources** **Offsite:** • X01 • X02 • X11 (back-fed via Main Transformer if already aligned) **Onsite:** • EDG1 • EDG2 • SBODG	**CA2** Loss of **ALL** offsite and **ALL** onsite AC power to essential buses for 15 minutes or longer (5 6 DEF) **CA2.1** Loss of **ALL** offsite and **ALL** onsite AC power capability to essential 4160V buses C1 and D1 for ≥ 15 min. (Note 1)	**CU2** Loss of **ALL** but one AC power source to essential buses for 15 minutes or longer (5 6 DEF) **CU2.1** AC power capability, Table C-2, to essential 4160V buses C1 and D1 reduced to a single power source for ≥ 15 min. (Note 1) **AND** **ANY** additional single power source failure will result in loss of **ALL** AC power to SAFETY SYSTEMS.
3 RCS Temp.	None	**Table C-3 RCS Heat-up Duration Thresholds** * If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is **not** applicable RCS Status — Containment Closure Status — Heat-up Duration Intact (but **not** reduced inventory) — N/A — 60 min. * **Not** intact **OR** At reduced inventory — established — 20 min. * **Not** intact **OR** At reduced inventory — **not** established — 0 min.	**CA3** Inability to maintain plant in cold shutdown (5 6) **CA3.1** UNPLANNED increase in RCS temperature to > 200°F for > Table C-3 duration. (Note 1, 10) **OR** UNPLANNED RCS pressure increase > 10 psig due to a loss of RCS cooling (This EAL does not apply during water-solid plant conditions).	**CU3** UNPLANNED increase in RCS temperature (5 6) **CU3.1** UNPLANNED increase in RCS temperature to > 200°F due to loss of decay heat removal capability. **CU3.2** Loss of **ALL** RCS temperature and RCS level indication for ≥ 15 min. (Note 1)
4 Loss of Essential DC Power	None	None		**CU4** Loss of essential DC power for 15 minutes or longer (5 6) **CU4.1** < 105 VDC voltage indications on Technical Specification **required** essential 125 VDC distribution panels for ≥ 15 min. (Note 1)
5 Loss of Comm.	None	None	**Table C-4 Communication Methods** System — Onsite — ORO — NRC Public Address (Gaitronics) — X Onsite Radios — X Plant Telephones — X — X — X Commercial Telephones — X — X — X 4-Way Ringdown Circuit — X NRC Emergency Telephone System (ETS) — X	**CU5** Loss of **ALL** onsite or offsite communications capabilities (5 6 DEF) **CU5.1** Loss of **ALL** Table C-4 onsite communication methods. **CU5.2** Loss of **ALL** Table C-4 ORO communication methods. **CU5.3** Loss of **ALL** Table C-4 NRC communication methods.
6 Hazardous Event Affecting Safety Systems	None	**Table C-5 Hazardous Events** • Seismic event (earthquake) • Internal or external FLOODING event • High winds or tornado strike • FIRE • EXPLOSION • Other events with similar hazard characteristics as determined by the Emergency Coordinator	**CA6** Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode (5 6) **CA6.1** The occurrence of **ANY** Table C-5 Hazardous Event **AND EITHER**: • Event has caused indications of degraded performance in at least one train of a SAFETY SYSTEM required for the current operating mode • The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure required for the current operating mode	None

DEFINITIONS

Selected terms used in Initiating Condition and Emergency Action Level statements are set in all capital letters (e.g., ALL CAPS). These words are defined terms that have specific meanings as used in this document. The definitions of these terms are provided below and can also be found in DBRM-EMER-1500A, Davis-Besse Emergency Action Level Basis Document.

ALERT - Events are in process, or have occurred, which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be small fractions of the EPA Protective Action Guideline exposure levels.

CONFINEMENT BOUNDARY - The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage. As related to the DBNPS Dry Fuel Storage Facility, CONFINEMENT BOUNDARY is defined as the Dry Shielded Canister (DSC)

CONTAINMENT CLOSURE - The procedurally defined actions taken to secure containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions.

