

Student ID Number

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STUDENTS

USE NO. 2 PENCIL ONLY

- **EXAMPLE:** `<A>` `<B>` `<C>`
- **MAKE *DARK* MARKS**
- **ERASE *COMPLETELY* TO CHANGE**
- **MAKE NO STRAY MARKS**

INSTRUCTORS

KEY MARKING INSTRUCTIONS

KEY: Must mark this box on Key sheet.

VERIFY: Prints correct response next to incorrect answers. If Verify is not

marked, a dash will print next to incorrect answers.

HESCORE: HEScore is a previously scored test. Automatically prints correct responses.

NAME JOHN DEY-AN-KUTIM
(Last) (First)

(First)

SUBJECT

DATE 9/29/2024

PERIOD



Accuscan
GUARANTEED

SCORE

KEY 100

RESCORE

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295023 AK3.03
	Importance Rating	3.7

Refueling Accidents**Knowledge of the reasons for the following responses or actions as they apply to Refueling Accidents: Ventilation isolation****Question #1**

An irradiated fuel bundle has been dropped in the Spent Fuel Pool and caused automatic isolation of Control Room Ventilation.

Which one of the following describes the reason for this isolation?

- A. Avoid damage to Control Room Ventilation fans
- B. Minimize noise on neutron monitoring instruments
- C. Ensure continued habitability for Control Room operators
- D. Prevent a secondary release from the Control Building to the outside environment

Answer: C

Explanation: Control Room Ventilation will automatically isolate if Control Building Outside Air Intake Duct radiation monitors exceed the high-high setpoint of 4.5 mR/hr. The reason for this isolation is to maintain Control Room habitability.

- A. Incorrect – The reason for this isolation is to maintain Control Room habitability. Plausible because the isolation does stop flow of contaminated air through the Control Room Ventilation fans, avoiding any possible damage.
- B. Incorrect – The reason for this isolation is to maintain Control Room habitability. Plausible because the isolation is due to high radiation levels and elevated radiation levels can cause issues with equipment operation, particularly sensitive instrumentation.
- D. Incorrect – The reason for this isolation is to maintain Control Room habitability. Plausible because the isolation of the Control Room does isolate an exhaust path from the Control Building, which could intake contamination and then re-release it to the environment. Also plausible because preventing release is the reason for other plant ventilation isolations.

Technical Reference(s): 05-1-02-III-5, TS B 3.3.7.1

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-Z5100 Obj 11

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(9)

Level of Difficulty (LOD): 2

Comments:

TRH 11/17/23 – Revised D based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295019 AK2.23
	Importance Rating	2.7

Partial or Complete Loss of Instrument Air

Knowledge of the relationship between Partial or Complete Loss of Instrument Air and the following systems or components: Service water

Question #2

The plant is operating at 100% power with the following:

- Plant Air Compressors 'A' and 'B' are running.
- Plant Air Compressor 'C' is in standby.

Then, TBCW Return from Air Compressor, P43-F289, fails closed due to a solenoid failure.

Which one of the following describes the impact on the running Plant Air Compressors?

The running Plant Air Compressors...

- A. immediately trip on low cooling water flow.
- B. remain running with backup cooling water automatically supplied from SSW.
- C. do NOT immediately trip on low cooling water flow, but are running without cooling water. Backup cooling water can be manually aligned from SSW.
- D. do NOT immediately trip on low cooling water flow, but are running without cooling water. NO backup cooling water source is available.

Answer: C

Explanation: P43-F289 is the TBCW Return for all 3 PACs. With this valve closed, the normal cooling water supply is lost to the PACs. The PACs do not have a low cooling water flow trip, therefore the running PACs do not immediately trip. The PACs would only trip after some time, once temperatures have risen enough to cause high temperature trips. SSW 'B' can be manually aligned to supply cooling water, but does not automatically supply the PACs.

- A. Incorrect – The running PACs do not immediately trip. Plausible that they would have a low cooling water flow trip, in addition to high temperature trips.
- B. Incorrect – Backup cooling water is not automatically supplied. Plausible because there is a backup cooling water source.
- D. Incorrect – There is a backup cooling water source for PACs. Plausible because many loads do not have backup cooling water sources.

Technical Reference(s): 04-1-02-1H13-P870-5A-E3, 04-S-02-SH13-P854-1A-A4

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-P5100 Obj 15

Question Source: Modified Bank – 5-2017 Audit #51

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Level of Difficulty (LOD): 3

Comments:

TRH 11/17/23 – Revised wording of A, C, and D slightly to focus on low cooling water flow trip based on NRC comment. Revised explanation accordingly.

Question 51

The plant is operating at rated power.

TBCW Return From Air Compressor, P43-F289, fails closed due to a solenoid failure.

Which of the following describes the response of the Plant Air Compressors (PACs)?

- A. Running PAC remains operating for some amount of time while its cabinet temperature rise;
Standby PAC will auto-start when the running PAC trips on high cabinet temperature.
- B. Running PAC immediately trips on low cooling water flow;
the low cooling flow prevents the Standby PAC from auto-starting.
- C. Running PAC remains operating for some amount of time while its cabinet temperature rise;
Standby PAC will not auto-start when the running PAC trips on high temperatures.
- D. Running PAC immediately trips on low cooling water flow;
Standby PAC auto-starts and remains operating.

Answer: A

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295038 EA2.02
	Importance Rating	3.2

High Offsite Radioactivity Release Rate

Ability to determine and/or interpret the following as they apply to High Offsite Radioactivity Release Rate: Total number of curies released or release rate/duration

Question #3

The plant is operating at 100% power with the following:

- A steam leak has developed from the RCIC supply piping into the Auxiliary Building.
- Attempts to isolate the leak have failed.
- Standby Gas Treatment (SBGT) is in service.
- Auxiliary Building pressure is +0.25" H₂O and stable.
- The Shift Manager has declared an Alert emergency condition due to high offsite release rate from SBGT.
- These conditions have existed for 20 minutes.

Entry into the Radioactivity Release Control section of EP-4 is currently...

- A. required due the offsite release exceeding the Alert level.
- B. required due to the presence of positive Auxiliary Building pressure.
- C. NOT required. Entry will become required if these conditions continue to exist for greater than one hour.
- D. NOT required. Entry will become required only if the emergency declaration is upgraded to a Site Area Emergency or General Emergency.

Answer: A

Explanation: Entry into the Radioactivity Release Control section of EP-4 is currently required due to the offsite release exceeding the Alert level

- B. Incorrect – The positive Auxiliary Building pressure does not require entry into EP-4. Plausible because this does mean that there is a ground level, un-filtered, un-monitored radioactivity release in progress. This could lead to EP-4 entry if field survey results exceeded offsite release rate limits.
- C. Incorrect – Entry into the Radioactivity Release Control section of EP-4 is currently required. Plausible because a release rate duration of greater than or equal to one hour is part of multiple emergency action levels. Also plausible because the surveillance requirement for Auxiliary Building D/P includes a one hour requirement.
- D. Incorrect – Entry into the Radioactivity Release Control section of EP-4 is currently required. Plausible because EP-4 entry is based on emergency action levels and further actions in EP-4 are based on a higher level of emergency.

Technical Reference(s): EP-4

References to be provided to applicants during examination: None

Learning Objective:

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Level of Difficulty (LOD): 3

Comments:

TRH 11/17/23 – Deleted “Which one of the following describes the need to enter the Radioactivity Release Control section of EP-4?” based on NRC comment.

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	600000 AK2.07
	Importance Rating	3.3

Plant Fire on Site

Knowledge of the relationship between Plant Fire on Site and the following systems or components: Electrical distribution system

Question #4

The plant is operating at 100% power with the following:

- A fire develops in one of the ESF Switchgear Rooms.
- Fire suppression has actuated.

Which one of the following describes a hazard associated with this fire suppression discharge?

- A. Asphyxiation is a concern due to displacement of O₂ by CO₂
- B. Asphyxiation is a concern due to displacement of O₂ by Halon
- C. Electrical shock is a concern due to presence of water on or near electrical gear
- D. Inability to open breakers is a concern due to chemical reaction with the fire suppression agent

Answer: A

Explanation: ESF Switchgear Rooms are protected by a CO₂ fire suppression system. With CO₂ discharged, the concern is the potential for asphyxiation due to displacement of O₂ by CO₂.

- B. Incorrect – Halon is not used as the fire suppression agent in this area. Plausible because Halon is used in other plant areas and is a gas.
- C. Incorrect – Water is not used as the fire suppression agent for this area. Plausible because the Fire Brigade may use water with a certain spray pattern for firefighting in this area. Also plausible because water could cause shock to nearby personnel when it comes into contact with electrical gear.
- D. Incorrect – CO₂ does not cause a chemical reaction that would prevent proper operation of breakers. Plausible that the fire suppression agent would chemically react in such a way that it would cause corrosion of plant equipment and affect operation.

Technical Reference(s): 04-1-01-P64-3

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-P6400 Obj. 8

Question Source: Bank – JAF 17-2 NRC #19

Question History: NMP2 4/23 NRC #19

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Level of Difficulty (LOD): 2

Comments:

TRH 11/17/23 – Revised original distractor C (and moved to D based on length) based on NRC comment.

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295030 EA1.01
	Importance Rating	4.0

Low Suppression Pool Water Level

Ability to operate and/or monitor the following as they apply to Low Suppression Pool Water Level: ECCSs

Question #5

The plant has experienced an accident with the following:

- Reactor water level is -25" and slowly rising.
- Reactor pressure is 100 psig and slowly lowering.
- Suppression Pool water level is 16.0' and stable.
- LPCS and HPCS are injecting.

Which one of the following describes the status of NPSH to LPCS and HPCS?

	<u>LPCS</u>	<u>HPCS</u>
A.	Has adequate NPSH	Has adequate NPSH
B.	Has adequate NPSH	Adequate NPSH is NOT assured
C.	Adequate NPSH is NOT assured	Has adequate NPSH
D.	Adequate NPSH is NOT assured	Adequate NPSH is NOT assured

Answer: A

Explanation: LPCS and HPCS both have adequate NPSH if Suppression Pool water level is >14.5'. With Suppression Pool water level currently at 16.0', adequate NPSH is assured for both systems.

- B. Incorrect – With Suppression Pool water level currently at 16.0', adequate NPSH is assured for both systems. Plausible that the systems would have different NPSH requirements. Also plausible because Suppression Pool water level is lowering than normal, which does reduce the available NPSH.
- C. Incorrect – With Suppression Pool water level currently at 16.0', adequate NPSH is assured for both systems. Plausible that the systems would have different NPSH requirements. Also plausible because Suppression Pool water level is lowering than normal, which does reduce the available NPSH.
- D. Incorrect – With Suppression Pool water level currently at 16.0', adequate NPSH is assured for both systems. Plausible that the systems would have different NPSH requirements. Also plausible because Suppression Pool water level is lowering than normal, which does reduce the available NPSH.

Technical Reference(s): 04-1-01-E21-1, 04-1-01-E22-1

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-E2201 obj. 15, GLP-OPS-E2100 obj. 7.3

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Level of Difficulty (LOD): 3

Comments:

TRH 11/17/23 – Revised level in stem from 17.2' to 16.0' to match explanations and revised from higher cog to lower cog, based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295031 EK1.01
	Importance Rating	4.7

Reactor Low Water Level

Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to Reactor Low Water Level: Adequate core cooling

Question #6

The plant has experienced a loss of coolant accident with the following:

- Reactor water level is -175" and stable.
- RHR A is injecting 7,500 gpm to the Reactor.
- NO other injection systems are available.

Which one of the following describes the status of core cooling?

Adequate core cooling is...

- A. NOT being provided
- B. being provided by Submergence
- C. being provided by Spray Cooling
- D. being provided by Steam Cooling

Answer: D

Explanation: A Reactor water level of -175" is below the level required for adequate core cooling to be provided by Submergence (-166"), but above the level required for adequate core cooling by Steam Cooling with Injection (-191"). This is also above the level required for Spray Cooling (-216"). However, only LPCS or HPCS, not LPCI, can be credited for Spray Cooling.

- A. Incorrect – Adequate core cooling is being provided by Steam Cooling with Injection. Plausible because this would be correct if Reactor water level was below -191".
- B. Incorrect – Reactor water level is -175", which is below the level required for Submergence. Plausible because Submergence would be provided if Reactor water level were above -166".
- C. Incorrect – Without LPCS or HPCS running, Spray Cooling is not available. Plausible because level is above that required for Spray Cooling (-216"), and LPCI flow is above the 7000 gpm flow requirement.

Technical Reference(s): GLP-OPS-EP01

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-EP01 Obj. 4a

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Level of Difficulty (LOD): 3

Comments:

TRH 12/15/23 – Deleted "With Injection" from the end of choice D based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295027 2.4.3
	Importance Rating	3.7

High Containment Temperature**Ability to identify post-accident instrumentation****Question #7**

Which one of the following Main Control Room indications is required Post Accident Monitoring (PAM) instrumentation, in accordance with Technical Specifications?

- A. SLC tank level
- B. Source Range Monitor (SRM)
- C. Containment air temperature
- D. Narrow range Reactor water level

Answer: C

Explanation: Per Tech Spec 3.3.3.1 Post Accident Monitoring Instrumentation, and 06-OP-1M71-M-0003 Post Accident Monitoring Instrumentation Channel Check, Containment temperature is required Post Accident Monitoring instrumentation.

- A. Incorrect – SLC tank level is not required Post Accident Monitoring instrumentation. Plausible because SLC tank level is required to be used to ensure compliance with Tech Spec 3.1.7.
- B. Incorrect – SRMs are not required Post Accident Monitoring instrumentation. Plausible because SRMs can be used post-accident to verify reactor subcriticality.
- D. Incorrect – Narrow range Reactor water level is not required Post Accident Monitoring instrumentation. Plausible because wide range and fuel zone Reactor water level indications are required PAM instrumentation.

Technical Reference(s): Technical Specification 3.3.3.1

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-M7101 Obj. 24

Question Source: Modified Bank – 5-2017 NRC #74

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Level of Difficulty (LOD): 4

Comments:

Question 74

Which of the following indications is considered to be "Post Accident" instrumentation in the Main Control Room?

- A. Reactor Vessel Water Level – H13-P680 Wide Range
- B. Reactor Vessel Pressure – H13-P680 Wide Range
- C. Standby Liquid Control Tank Level – H13-P601
- D. **Containment Pressure – H13-P870** Narrow Range

Answer: D

Explanation:

Per Tech Spec 3.3.3.1 Post Accident Monitoring Instrumentation and 06-OP-1M71-M-0003 Post Accident Monitoring Instrumentation Channel Check, Containment Pressure is required using both wide and Narrow ranges.

A is wrong – Reactor vessel water level wide range instrumentation is required but only on the P601 panel not the P680.

B is wrong - Reactor vessel pressure wide range instrumentation is required but only on the P601 panel not the P680.

C is wrong – SLC tank level is listed in tech specs but not listed as a post-accident indication.

D is correct – Containment pressure requires both Narrow range and Wide Range.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295003 AK1.01
	Importance Rating	3.8

Partial or Complete Loss of AC Power

Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to Partial or Complete Loss of AC Power: Battery capacity

Question #8

Which one of the following describes the minimum amount of time the ESF Div III Battery, 1C3, is designed to provide DC power to all connected essential loads if there are NO battery chargers in service?

- A. 2 hours
- B. 4 hours
- C. 8 hours
- D. 12 hours

Answer: A

Explanation: The 1C3 batteries have sufficient stored energy to operate connected essential loads continuously for a minimum of 2 hours.

- B. Incorrect – 2 hours is correct. Plausible because 4 hours is the time for the 1A3 and 1B3 batteries.
- C. Incorrect – 2 hours is correct. Plausible because 8 hours is the time for complete restoration of the 1C3 batteries by their chargers.
- D. Incorrect – 2 hours is correct. Plausible because 12 hours is the time for complete restoration of the 1A3 and 1B3 batteries by their chargers.

Technical Reference(s): UFSAR 8.3.2.1.7.5

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-L1100 Obj. 2

Question Source: Modified Bank - 12-2017 Audit #3 / 2013 NRC #3

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Level of Difficulty (LOD): 3

Comments:

TRH 11/17/23 – Changed “required emergency” to “connected essential” in question based on NRC comment.

Question 3

Which of the following describe the minimum amount of time by design the 1B3 batteries can provide DC power to all required emergency loads if there are no battery chargers in service?

- A. 2 hours
- B. 4 hours
- C. 8 hours
- D. 12 hours

Answer: B

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295024 EK1.02
	Importance Rating	4.0

High Drywell Pressure

Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to High Drywell Pressure: Containment building integrity (Mark III)

Question #9

Which one of the following describes a potential consequence of exceeding an internal drywell pressure of 30 psig during a LOCA?

- A. A loss of the pressure suppression function of Primary Containment
- B. Lowered efficiency of the charcoal filter train which may be used to vent the containment via 6" vent lines
- C. Additional leaks due to exceeding the pressure rating of various primary system components in the drywell
- D. Re-pressurization of the RPV from decay heat due to the inability of SRVs to remain open

Answer: A

Explanation: Exceeding 30 psig drywell internal pressure can result in a loss of drywell integrity, allowing the LOCA blowdown to bypass the horizontal vents and hence the suppression pool (which otherwise provides the pressure suppression function); i.e., drywell discharges would be directly to the Primary Containment air atmosphere. Without the pressure suppression function, Containment Building integrity could be challenged.

- B. Incorrect – The 6" vent valves are no longer available with drywell pressure above 1.23 psig. Additionally, pressure is not considered a factor in plant procedures for the efficiency of the charcoal filter train. Plausible that high pressure would affect filter train performance.
- C. Incorrect – The concern is loss of drywell integrity, not integrity of primary system components inside the drywell. Plausible there are many primary system components in the drywell and they are subjected to this higher than normal pressure.
- D. Incorrect – The SRVs have no limitation on the maximum drywell pressure against which they are able to remain open. Plausible because SRVs are affected by conditions inside the Drywell.

Technical Reference(s): UFSAR 6.2.1.1.5.1, Containment System Steam Bypass

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-M4101 obj. 5

Question Source: Bank - 2013 NRC #11

Question History: 2013 NRC #11

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(9)

Level of Difficulty (LOD): 3

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295016 2.4.34
	Importance Rating	4.2

Control Room Abandonment**Knowledge of RO responsibilities outside the main control room during an emergency**

Question #10

The plant is operating at 100% power with the following:

- A significant fire develops in the Main Control Room.
- The Shift Manager has decided to evacuate the Main Control Room.
- 05-1-02-II-1, Shutdown from the Remote Shutdown Panel, is entered.
- The Reactor Mode switch is placed in SHUTDOWN.
- All control rods fully insert.

Which one of the following additional actions is required in the field to ensure the Reactor remains scrammed, in accordance with 05-1-02-II-1?

- A. Manually vent the scram air header
- B. Pull all HCU fuses to scram pilot solenoids
- C. Place all HCU individual scram test switches to TEST
- D. Open RPS distribution panel breakers to all scram pilot solenoids

Answer: D

Explanation: Given a fire in the Main Control Room, 05-1-02-II-1 step 3.1.4 requires dispatching a Reactor Operator to perform Attachment XXII. This attachment opens RPS breakers to ensure scram pilot solenoids are de-energized.

- A. Incorrect – RPS breakers are opened. Plausible because this is an action in EP-1 for inserting control rods during an ATWS.
- B. Incorrect – RPS breakers are opened. Plausible because this is a similar action that would also remove power from scram pilot solenoids.
- C. Incorrect – RPS breakers are opened. Plausible because this is an action in EP-1 for inserting control rods during an ATWS.

Technical Reference(s): 05-1-02-II-1, 05-S-01-EP-1

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-ONEP Obj 6

Question Source: Modified Bank – NMP1 2017 NRC #21

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Level of Difficulty (LOD): 3

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	212000 2.4.34
	Importance Rating	4.2

RPS

Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects.

Proposed Question: #21

The plant is operating at 100% power with the following:

- A significant fire develops in the Control Room.
- N1-SOP-21.2, Control Room Evacuation, is entered.
- The Reactor Mode Switch is placed in SHUTDOWN.
- All RPS Group Scram Solenoid white lights remain lit.
- NO control rod motion is observed.
- The Control Room is evacuated **before** any other operator actions are taken.

Which one of the following describes the required field actions to be performed to scram the Reactor, in accordance with N1-SOP-21.2?

- A. Isolate and vent the scram air header.
- B. Pull RPS group scram solenoid fuses.
- C. Trip RPS Motor Generator Sets 131 and 141.
- D. Trip RPS Uninterruptible Power Supplies 162 and 172.

NMP1 2017 NRC

Proposed Answer: C

Explanation: N1-SOP-21.2 normally requires scrambling the Reactor and verifying control rod insertion from the Control Room prior to evacuation. In this event, with failure of the Mode Switch to insert control rods, alternate field actions are required to insert control rods. N1-SOP-21.2 requires a Reactor Operator to trip the RPS MG sets (131 and 141) from outside the Control Room.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295025 EA1.01
	Importance Rating	3.1

High Reactor Pressure

**Ability to operate and/or monitor the following as they apply to High Reactor Pressure:
Main and reheat steam**

Question #11

The plant is operating at 100% power with the following:

- Main Steam Line A inboard and outboard MSIVs, 1B21-F022A and 1B21-F028A, spuriously close simultaneously.
- Reactor pressure rises to a high of 1075 psig during the transient.
- Reactor power rises to a high of 109% during the transient.