EMERGENCY ACTION LEVEL (EAL) - A pre-determined, site-specific, observable threshold for an Initiating Condition that, when met or exceeded, places the plant in a given EMERGENCY CLASSIFICATION LEVEL.

EMERGENCY CLASSIFICATION LEVEL (ECL) – One of a set of names or titles established by the NRC for grouping off normal events or conditions according to (1) potential or actual effects or consequences, and (2) resulting ONSITE and OFFSITE response actions. The emergency classification levels, in ascending order of severity, are: UNUSUAL EVENT, ALERT, SITE AREA EMERGENCY, GENERAL EMERGENCY.

EPA PAGs – Environmental Protection Agency Protective Action Guidelines. The EPA PAGs are expressed in terms of dose commitment: 1 Rem TEDE or 5 Rem CDE Thyroid. Actual or projected offsite exposures in excess of the EPA PAGs requires DBNPS to recommend protective actions for the general public to offsite planning agencies.

EXPLOSION - A rapid, violent and catastrophic failure of a piece of equipment due to combustion, chemical reaction or overpressurization. A release of steam (from high energy lines or components) or an electrical component failure (caused by short circuits, grounding, arcing, etc.) should not automatically be considered an EXPLOSION. Such events require a post-event inspection to determine if the attributes of an EXPLOSION are present.

FAULTED - The term applied to a steam generator that has a steam leak on the secondary side of sufficient size to cause an uncontrolled drop in steam generator pressure or the steam generator to become completely depressurized.

FIRE - Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute FIRES. Observation of flame is preferred but is NOT required if large quantities of smoke and heat are observed.

FISSION PRODUCT BARRIER THRESHOLD - A pre-determined, site-specific, observable threshold indicating the loss or potential loss of a fission product barrier.

FLOODING – A condition where water is entering a room or area faster then installed equipment is capable of removal, resulting in a rise of water level within the room or area.

GENERAL EMERGENCY - Events are in process or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

HOSTAGE - A person(s) held as leverage against the station to ensure that demands will be met by the station.

HOSTILE ACTION - An act toward DBNPS or its personnel that includes the use of violent force to destroy equipment, take HOSTAGES, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, PROJECTILES, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on DBNPS. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the OWNER CONTROLLED AREA).

HOSTILE FORCE - One or more individuals, who are engaged in a determined assault, overtly or by stealth and deception, equipped with suitable weapons capable of killing, maiming, or causing destruction.

IMMINENT - The trajectory of events or conditions is such that an EAL will be met within a relatively short period of time regardless of mitigation or corrective actions.

IMPEDE(D) – Personnel access to a room or area is hindered to an extent that extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., requiring the use of protective equipment, such as SCBAs, that is not routinely employed).

INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) - A complex that is designed and constructed for the interim storage of spent nuclear fuel and other radioactive materials associated with spent fuel storage.

INITIATING CONDITION - An event or condition that aligns with the definition of one of the four emergency classification levels by virtue of the potential or actual effects or consequences.

OWNER CONTROLLED AREA - The property associated with the station and owned by the company. Access is normally limited to persons entering for official business.

PROJECTILE - An object directed toward a Nuclear Power Plant that could cause concern for its continued operability, reliability, or personnel safety.

PROTECTED AREA – An area that normally encompasses all controlled areas within the security protected area fence.

RCS Intact – The RCS should be considered INTACT when the RCS pressure boundary is in its normal condition for the cold shutdown mode of operation (e.g., no freeze seals or nozzle gams).

REFUELING PATHWAY - The reactor refueling cavity, spent fuel pool and fuel transfer canal comprise the refueling pathway.

RESTORE – Take the appropriate action required to return the value of an identified parameter to the applicable limits.

RUPTURED - The condition of a steam generator in which primary-to-secondary leakage is of sufficient magnitude to require a safety injection.