Which one of the following describes if a Reactor scram occurs, and if so, why?

The Reactor...

- A. does NOT scram
- B. scrams on an MSIV position signal
- C. scrams on a high Reactor power signal
- D. scrams on a high Reactor pressure signal

Answer: D

Explanation: A Reactor scram occurs on a high Reactor pressure signal because Reactor pressure rises above 1064.7 psig.

- A. Incorrect – A Reactor scram occurs on a high Reactor pressure signal because Reactor pressure rises above 1064.7 psig. Plausible because scram setpoints are not exceeded for MSIV position or Reactor power.
- B. Incorrect – With only 2 MSIVs closing in the same Main Steam Line, the Reactor scram logic is not satisfied for the MSIV position scram. Plausible because this would be correct if more MSIVs closed.
- C. Incorrect – Reactor power does not get high enough to exceed the scram setpoint. Plausible because Reactor power is high enough to exceed a rod block setpoint.

Technical Reference(s): ARI P680-5A-A2, P680-5A-A11, P680-5A-B2, Table TR3.3.1.1-1 note (c)

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-C7100 Obj. 9

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Level of Difficulty (LOD): 3

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295004 AA1.03
	Importance Rating	3.5

Partial or Complete Loss of DC Power**Ability to operate and/or monitor the following as they apply to Partial or Complete Loss of DC Power: AC electrical distribution**

Question #12

The plant is operating at 100% power with the following:

- Bus 15AA is de-energized due to a sustained electrical fault.
- Battery bus 11DA is de-energized due to a spurious breaker trip that CANNOT be reset.
- ESF XFMR 12 is available to energize bus 15AA.

Which one of the following describes the operation of circuit breaker 152-1511, FDR FRM ESF XFMR 12?

- A. Still has automatic trip capability
- B. Capable of closing and reopening once at the breaker
- C. Capable of being opened remotely but must be closed manually at the breaker
- D. NOT capable of any remote operation and NOT capable of any local operation

Answer: B

Explanation: This 4160 breaker cannot be controlled remotely without DC power, all operations must be done locally. Any breaker that is already closed when the loss of DC occurs can be opened because the opening springs are compressed by the closing of the breaker, the opening springs are also compressed when the breaker is closed. If the breaker is already open then the breaker can be closed and reopened once.

- A. Incorrect – The trip coil is DC powered. Plausible that the trip capability would be maintained by some other power source.
- C. Incorrect – The breaker will not operate remotely without DC power but will operate manually at the breaker as described above. Plausible that opening capability would be maintained by some other power source to maintain the ability to de-energize loads in an emergency.
- D. Incorrect – The breaker will not operate remotely without DC power but will operate manually at the breaker as described above. Plausible since all remote operating capability is lost.

Technical Reference(s): 05-1-02-I-4, 04-1-01-L11-1, BC06Sr4_Breakers page 5

References to be provided to applicants during examination: None

Learning Objective: BC06Sr4_Breakers Obj 10

Question Source: Bank - 2-2020 NRC #3

Question History: 2-2020 NRC #3

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Level of Difficulty (LOD): 3

Comments:

TRH 11/17/23 – Edited 1st 2 bullets to provide information that bus losses are due to abnormal conditions based on NRC comment. Edited original distractor B (and moved to D) based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295018 AK2.04
	Importance Rating	3.7

Partial or Complete Loss of CCW

Knowledge of the relationship between Partial or Complete Loss of Component Cooling Water and the following systems or components: Reactor recirculation system

Question #13

The plant is operating at 100% power when CCW flow to the Containment isolates and CANNOT be restored.

Which one of the following describes the required control of the Reactor Recirculation system, in accordance with 05-1-02-V-1, Loss of Component Cooling Water ONEP?

- A. Trip both Recirculation pumps
- B. Transfer both Recirculation pumps to Slow Speed operation
- C. Trip one Recirculation pump and maintain the other pump operating
- D. Lower Recirculation flow, but maintain both pumps operating in Fast Speed

Answer: A

Explanation: With CCW flow to the Containment isolated, 05-1-02-V-1 requires scrambling the Reactor and tripping both Reactor Recirculation pumps.

- B. Incorrect – Both Reactor Recirculation pumps must be tripped. Plausible that lowering Recirculation pump speed would be enough to lower the heat load on the pumps to within ambient losses.
- C. Incorrect – Both Reactor Recirculation pumps must be tripped. Plausible that one pump would be maintained operating to ensure forced core circulation.
- D. Incorrect – Both Reactor Recirculation pumps must be tripped. Plausible that lowering Recirculation flow would be enough to lower the heat load on the pumps to within ambient losses. Also plausible because lowering Recirculation flow is an action in 05-1-02-V-1 for partial loss of CCW.

Technical Reference(s): 05-1-02-V-1

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-ONEP Obj. 6

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Level of Difficulty (LOD): 3

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295001 AK3.07
	Importance Rating	3.3

Partial or Complete Loss of Forced Core Flow Circulation

Knowledge of the reasons for the following responses or actions as they apply to Partial or Complete Loss of Forced Core Flow Circulation: Recirculation pump discharge/suction valve manipulation

Question #14

The plant is operating at 60% power with the following:

- Recirc pump A has tripped.
- ONEP 05-1-02-III-3, Reduction in Recirculation System Flow Rate, has been entered.
- An hour has passed.
- All required actions in ONEP 05-1-02-III-3 have been completed.

Which one of the following describes the status of B33-F067A, Recirc pump A discharge valve?

B33-F067A is...

- A. open to keep the loop warm
- B. closed to prevent a cold water addition
- C. closed to ensure accurate core flow indications
- D. open to ensure accurate Reactor water level indications

Answer: A

Explanation: ONEP 05-1-02-III-3 initially closes B33-F067A, but then re-opens it. This keeps the loop warm to maintain differential temperature between the loops within limits.

- B. Incorrect – B33-F067A is open. Plausible because this valve is temporarily closed during execution of this procedure. Also, plausible because cold water additions are undesired and keeping the valve closed would prevent any cooler water in the idle loop from reaching the Reactor.
- C. Incorrect – B33-F067A is open. Plausible because this valve is temporarily closed during execution of this procedure. Also, plausible because closing the valve would stop flow in the idle loop. Recirc loop flow could be confused with jet pump flow (core flow indication).
- D. Incorrect – The reason is to keep the idle loop warm. Plausible that the reason would be related to Reactor water level indications since the recirc loop flow hydraulically interacts with the variable legs connected to the downcomer.

Technical Reference(s): 05-1-02-III-3

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-ONEP Obj. 7

Question Source: Bank - SSES LO26R NRC #46 (2015)

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Level of Difficulty (LOD): 3

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295005 AK3.04
	Importance Rating	3.2

Main Turbine Generator Trip

Knowledge of the reasons for the following responses or actions as they apply to Main Turbine Generator Trip: Main generator trip

Question #15

The plant is operating at 24% power with the Main Generator synchronized to the grid.

Then, the Reactor is manually scrammed.

Which one of the following describes the resulting status of the Main Generator output breakers (J5228 and J5232) ten (10) minutes later?

The Main Generator output breakers are...

- A. open to prevent de-stabilizing the power grid
- B. open to limit reverse powering the Main Generator
- C. closed to ensure proper alignment of the power grid
- D. closed to prevent voltage surges that exceed the output breakers rating

Answer: B

Explanation: When the Reactor is scrammed, the Main Turbine and Main Generator initially remain online because there is no direct signal to trip them. The Main Turbine control valves will close to prevent depressurizing the Reactor. The Main Generator loses motive power. This results in the Main Generator receiving a reverse power trip signal. The Main Generator output breakers open to disconnect the Main Generator from the grid and prevent continued reverse power of the Main Generator.

- A. Incorrect – The reason is to prevent reverse power the Main Generator. Plausible that maintaining the Main Generator connected to the grid could also affect grid conditions.
- C. Incorrect – The Main Generator output breakers open. Plausible because the Reactor scram does not directly trip the Main Turbine or Main Generator. Also plausible because 05-1-02-I-2 requires manually reclosing these breakers after a trip for proper alignment of the grid..
- D. Incorrect – The Main Generator output breakers open. Plausible because the Reactor scram does not directly trip the Main Turbine or Main Generator. Also plausible because there is a caution in 05-1-02-I-2 about having the output breakers open for an extended period with the disconnect closed causing voltage surges exceeding the breakers rating.

Technical Reference(s): 05-1-02-I-2

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-ONEP Obj 7

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(5)

Level of Difficulty (LOD): 3

Comments:

TRH 11/17/23 – Adjusted wording of question and answer choices to raise plausibility of C and D, based on NRC comment.

TRH 12/15/23 – Revised to further raise plausibility of C and D after prep week discussion.

Note – simulator testing showed that the reverse power trip occurred within one minute.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	700000 AA2.06
	Importance Rating	3.2

Generator Voltage and Electric Grid Disturbances

Ability to determine and/or interpret the following as they apply to Generator Voltage and Electric Grid Disturbances: Generator frequency limitations

Question #16

The plant is operating at 100% power with the following:

- Grid instabilities are occurring.
- 05-1-02-I-8, Loss of BOP AC Power / Grid Instability ONEP, is entered.
- Grid frequency is 58 Hz and lowering slowly.

Which one of the following identifies:

- a concern with this frequency
and
- the need for a Reactor scram,

in accordance with 05-1-02-I-8?

	Concern	Reactor Scram
A.	Trip of EPA breakers	Required
B.	Trip of EPA breakers	NOT required
C.	Over-heating of Main Generator	Required
D.	Over-heating of Main Generator	NOT required

Answer: A

Explanation: The concern in 05-1-02-I-8 related to low grid frequency is trip of EPA breakers on underfrequency. With frequency less than 58.5 Hz, the immediate operator action override in 05-1-02-I-8 requires placing the Reactor Mode Switch in SHUTDOWN.

- B. Incorrect – A Reactor scram is required. Plausible because this would be correct if frequency was >58.5 Hz.
- C. Incorrect – The concern in 05-1-02-I-8 related to low grid frequency is trip of EPA breakers on underfrequency. Plausible because this would be a concern for other conditions in the procedure, such as high MVARs from low grid voltage.
- D. Incorrect – A Reactor scram is required. Plausible because this would be correct if frequency was >58.5 Hz. The concern in 05-1-02-I-8 related to low grid frequency is trip of EPA breakers on underfrequency. Plausible because this would be a concern for other conditions in the procedure, such as high MVARs from low grid voltage.

Technical Reference(s): 05-1-02-I-8

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-ONEP Obj 5

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Level of Difficulty (LOD): 3

Comments:

TRH 11/17/23 – Raised grid frequency from 56 to 58 Hz and deleted “manual” before “Reactor scram” based on NRC comments.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295026 EK3.04
	Importance Rating	3.6

Suppression Pool High Water Temperature**Knowledge of the reasons for the following responses or actions as they apply to Suppression Pool High Water Temperature: SLCS injection**

Question #17

The plant has experienced an ATWS with the following:

- Reactor power is 2.5%.
- EP-2A, ATWS RPV Control, is being executed.