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10 CFR 50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition;

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

SECURITY CONDITION - **ANY** Security Event as listed in the approved security contingency plan that constitutes a threat/compromise to site security, threat/risk to site personnel, or a potential degradation to the level of safety of the plant. A SECURITY CONDITION does not involve a HOSTILE ACTION.

SITE AREA EMERGENCY - Events are in process or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts; (1) toward site personnel or equipment that could lead to the likely failure of or; (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guidelines exposure levels beyond the site boundary.

SITE BOUNDARY – Area as depicted in UFSAR Figure 1.2-12 Site Plan. The SITE BOUNDARY is defined at a minimum exclusion distance of 0.75 miles. This Is the nearest distance from potential release points at which protective actions would be required for members of the public.

UNISOLABLE - An open or breached system line that cannot be isolated, remotely or locally.

UNPLANNED - A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

UNUSUAL EVENT - Events are in progress or have occurred which indicate a potential degradation in the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of SAFETY SYSTEMS occurs.

VALID - An indication, report, or condition, is considered to be VALID when it is verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) by direct observation by plant personnel, such that doubt related to the indicator's operability, the condition's existence, or the report's accuracy is removed. Implicit in this definition is the need for timely assessment.

VISIBLE DAMAGE - Damage to a component or structure that is readily observable without measurements, testing, or analysis. The visual impact of the damage is sufficient to cause concern regarding the operability or reliability of the affected component or structure.

Modes: 1 Power Operation | 2 Startup | 3 Hot Standby | 4 Hot Shutdown | 5 Cold Shutdown | 6 Refueling | DEF Defueled

Davis-Besse Nuclear Power Station

DBRM-EMER-1500B, Rev. 3
Davis-Besse EAL Wallboards

COLD CONDITIONS (RCS ≤ 200°F)

1.0 USE AND APPLICATION

1.3 Completion Times

PURPOSE	The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.
BACKGROUND	Limiting Conditions for Operation (LCOs) specify minimum requirements for ensuring safe operation of the unit. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s).
DESCRIPTION	The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the discovery of a situation (e.g., inoperable equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the unit is in a MODE or specified condition stated in the Applicability of the LCO. Unless otherwise specified, the Completion Time begins when a senior licensed operator on the operating shift crew with responsibility for plant operations makes the determination that an LCO is not met and an ACTIONS Condition is entered. The “otherwise specified” exceptions are varied, such as a Required Action Note or Surveillance Requirement Note that provides an alternative time to perform specific tasks, such as testing, without starting the Completion Time. While utilizing the Note, should a Condition be applicable for any reason not addressed by the Note, the Completion Time begins. Should the time allowance in the Note be exceeded, the Completion Time begins at that point. The exceptions may also be incorporated into the Completion Time. For example, LCO 3.8.1, “AC Sources – Operating,” Required Action B.2, requires declaring required feature(s) supported by an inoperable diesel generator, inoperable when the redundant required feature(s) are inoperable. The Completion Time states, “4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s).” In this case the Completion Time does not begin until the conditions in the Completion Time are satisfied. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the unit is not within the LCO Applicability. If situations are discovered that require entry into more than one Condition at a time within a single LCO (multiple Conditions), the Required Actions for each Condition must be performed within the associated Completion Time. When in multiple Conditions, separate

1.3 Completion Times

DESCRIPTION (continued)

Completion Times are tracked for each Condition starting from the discovery of the situation that required entry into the Condition, unless otherwise specified.

Once a Condition has been entered, subsequent trains, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition, unless specifically stated. The Required Actions of the Condition continue to apply to each additional failure, with Completion Times based on initial entry into the Condition, unless otherwise specified.

However, when a subsequent train, subsystem, component, or variable, expressed in the Condition, is discovered to be inoperable or not within limits, the Completion Time(s) may be extended. To apply this Completion Time extension, two criteria must first be met. The subsequent inoperability:

a. Must exist concurrent with the first inoperability; and

b. Must remain inoperable or not within limits after the first inoperability is resolved.