Which one of the following describes:

- the requirement for injecting SLC based on Suppression Pool temperature and
- the reason for this requirement,

in accordance with EP-2A?

	Inject SLC...	Reason
A.	before 110°F is exceeded	Provide for shutting down the reactor before exceeding Containment temperature and pressure limits
B.	before 110°F is exceeded	To suppress large power oscillations leading to core damage
C.	when 110°F is exceeded	Provide for shutting down the reactor before exceeding Containment temperature and pressure limits
D.	when 110°F is exceeded	To suppress large power oscillations leading to core damage

Answer: A

Explanation: SLC injection is required before Suppression Pool temperature exceeds 110°F per EP-2A. The 110° F limit is based on Technical Specifications:

TS Bases for LCO B 3.6.2.1 "A limitation on the suppression pool average temperature is required to assure that the primary containment conditions assumed for the safety analyses are met. This limitation subsequently ensures that peak primary containment pressures and temperatures do not exceed maximum allowable values during a postulated DBA or any transient resulting in heatup of the suppression pool.

- B. Incorrect – Large power instabilities are not expected less than 5% power. Plausible because the bases do discuss this as a concern in other conditions. "If reactor power is at or below 5%, thermal-hydraulic instabilities are not of significant concern since the core boiling boundary will be relatively high and the core void content will be relatively low. Some oscillations may still occur, but large-scale instabilities leading to core damage are not expected. Boron injection then need not be injected unless primary containment integrity is jeopardized."
- C. Incorrect – The requirement is before, not after, 110°F. Plausible because the EPs use both "before" and "after" requirements.
- D. Incorrect – The requirement is before, not after, 110°F. Plausible because the EPs use both "before" and "after" requirements. Large power instabilities are not expected less than 5% power. Plausible because the bases do discuss this as a concern in other conditions. See B

Technical Reference(s): EP-2A, TS Bases B 3.6.2.1

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-EP02A Obj. 7

Question Source: Bank – 2-2020 NRC #13

Question History: 2-2020 NRC #13

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Level of Difficulty (LOD): 3

Comments:

TRH 11/17/23 – Revised 1st half of C and D based on NRC comment, and then revised 1st half of A and B to match.

TRH 12/15/23 – Revised "SBLC" to "SLC" for consistency with other questions, based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295037 EK2.12
	Importance Rating	3.7

Scram Condition Present and Reactor Power Above APRM Downscale or Unknown

Knowledge of the relationship between SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown and the following systems or components: RCIS (BWR 6)

Question #18

The plant has experienced an ATWS with the following:

- Reactor power is 15%.
- Reactor water level is -80" and slowly lowering.

Given the following possible actions:

- (1) Restoration of the Auxiliary Building
- (2) Installation of EP-1 Attachment 18, Defeating ARI/RPT Logic Trips
- (3) Installation of EP-1 Attachment 19, Defeating RPS Logic Trips
- (4) Installation of EP-1 Attachment 20, Defeating RC&IS Control Rod Drive Blocks

Which one of the following identifies which of these actions is required to ensure the ability to insert control rods by using the IN TIMER SKIP pushbutton?

- (1) and (4) only
- (2) and (3) only
- (2), (3), and (4) only
- (1), (2), (3), and (4)

Answer: D

Explanation: With Reactor power at 15%, RPC enforces insert blocks and level is intentionally lowered per EP-2A below the RPS and ATWS RPT initiation setpoints. To be able to insert control rods with RCIS, the scram must be reset (Att. 19), air restored to the Auxiliary building, and ATWS/ARI reset to provide air to the scram air header (Att. 18), and all Control Rod Insert blocks cleared (Att 20)

- A. Incorrect – (2) and (3) are also required. Plausible that the only things that would be preventing rod movement with RCIS would be rod blocks and lack of air to the Auxiliary Building. Plausible that (2) and (3) would only be needed for repeated manual scram attempts.
- B. Incorrect – (1) and (4) are also required. Plausible that resetting ARI and the scram would be the only required actions to restore CRD pressure to allow rod movement with RCIS.
- C. Incorrect – (1) is also required. Plausible that the only required actions would be to reset ARI and the scram (to restore CRD pressure) and to override rod blocks.

Technical Reference(s): EP-2A, EP-1

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-EP01 Obj. 5

Question Source: Bank – 2013 Audit #82

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Level of Difficulty (LOD): 3

Comments:

TRH 11/17/23 – Revised to have (1) in more than one answer choice, based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295021 AA2.02
	Importance Rating	4.1

Loss of Shutdown Cooling

Ability to determine and/or interpret the following as they apply to Loss of Shutdown Cooling: RHR/shutdown cooling system flow

Question #19

The plant is in Mode 5 with core shuffle in progress with the following:

- RHR A is operating in Shutdown Cooling (SDC).
- RHR B is in Standby.
- A leak develops.
- Operators isolate the leak by closing the following valves:
 - E12-F008, RHR SHUTDN CLNG OTBD SUCT VLV
 - E12-F009, RHR SHUTDN CLNG INBD SUCT VLV
- RHR A is placed in Standby.

If E12-F008 and E12-F009 remain closed, which one of the following describes the ability to:

- restore SDC flow using RHR A using an alternate suction source
- and
- restore SDC flow using RHR B using the common suction line?

	<u>RHR A using an alternate suction source</u>	<u>RHR B using the common suction line</u>
A.	Can restore SDC flow	Can restore SDC flow
B.	Can restore SDC flow	CANNOT restore SDC flow
C.	CANNOT restore SDC flow	Can restore SDC flow
D.	CANNOT restore SDC flow	CANNOT restore SDC flow

Answer: B

Explanation: 04-1-01-E12-2 section 6.5 can be used to restore SDC flow using RHR A with suction from the Spent Fuel Pool. SDC flow cannot be restored using RHR B using the common suction line, as that would require E12-F008 and E12-F009 to be open.

- A. Incorrect – SDC flow cannot be restored using RHR B using the common suction line. Plausible that RHR B would have a completely separate suction line from RHR A. Also plausible because this would be correct if the leak had been isolated by a valve or valves specific to RHR A.
- C. Incorrect – SDC flow can be restored using RHR A with suction from an alternate source (the Spent Fuel Pool). Plausible that the alternate flow path would also utilize these valves. Also plausible because this would be correct if plant conditions were different (eg. Mode 4). SDC flow cannot be restored using RHR B using the common suction line. Plausible that RHR B would have a completely separate suction line from RHR A. Also plausible because this would be correct if the leak had been isolated by a valve or valves specific to RHR A.
- D. Incorrect – SDC flow can be restored using RHR A with suction from an alternate source (the Spent Fuel Pool). Plausible that the alternate flow path would also utilize these valves. Also plausible because this would be correct if plant conditions were different (eg. Mode 4).

Technical Reference(s): 04-1-01-E12-2

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-E1200 obj. 3.4

Question Source: Modified Bank – 2010 Audit #8

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Level of Difficulty (LOD): 3

Comments:

Question 8

Fuel moves between the reactor and the Spent Fuel Pool are in progress.

Recirc Pump 'A' is operating in SLOW speed.

RHR 'A' is operating in Shutdown Cooling (SDC) when a leak develops on the common suction line.

Operators close E12-F008 and F009 (SDC Outboard and Inboard Suctions) to stop the leak.

Neither RHR 'B' nor ADHR is available for decay heat removal.

What can operators do, if anything, to re-establish decay heat removal without having to suspend the fuel moves?

- A. Raise reactor water level to +82" using the Inadequate Decay Heat Removal ONEP.
- B. Establish a flow path between the reactor and the suppression pool via the SRVs and RHR Pump 'A' using the Inadequate Decay Heat Removal ONEP.
- C. Place RHR 'A' in Shutdown Cooling with suction from the Spent Fuel Pool using the E12-2 Shutdown Cooling SOI.
- D. Nothing can be done without suspending the fuel moves.

Answer: C

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295006 2.1.7
	Importance Rating	4.4

Scram

Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation

Question #20

The plant is operating at 55% power with the following:

- Recirc pump B is secured but available for start.
- A malfunction with Recirc A flow control causes Recirc loop A flow to rapidly lower.
- ONEP 05-1-02-III-3, Reduction in Recirculation System Flow Rate, is entered.
- Periodic LPRM upscale and downscale alarms are being received.
- Reactor power is oscillating between 25% and 40% on APRMs.
- Operation is in the Controlled Entry Region of the power-flow map.

Which one of the following actions is required, in accordance with 05-1-02-III-3?

- A. Insert rods with RC&IS
- B. Manually scram the Reactor
- C. Raise core flow by starting Recirc pump B
- D. Perform a normal plant shutdown per IOI 03-1-01-3

Answer: B

Explanation: The given LPRM alarms and APRM oscillations meet the definition of Thermal Hydraulic Instability (THI). With THI present, a manual Reactor scram is required.

- A. Incorrect – A manual Reactor scram is required. Plausible because without THI present, inserting control rods with RC&IS would be a required action to exit the Controlled Entry Region.
- C. Incorrect – A manual Reactor scram is required. Plausible because the initiating condition was a reduction in core flow and Recirc pump B is stated as being available for start.
- D. Incorrect – A manual Reactor scram is required. Plausible because a shutdown is required, however it must be accomplished by an immediate Reactor scram, not by a normal controlled shutdown.

Technical Reference(s): 05-1-02-III-3

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-ONEP Obj. 5

Question Source: Bank - NMP1 2015 NRC #56

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(5)

Level of Difficulty (LOD): 3

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	500000 EK3.10
	Importance Rating	3.6

High Containment Hydrogen Concentration

Knowledge of the reasons for the following responses or actions as they apply to High Containment Hydrogen Concentration: Operation of the hydrogen igniters (BWR 6)

Question #21

Which one of the following describes the reason why EP-3, Primary Containment Control, directs operation of the hydrogen igniters?

- A. Lower Containment pressure
- B. Prevent an explosion in the Containment
- C. Scrub Containment atmosphere of radionuclides
- D. Avoid periodic spikes in Containment temperature

Answer: B

Explanation: Hydrogen igniters are operated to burn off hydrogen in the Containment atmosphere to prevent the concentration from building to the point where a hydrogen explosion could occur.

- A. Incorrect – The reason is to prevent an explosion. Plausible because other actions in EP-3 are performed to lower Containment pressure. Also plausible that burning off any hydrogen and oxygen gas in the Containment would reduce the overall volume of gasses, thereby lowering overall pressure.
- C. Incorrect – The reason is to prevent an explosion. Plausible because other actions in EP-3 do assist in scrubbing the Containment atmosphere of radionuclides (operation of Containment spray), which is of benefit to reduce the release if venting is required. Also plausible that the burning of hydrogen in the Containment atmosphere would lead to a reduction in radionuclides in the atmosphere.
- D. Incorrect – The reason is to prevent an explosion. Plausible because there is a caution in 04-1-01-E61-1 about hydrogen igniters potentially *causing* periodic spikes in Containment temperature. Also plausible because preventing an explosion does prevent a larger Containment pressure/temperature excursion.

Technical Reference(s): 05-S-01-EP-3M1-3, 04-1-01-E61-1

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-EP3 Obj. 7

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Level of Difficulty (LOD): 3

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295010 AK1.03
	Importance Rating	3.8

High Drywell Pressure

Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to High Drywell Pressure: Drywell temperature increase

Question #22

A loss of coolant accident has occurred with the following:

- Drywell pressure is 2.8 psig.
- Drywell temperature is 211°F.
- A complete loss of offsite power has occurred, including the Port Gibson line.
- Diesel Generators have re-powered the ESF buses.
- NO Operator actions have been performed.

Which one of the following identifies the method of heat removal from the Drywell under the current plant conditions?

- A. Heat removal from the drywell is only by ambient losses without air circulation
- B. Drywell Coolers and Drywell Chilled Water System 'B' are operating on SSW 'B'
- C. 'A' Drywell Cooler fans are circulating air but the coolers are without cooling water flow
- D. 'B' Drywell Cooler fans are circulating air but the coolers are without cooling water flow

Answer: A

Explanation: Drywell pressure caused an LSS LOCA Shed and Sequence. The Drywell Coolers are powered from 15B42 and 16B42, which are shed on a LOCA and not automatically re-energized. Operator action is required to restore power to the Drywell Coolers. 15B42 and 16B42 are shed and re-energized automatically on a loss of power, however the LOCA signal will override the loss of power sequence. Additionally, the Drywell Chilled Water supply and return valves will be automatically isolated due to the Drywell Pressure.

- B. Incorrect – Plausible, procedures does not allow Drywell Chilled Water to be restored by reopening isolation valves without complete system startup when Drywell temperature is > 200°F., SSW 'B' is aligned for LOCA cooling which is not aligned to the DW chilled water system and 'B' Chilled water system is powered from Div 2 16AB which is locked out on a LOCA by LSS.
- C. Incorrect – Plausible, the Drywell Coolers are powered from 15B42 and 16B42, which are shed on a LOCA and not automatically re-energized
- D. Incorrect – Plausible, the Drywell Coolers are powered from 15B42 and 16B42, which are shed on a LOCA and not automatically re-energized

Technical Reference(s): 04-1-01-R21-1 Table 1 page 12

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-M5100.00 Objective: 9.1, 9.2

Question Source: Bank – 2-2020 NRC #22

Question History: 2-2020 NRC #22

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Level of Difficulty (LOD): 3

Comments:

TRH 11/17/23 – Lowered LOD rating based on NRC comment.

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295008 AK2.04
	Importance Rating	3.6

High Reactor Water Level

Knowledge of the relationship between High Reactor Water Level and the following systems or components: PCIS/NSSSS

Question #23

The plant has experienced a scram with the following:

- Reactor water level reached a low of -50" following the scram.
- Reactor water level is now +48" and rising.
- It is desired to place Reactor Water Cleanup (RWCU) reject in service.

Which one of the following describes the status of RWCU suction isolation valves, G33-F001 and G33-F004, for placing RWCU reject in service?

G33-F001 and G33-F004 are...

- A. open because Reactor water level did NOT go low enough to close them.
- B. open because they automatically re-opened when Reactor water level was restored.
- C. closed. Manual action needs to be taken to reset their isolation signal prior to re-opening them.
- D. closed. NO manual action needs to be taken to reset their isolation signal prior to re-opening them.

Answer: C

Explanation: G33-F001 and G33-F004 automatically closed when Reactor water level went below -41.6". When Reactor water level rises back above -41.6", the isolation signal remains sealed-in until manual action is taken to reset it (depressing pushbuttons NSSSS INBD ISOL RESET and NSSSS OTBD ISOL RESET).

- A. Incorrect – G33-F001 and G33-F004 automatically closed when Reactor water level went below -41.6". Plausible because this would be correct if Reactor water did not get quite so low.
- B. Incorrect – G33-F001 and G33-F004 automatically closed when Reactor water level went below -41.6". They remained closed when Reactor water level rose back above this value. Plausible because this would be correct if the isolation signal automatically cleared and the valves had a sealed-in open signal based on switch position.
- D. Incorrect – When Reactor water level rises back above -41.6", the isolation signal remains sealed-in until manual action is taken to reset it (depressing pushbuttons NSSSS INBD ISOL RESET and NSSSS OTBD ISOL RESET). Plausible that the isolation signal would automatically clear when Reactor water level rose back above the isolation setpoint.

Technical Reference(s): 05-1-02-III-5, 05-1-02-I-1 Attachment 4

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-G3336 Obj. 7.6

Question Source: Bank – NMP2 2017 NRC #60

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Level of Difficulty (LOD): 3

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295011 AA2.02
	Importance Rating	3.8

High Containment Temperature

Ability to determine and/or interpret the following as they apply to High Containment Temperature: Containment pressure

Question #24

The plant has experienced a steam leak in the Containment with the following:

- EP-3, Primary Containment Control, is being executed.
- Containment Spray has been initiated due to high Containment temperature.

Which one of the following describes the threshold value that requires securing Containment Spray, in accordance with EP-3, Primary Containment Control?

Containment Spray is required to be secured BEFORE Containment...

- A. pressure reaches the threshold value of 0 psig
- B. pressure reaches the threshold value of 1.23 psig
- C. temperature reaches the threshold value of 95°F
- D. temperature reaches the threshold value of 185°F

Answer: A

Explanation: EP-3 requires securing Containment Spray before Containment pressure reaches the limit of 0 psig.

- B. Incorrect – The limit is 0 psig, not 1.23 psig. Plausible because 1.23 psig is another significant pressure in EP-3 (entry condition).
- C. Incorrect – Containment Spray is secured based on Containment pressure, not temperature. Plausible because it was initiated based on temperature and 95°F is a significant temperature in EP-3 (entry condition).
- D. Incorrect – Containment Spray is secured based on Containment pressure, not temperature. Plausible because it was initiated based on temperature and 185°F is a significant temperature in EP-3 (EP-2 entry and initiation of spray).

Technical Reference(s): EP-3

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-EP3 Obj. 9

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Level of Difficulty (LOD): 3

Comments:

TRH 11/17/23 – Re-worded question and answers slightly to avoid B being arguable as correct and capitalized “before” based on NRC comments.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295009 2.1.45
	Importance Rating	4.3

Low Reactor Water Level**Ability to identify and interpret diverse indications to validate the response of another indication**

Question #25

The plant has experienced a transient with the following:

- Reactor pressure is 950 psig and steady.
- Wide range Reactor water level indicates -130" and lowering slowly.
- Fuel zone Reactor water level indicates -180" and lowering slowly.

Which one of the following describes the reason for the difference in these Reactor water level indications and the status of actual Reactor water level?

The difference in Reactor water level indication is due to (1) .

Actual Reactor water level is (2) Top of Active Fuel (TAF).

- A. (1) core flow
 (2) above
- B. (1) core flow
 (2) below
- C. (1) calibration conditions
 (2) above
- D. (1) calibration conditions
 (2) below

Answer: C

Explanation: Wide range level indication is calibrated for rated pressure conditions. Fuel zone indication is calibrated for accident conditions (reactor depressurized, hot drywell, and no core flow). As a result, fuel zone indication will read lower than actual at 950 psig, (by as much as 70 inches). Therefore, wide range level is representative of actual level, and it is above top of active fuel.

- A. Incorrect – Based on the given wide range level indication, recirculation pumps are tripped and core flow due to natural circulation is minimal. Additionally, forced recirculation causes fuel zone indication to read higher not lower. 2nd part is correct. Plausible if not recognize that level differences are due to calibration conditions and the effects of those conditions.
- B. Incorrect – Actual level is approximately -130" which is above top of active fuel (-166") Plausible if not recognize that level differences are due to calibration conditions and the effects of those conditions. Based on the given wide range level indication, recirculation pumps are tripped and core flow due to natural circulation is minimal. Additionally, forced recirculation causes fuel zone indication to read higher not lower. 2nd part is correct. Plausible if not recognize that level differences are due to calibration conditions and the effects of those conditions.
- D. Incorrect – Actual level is approximately -130" which is above top of active fuel (-166") Plausible if not recognize that level differences are due to calibration conditions and the effects of those conditions.

Technical Reference(s): GLP-OPS-B2101

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-B2101 Obj. 11.1

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(2)

Level of Difficulty (LOD): 3

Comments:

TRH 11/17/23 – Swapped choices B and C and un-capitalized "Below" based on NRC comments.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295014 AA1.10
	Importance Rating	3.5

Inadvertent Reactivity Addition**Ability to operate and/or monitor the following as they apply to Inadvertent Reactivity Addition: HPCS**

Question #26

The plant is operating at 100% power following a refueling outage.

Then, HPCS spuriously starts and is injecting into the Reactor.

Which one of the following describes:

- the response of Reactor power to the HPCS injection
and
- the **first** required action to stop HPCS injection, in accordance with 04-1-01-E22-1, High Pressure Core Spray System SOI?

	<u>Reactor Power</u>	<u>First Required Action to Stop HPCS Injection</u>
A.	Rises	Trip HPCS pump
B.	Rises	Close 1E22-F004, HPCS INJ SHUTOFF VLV
C.	Lowers	Trip HPCS pump
D.	Lowers	Close 1E22-F004, HPCS INJ SHUTOFF VLV

Answer: A

Explanation: The HPCS injection adds relatively cold water to the Reactor and raises Reactor water level. These changes cause Reactor power to rise. 04-1-01-E22-1 Attachment V directs first tripping the HPCS pump.

- B. Incorrect – 04-1-01-E22-1 Attachment V directs first tripping the HPCS pump. Plausible because this is another action that will stop HPCS flow and is directed next by the procedure.
- C. Incorrect – Reactor power rises. Plausible because this would be correct if HPCS injected at EOL due to top peaking flux in the core resulting in a reverse reactivity effect.
- D. Incorrect – Reactor power rises. Plausible because this would be correct if HPCS injected at EOL due to top peaking flux in the core resulting in a reverse reactivity effect. 04-1-01-E22-1 Attachment V directs first tripping the HPCS pump. Plausible because this is another action that will stop HPCS flow and is directed next by the procedure.

Technical Reference(s): 04-1-01-E22-1 Attachment V

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-E2201 Obj 19.2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(5)

Level of Difficulty (LOD): 3

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	400000 K3.19
	Importance Rating	2.8

Component Cooling Water

Knowledge of the effect that a loss or malfunction of the Component Cooling Water System will have on the following systems or system parameters: Reactor/turbine pressure regulating system

Question #27

The plant is operating at 100% power when the sensed temperature to TIC-R010, TBCW Heat Exchanger Outlet Temperature Controller, fails low.

Which one of the following describes an impact of this failure?

- A. EHC Coolers receive colder cooling water
- B. EHC Coolers receive warmer cooling water
- C. Circulating Pump Motor Thrust Bearing Coolers receive colder cooling water
- D. Circulating Pump Motor Thrust Bearing Coolers receive warmer cooling water

Answer: B

Explanation: TIC-R010 operates in automatic to control TBCW heat exchanger outlet temperature. With sensed temperature failing low, the TIC will attempt to raise temperature back towards setpoint. Actual TBCW outlet temperature will rise. EHC Coolers are supplied cooling water from TBCW, therefore EHC Coolers will receive warmer cooling water.

- A. Incorrect – EHC Coolers will receive warmer cooling water. Plausible because sensed temperature is colder and this would be correct for the opposite failure.
- C. Incorrect – EHC Coolers will be affected, not Circulating Pump Motor Thrust Bearing Coolers. Plausible because Circulating Pump Motor Thrust Bearing Coolers is cooled by Plant Chilled Water.
- D. Incorrect – EHC Coolers will be affected, not Circulating Pump Motor Thrust Bearing Coolers. Plausible because Circulating Pump Motor Thrust Bearing Coolers is cooled by Plant Chilled Water.

Technical Reference(s): 05-1-02-V-2

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-N3201 Obj. 10.1

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Level of Difficulty (LOD): 4

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	215004 K3.03
	Importance Rating	3.9

Source-Range Monitor

Knowledge of the effect that a loss or malfunction of the Source Range Monitor System will have on the following systems or system parameters: RCIS

Question #28

A normal Reactor startup is in progress with the following:

- The Reactor is subcritical.
- Control rods are being withdrawn to achieve criticality.

Then, SRM 'E' spikes to 2.5×10^5 cps.

Which one of the following describes the response of RCIS and RPS?

- A. NO rod withdrawal block and NO half scram
- B. Rod withdrawal block and NO half scram
- C. Rod withdrawal block and half scram, only
- D. Rod withdrawal block and full scram

Answer: B

Explanation: A spike of any SRM during startup ($IRM < \text{range } 8$) of 2.5×10^5 cps is above the control rod withdrawal block setpoint of 1×10^5 cps. During a normal Reactor startup, the SRM shorting links are installed, therefore NO half scram occurs.

- A. Incorrect – A rod block is received. Plausible because this would be correct if the spike was below 1×10^5 cps
- C. Incorrect – NO half scram occurs. Plausible because the spike is above the upscale setpoint of 2×10^5 cps if shorting links were removed.
- D. Incorrect – NO half or full scram occurs. Plausible because the spike is above the upscale setpoint of 2×10^5 cps if shorting links were removed.

Technical Reference(s): GLP-OPS-C5101

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-C5101 obj. 8.2

Question Source: Bank – 5-2017 NRC #37

Question History: 5-2017 NRC #37

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Level of Difficulty (LOD): 3

Comments:

TRH 11/17/23 – Revised answer choices to get rid of unbalanced 2x2 table based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	215003 K2.01
	Importance Rating	3.4

Intermediate-Range Monitor**Knowledge of electrical power supplies to the following: IRM channels/detectors**

Question #29

Which one of the following is the power supply to the IRM detectors?

- A. BOP DC
- B. ESF Inverters
- C. BOP Inverters
- D. ESF MCC AC

Answer: B

Explanation: IRMs are powered from ESF Inverters 1Y87 (Panel 1Y89), 1Y88 (Panel 1Y84), 1Y95 (Panel 1Y86), and 1Y96 (Panel 1Y85).

- A. Incorrect – IRMs are powered from ESF Inverters. Plausible because BOP DC supplies power to many plant instrumentation loads.
- C. Incorrect – IRMs are powered from ESF Inverters. Plausible because BOP inverters supply similar power to non-ESF systems.
- D. Incorrect – IRMs are powered from ESF Inverters. Plausible because they do receive ESF AC power.

Technical Reference(s): 04-1-01-L62-1

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-C5102 obj. 6.3

Question Source: Bank – 2022 NRC #34

Question History: 2022 NRC #34

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Level of Difficulty (LOD): 4

Comments:

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Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	300000 K5.14
	Importance Rating	3.3

Instrument Air

Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the Instrument Air System: Air leaks

Question #30

The plant is operating at 100% power with the following:

- A significant leak develops in the Instrument Air system.
- Plant Air Receiver pressure is 88 psig and slowly lowering.

Which one of the following describes the resulting status of P51-F501, Service Air Pressure Reducing Station Valve, and P51-F504, Instrument Air Pressure Reducing Station "Backup" Valve?

	<u>P51-F501, Service Air Pressure Reducing Station Valve</u>	<u>P51-F504, Instrument Air Pressure Reducing Station "Backup" Valve</u>
A.	Open	Open
B.	Open	Closed
C.	Closed	Open
D.	Closed	Closed

Answer: C

Explanation: P51-F501, Service Air Pressure Reducing Station Valve, automatically closes when Plant Air Receiver pressure lowers below 100 psig in an attempt to preserve Instrument Air capacity. P51-F504, Instrument Air Pressure Reducing Station "Backup" Valve, automatically opens when downstream Instrument Air pressure lowers below 100 psig, in an attempt to maintain Instrument Air pressure at 100 psig. With Plant Air Receiver pressure at 85 psig, Instrument Air pressure must be below 100 psig, therefore P51-F504 is open.

- A. Incorrect – P51-F501 is closed. Plausible because the valve is normally open and would be open if air pressure was higher.
- B. Incorrect – P51-F501 is closed. Plausible because the valve is normally open and would be open if air pressure was higher. P51-F504 is open. Plausible because the valve is normally closed and would be closed if air pressure was higher.
- D. Incorrect – P51-F504 is open. Plausible because the valve is normally closed and would be closed if air pressure was higher.

Technical Reference(s): GLP-OPS-P5100

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-P5100 Objectives 18 and 19

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Level of Difficulty (LOD): 3

Comments:

TRH 11/20/23 – Wrote new question based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	263000 K4.03
	Importance Rating	2.9

DC Electrical Distribution

Knowledge of DC Electrical Distribution design features and/or interlocks that provide for the following: Ground detection

Question #31

Which one of the following describes the ground detector indication for DC Bus 11DA?

- A. Indicating lights on the distribution panel
- B. Indicating lights in the Main Control Room
- C. Digital ammeter on the distribution panel
- D. Digital ammeter in the Main Control Room

Answer: C

Explanation: The ground detector indication for DC Bus 11DA is a digital ammeter on the distribution panel.

- A. Incorrect – The ground detector indication for DC Bus 11DA is a digital ammeter on the distribution panel. Plausible because some ground detector indications consist of indicating lights.
- B. Incorrect – The ground detector indication for DC Bus 11DA is a digital ammeter on the distribution panel. Plausible because some ground detector indications consist of indicating lights. Also plausible because there is a Control Room annunciator that alarms given a ground.
- D. Incorrect – The ground detector indication for DC Bus 11DA is a digital ammeter on the distribution panel. Plausible because there is a Control Room annunciator that alarms given a ground.

Technical Reference(s): 04-1-01-L11-1

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-L1100 Obj. 9

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Level of Difficulty (LOD): 4

Comments:

TRH 11/17/23 – Revised “Battery” to “DC Bus” based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	263000 K5.01
	Importance Rating	3.1

DC Electrical Distribution

Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the DC Electrical Distribution: Hydrogen generation

Question #32

Which one of the following describes:

- an operational concern with charging of the DC system batteries and
- the type of charging that has the most risk of this issue?

Production of ____ (1) ____ is an operational concern when charging batteries.

This issue is of greatest risk during a(n) ____ (2) ____ charge.

	(1)	(2)
A.	Hydrogen	float
B.	Hydrogen	equalizing
C.	Hydrogen sulfide	float
D.	Hydrogen sulfide	equalizing

Answer: B

Explanation: An operational concern with charging of DC system batteries is generation of hydrogen. This risk is greatest during an equalizing charge.

- A. Incorrect – This risk is greatest during an equalizing charge. Plausible because this is another type of charge and does generate some hydrogen.
- C. Incorrect – The risk is hydrogen gas generation. Plausible because very small quantities of hydrogen sulfide are produced during heavy charging, but it is not an operational concern. This risk is greatest during an equalizing charge. Plausible because this is another type of charge and does generate some hydrogen.
- D. Incorrect – The risk is hydrogen gas generation. Plausible because very small quantities of hydrogen sulfide are produced during heavy charging, but it is not an operational concern.

Technical Reference(s): 04-1-01-L11-1

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-L1100 Obj 11.2

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Level of Difficulty (LOD): 3

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	400000 K6.13
	Importance Rating	3.3

Component Cooling Water

Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the Component Cooling Water System: Fuel pool cooling and cleanup system

Question #33

The plant is operating at 100% power when a tube leak develops in an in-service Fuel Pool Cleanup and Cooling heat exchanger.

Which one of the following describes the response of CCW radiation monitor indication and CCW surge tank level?

	<u>CCW Radiation Monitor Indication</u>	<u>CCW Surge Tank</u>
A.	Rises	Rises
B.	Rises	Lowers
C.	Remains the same	Rises
D.	Remains the same	Lowers

Answer: A

Explanation: FPCC pressure is greater than CCW pressure in the FPCC heat exchangers. Therefore, leakage will be from FPCC into CCW. This causes CCW radiation monitor indication to rise and makeup tank level to rise.

- B. Incorrect – CCW surge tank level rises. Plausible because tank level would lower given a leak in a different location.
- C. Incorrect – CCW radiation monitor indication rises. Plausible because this would be correct if CCW pressure was higher than FPCC pressure in the heat exchanger or if the leak was from a non-contaminated system.
- D. Incorrect – CCW radiation monitor indication rises. Plausible because this would be correct if CCW pressure was higher than FPCC pressure in the heat exchanger or if the leak was from a non-contaminated system. CCW surge tank level rises. Plausible because tank level would lower given a leak in a different location.

Technical Reference(s): ARI P870-5A-D2, ARI P601-18A-G5

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-G4146 Obj 13.1

Question Source: Modified Bank - NMP1 2020 NRC #25

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Level of Difficulty (LOD): 3

Comments:

TRH 11/17/23 – Specified that this is a tube leak based on NRC comment.

Examination Outline Cross-Reference: Level RO
 Tier # 2
 Group # 1
 K/A # 400000 K6.06
 Importance Rating 2.9

Component Cooling Water**Knowledge of the effect that a loss or malfunction of the following will have on the CCWS: Heat exchangers and condensers**

Proposed Question: #25

The plant is operating at 100% power with the following:

- All five Reactor Recirculation pumps (RRPs) are operating.
- A leak develops in the RRP 11 pump cooler.

Which one of the following identifies the response of RBCLC radiation monitor indication and CLC Makeup Tank level?

	<u>RBCLC Radiation Monitor Indication</u>	<u>CLC Makeup Tank Level</u>
A.	Rises	Rises
B.	Rises	Remains the same
C.	Remains the same	Rises
D.	Remains the same	Remains the same

Proposed Answer: B

Explanation: The RRP 11 pump cooler has Reactor coolant (~1020 psig, 545°F) on one side of the heat exchanger and RBCLC water (~100 psig, 90°F) on the other side. With a leak in the heat exchanger, the higher pressure and contaminated Reactor coolant enters the RBCLC system. This causes the RBCLC radiation monitor indication to rise. It also causes excess RBCLC water to flow through the overflow line near the CLC makeup tank. However, a check valve prevents this water from entering the CLC makeup tank and causing level to rise.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	209002 A4.14
	Importance Rating	3.3

High-Pressure Core Spray**Ability to manually operate and/or monitor in the control room: Test return valve**

Question #34

The plant is operating at 100% power with the following:

- HPCS is operating in test return to the CST with the following valves open:
 - E22-F010, HPCS INBD TEST RTN TO CST
 - E22-F011, HPCS OTBD TEST RTN TO CST
 - P11-F064, HPCS/RCIC TEST RTN TO CST
 - P11-F065, HPCS/RCIC TEST RTN TO CST

Then, a small loss of coolant accident occurs with the following:

- The Reactor is manually scrammed.
- Reactor water level reaches a low of -20" during the scram transient.
- Reactor water level is now 35" and stable.
- Drywell pressure is 2 psig and rising slowly.

Which one of the following describes the response of the test return valves?

	<u>E22-F010 and E22-F011</u>	<u>P11-F064 and P11-F065</u>
A.	Close	Close
B.	Close	Remain open
C.	Remain open	Close
D.	Remain open	Remain open

Answer: A

Explanation: E22-F010 and E22-F011 automatically close because a HPCS initiation signal is present (Drywell pressure >1.39 psig). P11-F064 and P11-F065 automatically close because Drywell pressure is above 1.23 psig.

- B. Incorrect – P11-F064 and P11-F065 automatically close. Plausible because Reactor water level remained above the HPCS initiation setpoint of -41.6" throughout the transient. Also plausible that only one pair of these valves would need to automatically close to isolate the flow path.
- C. Incorrect – E22-F010 and E22-F011 automatically close. Plausible because Reactor water level remained above the HPCS initiation setpoint of -41.6" throughout the transient. Also plausible that only one pair of these valves would need to automatically close to isolate the flow path.
- D. Incorrect – All of the given valves close. Plausible because Reactor water level remained above the HPCS initiation setpoint of -41.6" throughout the transient.

Technical Reference(s): ARI P601-16A-B4, ARI P601-16A-C3, GLP-OPS-E2201

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-E2201, Objective 9.1, 9.2

Question Source: Modified Bank – 2012 Audit #33

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Level of Difficulty (LOD): 3

Comments:

Examination Outline Cross-Reference	Level	RO
209002 High Pressure Core Spray System (HPCS) Ability to predict and/or monitor changes in parameters associated with operating the HIGH PRESSURE CORE SPRAY SYSTEM (HPCS) controls including: A1.08 System lineup: BWR-5,6	Tier #	2
	Group #	1
	K/A #	209002 A1.08
	Rating	3.1
	Revision	0

Question 33

HPCS is operating in test return to the CST when a small LOCA occurs.

Drywell pressure is 1.4 psig.

Reactor water level is +35" narrow range; wide range indicates +55".

Consider the following valves:

- E22-F004, HPCS INJ SHUTOFF VLV
- E22-F012, HPCS MIN FLO TO SUPP POOL
- E22-F010, HPCS INBD TEST RTN TO CST
- E22-F011, HPCS OTBD TEST RTN TO CST
- P11-F064, HPCS/RCIC TEST RTN TO CST
- P11-F065, HPCS/RCIC TEST RTN TO CST

Which of these valves are OPEN (all valve strokes are complete)?

- A. E22-F004, only
- B. All valves except E22-F004
- C. E22-F012, P11-F064 and P11-F065, only
- D. E22-F012, only

Answer: D			
Explanation:			
With a HPCS initiation signal present (>1.39psig) the E22-F010 and 11 auto closed and interlocked closed until initiation is reset. P11-F064 and 65 will auto close when drywell pressure reaches 1.23 psig. The E22-F004 auto closes when reactor water level reaches 53.5" wide range and will cause the E22-F012 to auto open on low flow.			
'D' is the only correct answer.			
Technical References:			

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	205000 K1.18
	Importance Rating	3.5

Shutdown Cooling

Knowledge of the physical connections and/or cause and effect relationships between the Shutdown Cooling System (RHR Shutdown Cooling Mode) and the following systems: Feedwater system

Question #35

Which one of the following systems can be aligned as the return flow path for RHR Shutdown Cooling flow to the Reactor?

- A. RCIC
- B. HPCS
- C. Recirc
- D. Feedwater

Answer: D

Explanation: RHR SDC flow to the Reactor can be aligned through Feedwater using E12-F053A/B.

- A. Incorrect – RHR SDC flow to the Reactor cannot be aligned to RCIC. Plausible because RCIC does inject to the Reactor.
- B. Incorrect – RHR SDC flow to the Reactor cannot be aligned to HPCS. Plausible because RCIC does inject to the Reactor.
- C. Incorrect – RHR SDC flow to the Reactor cannot be aligned to Recirc. Plausible because Recirc does inject to the Reactor and is used as the suction path for RHR SDC flow.

Technical Reference(s): 04-1-01-E12-2

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-E1200 Obj. 3.4

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(3)

Level of Difficulty (LOD): 2

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	215005 A2.04
	Importance Rating	4.4

Average Power Range Monitor/Local Power Range Monitor

Ability to (a) predict the impacts of the following on the Average Power Range Monitor/Local Power Range Monitor System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: SCRAM trip signals

Question #36

The plant is operating at 100% power with the following:

- APRM 4 spikes upscale for five (5) seconds and then returns to a normal reading.
- Annunciator P680-7A-B11, APRM CH 4 UPSC TRIP/OPRM TRIP/INOP, alarms.

Which one of the following describes the impact of this failure?

	<u>Initial Impact of Upscale Condition</u>	<u>Impact after Upscale Condition Clears</u>
A.	Half scram is received	Half scram automatically resets
B.	Half scram is received	Half scram remains until manually reset
C.	Half-trip signal to voters occurs, but NO half scram is received	Half-trip signal to voters automatically resets
D.	Half-trip signal to voters occurs, but NO half scram is received	Half-trip signal to voters remains until manually reset

Answer: C

Explanation: 4 digital APRM channels feed a single VOTER logic trip system. For a single APRM upscale trip, no half scram is produced, but only one "vote". Any 2 APRM channels will now produce a full scram, not a half-scram. No single APRM channel will produce a half-scram. This is different than all other RPS logic, which requires a trip in each of 2 trip systems to produce a full scram and where any channel tripping produces a half-scram. When APRM 4 returns to the normal value, the half-trip signal to the voters automatically resets without manual action.

- A. Incorrect – No half scram is received. Plausible because this would be correct for IRMs or for the original plant APRMs.
- B. Incorrect – No half scram is received. Plausible because this would be correct for IRMs or for the original plant APRMs.
- D. Incorrect – When APRM 4 returns to the normal value, the half-trip signal to the voters automatically resets without manual action. Plausible because if a half scram occurs, manual action is required to reset the half scram.

Technical Reference(s): ARI P680-7A-B11, 04-S-01-C51-1

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-C5104 Obj. 7.2

Question Source: Modified Bank - 2013 NRC #88

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Level of Difficulty (LOD): 3

Comments:

TRH 11/17/23 – Revised to avoid overlap with scenario event, based on NRC comment.

Question 88

The plant is at rated power.

APRM 4 fails to 120%.

The crew places APRM 4 in BYPASS on 1H13P680.

- (1) What is the impact of APRM 4 failing to 120%?
- (2) In addition to placing APRM 4 in BYPASS on 1H13P680, which of the following are other actions required per plant procedures?
- A. (1) Half-trip signal to voters, only. No half-scram.
(2) Initiate a Technical Specification (TS) LCO tracking report in ESOMS for APRM4.
- B. (1) Half-trip signal to voters, only. No half-scram.
(2) Initiate a Potential Tech Spec (PTS) LCO tracking report in ESOMS for APRM4.
- C. (1) Division 2 half-scram.
(2) Reset the half scram, and initiate a Technical Specification (TS) LCO tracking report in ESOMS for APRM4.
- D. (1) Division 2 half-scram.
(2) Reset the half scram, and initiate a Potential Tech Spec (PTS) LCO tracking report in ESOMS for APRM4.

3

Answer: B

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	264000 K3.04
	Importance Rating	3.9

Emergency Generators (Diesel/Jet)

Knowledge of the effect that a loss or malfunction of the Emergency Generators will have on the following systems or system parameters: Bus frequency/voltage

Question #37

A loss of offsite power and loss of coolant accident have occurred with the following:

- Drywell pressure is 3 psig and rising slowly.
- All EDGs have started and are powering their respective buses.

Then, the following occur:

- A lube oil leak causes Division 1 EDG lube oil pressure to lower to 5 psig.
- A governor failure causes Division 2 EDG speed to rise at 10 rpm per second.

Which one of the following describes the resulting status of Bus 15AA and Bus 16AB ten (10) seconds later?

	15AA	16AB
A.	Energized	Energized
B.	Energized	De-energized
C.	De-energized	Energized
D.	De-energized	De-energized

Answer: B

Explanation: The Division 1 EDG is initially powering Bus 15AA. Lube oil pressure drops below the normal trip setpoint of 30 psig. However, this trip is bypassed due to the high Drywell pressure (>1.39 psig). Therefore, Bus 15AA continues to be energized. The Division 2 EDG is initially powering Bus 16AB. Within 10 seconds, Division 2 EDG rises above the overspeed trip setpoint of 517.5 rpm ($450 \text{ rpm} + 10 \text{ rpm/sec} \times 10 \text{ sec} = 550 \text{ rpm}$). This trip is not bypassed by the high Drywell pressure signal, therefore the EDG trips and Bus 16AB de-energizes.

- A. Incorrect – Bus 16AB de-energizes. Plausible because other EDG trips are bypassed by the high Drywell pressure signal.
- C. Incorrect – Bus 15AA remains energized. Plausible because it would be de-energized if not for the high Drywell pressure signal. Bus 16AB de-energizes. Plausible because other EDG trips are bypassed by the high Drywell pressure signal.
- D. Incorrect – Bus 15AA remains energized. Plausible because it would be de-energized if not for the high Drywell pressure signal.

Technical Reference(s): 04-1-01-P75-1

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-P7500 Obj 13

Question Source: Bank – NMP2 2014 NRC #5

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Level of Difficulty (LOD): 3

Comments:

TRH 11/17/23 – Revised time in question from 20 seconds to 10 seconds based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	217000 2.1.23
	Importance Rating	4.3

Reactor Core Isolation Cooling**Ability to perform general or normal operating procedures during any plant condition****Question #38**

A small Feedwater line leak in the Drywell occurs with the following:

- Operators insert a manual Reactor scram.
- Both RFPTs trip.
- RCIC automatically initiates.
- All ECCS systems automatically initiate on high drywell pressure.
- Operators secure Condensate and Feedwater and isolate the leak.
- Reactor water level is restored to its normal band.
- All ECCS initiation logics have been reset.

Which one of the following describes the required operator action, if any, to return the RCIC gland seal compressor to its normal standby configuration, in accordance with 04-1-01-E51-1, RCIC SOI?

- A. Depress the RCIC INIT RESET pushbutton
- B. Open the gland seal compressor supply breaker then re-close it
- C. Momentarily take the RCIC GL SEAL COMPR handswitch to STOP and then release it
- D. Momentarily take the RCIC GL SEAL COMPR handswitch to START and then release it

Answer: C

Explanation: Normal standby configuration for the gland seal compressor is to have its “auto-start” feature armed. This is done per the system SOI, step 4.1.2.f, where the RO momentarily places the compressor handswitch to STOP and releases it back to its spring-return to neutral position. This allows the compressor to auto-start on a reactor water Level 2 automatic RCIC initiation (or manual pushbutton initiation from P601), so long as no Div 1 ECCS LOCA initiation signal is present at that time. Once running, the compressor will automatically trip if a Div 1 ECCS LOCA initiation signal is received. Once tripped, it cannot be manually re-started until an operator resets the Div 1 ECCS LOCA initiation logic. And even after that, in order to “return it to its normal standby configuration” (i.e., auto-start feature enabled), the operator must again momentarily place its handswitch to STOP.

- A. Incorrect – The RCIC GL SEAL COMPR handswitch must be taken to STOP and released. Plausible because this is an action related to resetting certain RCIC conditions.
- B. Incorrect – The RCIC GL SEAL COMPR handswitch must be taken to STOP and released. Plausible because this is similar in nature to cycling the handswitch.
- D. Incorrect – The RCIC GL SEAL COMPR handswitch must be taken to STOP and released. Plausible that the switch would need to be taken to START to either start it or enable automatic starting.

Technical Reference(s): 04-1-01-E51-1

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-E5100, Objective 7.16

Question Source: Bank – 2015 Audit #41

Question History: 2011 NRC #64

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Level of Difficulty (LOD): 3

Comments:

TRH 11/17/23 – Replaced original choice A based on NRC comment and moved it to D.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	223002 A3.01
	Importance Rating	4.0

Primary Containment Isolation/Nuclear Steam Supply Shutoff

Ability to monitor automatic operation of the Primary Containment Isolation System/Nuclear Steam Supply Shutoff, including: System indicating lights and alarms

Question #39

The plant is operating at 100% power.

E31-N604D, Main Steam Line D temperature monitor, fails upscale causing the following alarms:

- MSL PIPE TNL CH-D TEMP HI/INOP (P601-19A-A4)
- MSIV/DR VLV TRIP INIT (P601-19A-E4)

Then, the RPS 'A' M/G set trips.

Which of the following describes the impact on the MSIVs?

- A. NO MSIV isolation signals are received
- B. A half MSIV isolation signal is received on Div 1, only
- C. A half MSIV isolation signal is received on Div 2, only
- D. All MSIVs close

Answer: D

Explanation: RPS 'A' M/G set tripping causing a ½ scram and de-energizing the 'A' MSIV solenoids. The 'D' MSL Pipe Tunnel High causes the 'B' MSIV solenoids to de-energize, causing all MSIVs to close.

- A. Incorrect – All MSIVs close. Plausible that an isolation signal would require power (energize to function) and would require more than a single instrument to fail.
- B. Incorrect – All MSIVs close. Plausible because this would be correct given only one of these two failures.
- C. Incorrect – All MSIVs close. Plausible because this would be correct given only one of these two failures.

Technical Reference(s): 04-1-02-1H13-P601-19A-A4, 04-1-02-1H13-P601-19A-E4, 05-1-02-III-2

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-M7100 obj. 8.1, 11

Question Source: Bank - 2012 Audit #47

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Level of Difficulty (LOD): 3

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	212000 K1.06
	Importance Rating	3.9

Reactor Protection System

Knowledge of the physical connections and/or cause and effect relationships between the Reactor Protection System and the following systems: Control rod drive hydraulic system

Question #40

The plant is operating at 100% power with the in-service CRD Flow Control Valve in automatic when a Reactor scram occurs.

Which one of the following describes the CRD Flow Control Valve response?

The CRD Flow Control Valve...

- A. closes to allow maximum charging water header flow
- B. closes to allow maximum drive and cooling water header flow
- C. opens to allow maximum charging water header flow
- D. opens to allow maximum drive and cooling water header flow

Answer: A

Explanation: During normal operation, the CRD Flow Control Valve automatically maintains CRD total system flow at approximately 60 gpm. During a scram, CRD total system flow rises significantly due to flow through all scram inlet valves. This flow is sensed through CRD system flow element which is upstream from the charging header. The high flow signal is transmitted to the Flow Control Valve, which closes in an attempt to limit total system flow below setpoint. This allows maximum charging water header flow and restricts flow to the drive and cooling water headers.

- B. Incorrect – This allows maximum charging water header flow and restricts flow to the drive and cooling water headers. Plausible drive water and/or cooling water header flow would be important post-scram.
- C. Incorrect – The CRD Flow Control Valve closes in response to rising system flow. Plausible that it would open to provide the additional flow necessary to accomplish the scram.
- D. Incorrect – The CRD Flow Control Valve closes in response to rising system flow. Plausible that it would open to provide the additional flow necessary to accomplish the scram. This allows maximum charging water header flow and restricts flow to the drive and cooling water headers. Plausible drive water and/or cooling water header flow would be important post-scram.

Technical Reference(s): GLP-OPS-C111A

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-C111A Obj. 9.3

Question Source: Bank - JAF 16-1 NRC #31

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(6)

Level of Difficulty (LOD): 3

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	264000 K6.10
	Importance Rating	3.3

Emergency Generators (Diesel/Jet)**Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the Emergency Generators: Jacket water components****Question #41**

The plant is operating at 100% power when it is discovered that the Auxiliary Jacket Water pump for Diesel Generator (DG) 11 is unavailable for use.

Which one of the following describes the response of DG 11 if the DSL ENG RMT MAN START pushbutton on 1H13-P864 is pressed.?

DG 11 will...

- A. NOT start
- B. start and immediately trip on a low jacket water pressure signal
- C. start and eventually trip on a high jacket water temperature signal
- D. start and run normally

Answer: D

Explanation: The Auxiliary Jacket Water pump is a full capacity backup to the Engine Drive Jacket Water pump. It is not required for the DG to be operable. The DG will operate normally with just the Engine Driven Jacket Water pump available.

- A. Incorrect – The DG will operate normally with just the Engine Driven Jacket Water pump available. Plausible that pressure from the Auxiliary Jacket Water pump would be a start permissive for the DG.
- B. Incorrect – The DG will operate normally with just the Engine Driven Jacket Water pump available. Plausible that low pressure would result and be a trip of the DG.
- C. Incorrect – The DG will operate normally with just the Engine Driven Jacket Water pump available. Plausible that high temperature would result and be a trip of the DG (it is an alarm condition and does have a trip).

Technical Reference(s): 04-1-01-P75-1

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-P7500 Obj. 19

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Level of Difficulty (LOD): 3

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	262001 K4.01
	Importance Rating	3.5

AC Electrical Distribution

Knowledge of AC Electrical Distribution design features and/or interlocks that provide for the following: Lockouts

Question #42

The plant is operating at 100% power.

Which one of the following will trip breakers that result in de-energizing BOP Bus 11HD?

A lockout of transformer...

- A. BOP-11A
- B. BOP-12A
- C. BOP-13
- D. BOP-14

Answer: B

Explanation: Bus 11HD is normally powered from Bus 13R via transformer BOP-12B. Transformer BOP-12A also is associated with Bus 13R. Therefore, a lockout of transformer BOP-12A results in a trip of breakers that results in Buss 11HD de-energizing.

- A. Incorrect – Lockout of this transformer does not lead to de-energizing Bus 11HD. Plausible because it is similar in nature to transformer BOP-11A and would result in de-energizing other busses.
- C. Incorrect – Lockout of this transformer does not lead to de-energizing Bus 11HD. Plausible because it is similar in nature to transformer BOP-11A and would result in de-energizing other busses.
- D. Incorrect – Lockout of this transformer does not lead to de-energizing Bus 11HD. Plausible because it is similar in nature to transformer BOP-11A and would result in de-energizing other busses.

Technical Reference(s): 04-S-02-SH13-P807-2A-C6

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-R2700 Obj. 7

Question Source: Bank – 2011 NRC #72

Question History: 2011 NRC #72

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Level of Difficulty (LOD): 3

Comments:

TRH 12/15/23 – Deleted “directly” in question based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	262001 K5.01
	Importance Rating	3.9

AC Electrical Distribution

Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the AC Electrical Distribution: Paralleling AC sources

Question #43

The plant is operating at 100% power with the following:

- 06-OP-1P75-M-0002, Standby Diesel Generator 12 Function Test, is being performed.
- Diesel Generator (DG) 12 has just been started and is about to be connected to Bus 16AB.

Which one of the following describes the required manipulation of the DG 12 PRL CONT handswitch (Unit Parallel Switch on P864) to prepare for connecting DG 12 to 16AB, in accordance with 06-OP-1P75-M-0002?

Take the DG 12 PRL CONT handswitch to...

- A. PRL to place the governor in droop mode
- B. PRL to place the governor in isochronous mode
- C. RESET to place the governor in droop mode
- D. RESET to place the governor in isochronous mode

Answer: A

Explanation: To connect DG 12 to 16AB, the DG 12 PRL CONT handswitch is taken to PRL. This defeats parallel interlock and places the governor in the droop mode. This is required to ensure proper load sharing while in parallel with another AC source.

- B. Incorrect – This places the governor in the droop mode. Plausible because the isochronous mode is the other mode of governor operation.
- C. Incorrect – The switch is taken to PRL, not RESET. Plausible because the switch is taken to RESET at the end of the surveillance to restore the DG to the normal standby lineup.
- D. Incorrect – The switch is taken to PRL, not RESET. Plausible because the switch is taken to RESET at the end of the surveillance to restore the DG to the normal standby lineup
Plausible because the isochronous mode is the other mode of governor operation.

Technical