The total Completion Time allowed for completing a Required Action to address the subsequent inoperability shall be limited to the more restrictive of either:

a. The stated Completion Time, as measured from the initial entry into the Condition, plus an additional 24 hours; or

b. The stated Completion Time as measured from discovery of the subsequent inoperability.

The above Completion Time extensions do not apply to those Specifications that have exceptions that allow completely separate re-entry into the Condition (for each train, subsystem, component, or variable expressed in the Condition) and separate tracking of Completion Times based on this re-entry. These exceptions are stated in individual Specifications.

The above Completion Time extension does not apply to a Completion Time with a modified "time zero." This modified "time zero" may be expressed as a repetitive time (i.e., "once per 8 hours," where the Completion Time is referenced from a previous completion of the Required Action versus the time of Condition entry) or as a time modified by the phrase "from discovery . . ."

1.3 Completion Times

EXAMPLES

The following examples illustrate the use of Completion Times with different types of Conditions and changing Conditions.

EXAMPLE 1.3-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

Condition B has two Required Actions. Each Required Action has its own separate Completion Time. Each Completion Time is referenced to the time that Condition B is entered.

The Required Actions of Condition B are to be in MODE 3 within 6 hours AND in MODE 5 within 36 hours. A total of 6 hours is allowed for reaching MODE 3 and a total of 36 hours (not 42 hours) is allowed for reaching MODE 5 from the time that Condition B was entered. If MODE 3 is reached within 3 hours, the time allowed for reaching MODE 5 is the next 33 hours because the total time allowed for reaching MODE 5 is 36 hours.

If Condition B is entered while in MODE 3, the time allowed for reaching MODE 5 is the next 36 hours.

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pump inoperable.	A.1 Restore pump to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

When a pump is declared inoperable, Condition A is entered. If the pump is not restored to OPERABLE status within 7 days, Condition B is also entered and the Completion Time clocks for Required Actions B.1 and B.2 start. If the inoperable pump is restored to OPERABLE status after Condition B is entered, Conditions A and B are exited, and therefore, the Required Actions of Condition B may be terminated.

When a second pump is declared inoperable while the first pump is still inoperable, Condition A is not re-entered for the second pump. LCO 3.0.3 is entered, since the ACTIONS do not include a Condition for more than one inoperable pump. The Completion Time clock for Condition A does not stop after LCO 3.0.3 is entered, but continues to be tracked from the time Condition A was initially entered.

While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition B. The Completion Time for Condition B is tracked from the time the Condition A Completion Time expired.

On restoring one of the pumps to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first pump was declared inoperable. This Completion Time may be extended if the pump restored to OPERABLE status was the first inoperable pump. A 24 hour extension to the stated 7 days is allowed, provided this does not result in the second pump being inoperable for > 7 days.

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Function X train inoperable.	A.1 Restore Function X train to OPERABLE status.	7 days
B. One Function Y train inoperable.	B.1 Restore Function Y train to OPERABLE status.	72 hours
C. One Function X train inoperable. AND One Function Y train inoperable.	C.1 Restore Function X train to OPERABLE status. OR C.2 Restore Function Y train to OPERABLE status.	72 hours 72 hours

1.3 Completion Times

EXAMPLES (continued)

When one Function X train and one Function Y train are inoperable, Condition A and Condition B are concurrently applicable. The Completion Times for Condition A and Condition B are tracked separately for each train starting from the time each train was declared inoperable and the Condition was entered. A separate Completion Time is established for Condition C and tracked from the time the second train was declared inoperable (i.e., the time the situation described in Condition C was discovered).

If Required Action C.2 is completed within the specified Completion Time, Conditions B and C are exited. If the Completion Time for Required Action A.1 has not expired, operation may continue in accordance with Condition A. The remaining Completion Time in Condition A is measured from the time the affected train was declared inoperable (i.e., initial entry into Condition A).

It is possible to alternate between Conditions A, B, and C in such a manner that operation could continue indefinitely without ever restoring systems to meet the LCO. However, doing so would be inconsistent with the basis of the Completion Times. Therefore, there shall be administrative controls to limit the maximum time allowed for any combination of Conditions that result in a single contiguous occurrence of failing to meet the LCO. These administrative controls shall ensure that the Completion Times for those Conditions are not inappropriately extended.

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve(s) to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

A single Completion Time is used for any number of valves inoperable at the same time. The Completion Time associated with Condition A is based on the initial entry into Condition A and is not tracked on a per valve basis. Declaring subsequent valves inoperable, while Condition A is still in effect, does not trigger the tracking of separate Completion Times.

Once one of the valves has been restored to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first valve was declared inoperable. The Completion Time may be extended if the valve restored to OPERABLE status was the first inoperable valve. The Condition A Completion Time may be extended for up to 4 hours provided this does not result in any subsequent valve being inoperable for > 4 hours.

If the Completion Time of 4 hours (plus the extension) expires while one or more valves are still inoperable, Condition B is entered.

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-5

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each inoperable valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

The Note above the ACTIONS Table is a method of modifying how the Completion Time is tracked. If this method of modifying how the Completion Time is tracked was applicable only to a specific Condition, the Note would appear in that Condition rather than at the top of the ACTIONS Table.

The Note allows Condition A to be entered separately for each inoperable valve, and Completion Times tracked on a per valve basis. When a valve is declared inoperable, Condition A is entered and its Completion Time starts. If subsequent valves are declared inoperable, Condition A is entered for each valve and separate Completion Times start and are tracked for each valve.

If the Completion Time associated with a valve in Condition A expires, Condition B is entered for that valve. If the Completion Times associated with subsequent valves in Condition A expire, Condition B is entered separately for each valve and separate Completion Times start and are tracked for each valve. If a valve that caused entry into Condition B is restored to OPERABLE status, Condition B is exited for that valve.

1.3 Completion Times

EXAMPLES (continued)

Since the Note in this example allows multiple Condition entry and tracking of separate Completion Times, Completion Time extensions do not apply.

EXAMPLE 1.3-6

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One channel inoperable.	A.1 Perform SR 3.x.x.x. OR A.2 Reduce THERMAL POWER to ≤ 50% RTP.	Once per 8 hours 8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

Entry into Condition A offers a choice between Required Action A.1 or A.2. Required Action A.1 has a "once per" Completion Time, which qualifies for the 25% extension, per SR 3.0.2, to each performance after the initial performance. The initial 8 hour interval of Required Action A.1 begins when Condition A is entered and the initial performance of Required Action A.1 must be complete within the first 8 hour interval. If Required Action A.1 is followed and the Required Action is not met within the Completion Time (plus the extension allowed by SR 3.0.2), Condition B is entered. If Required Action A.2 is followed and the Completion Time of 8 hours is not met, Condition B is entered.

If after entry into Condition B, Required Action A.1 or A.2 is met, Condition B is exited and operation may then continue in Condition A.

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-7

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One subsystem inoperable.	A.1 Verify affected subsystem isolated.	1 hour AND Once per 8 hours thereafter
	AND	
	A.2 Restore subsystem to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 5.	36 hours

Required Action A.1 has two Completion Times. The 1 hour Completion Time begins at the time the Condition is entered and each "Once per 8 hours thereafter" interval begins upon performance of Required Action A.1.

If after Condition A is entered, Required Action A.1 is not met within either the initial 1 hour or any subsequent 8 hour interval from the previous performance (plus the extension allowed by SR 3.0.2), Condition B is entered. The Completion Time clock for Condition A does not stop after Condition B is entered, but continues from the time Condition A was initially entered. If Required Action A.1 is met after Condition B is entered, Condition B is exited and operation may continue in accordance with Condition A, provided the Completion Time for Required Action A.2 has not expired.

1.3 Completion Times

IMMEDIATE COMPLETION TIME	When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner.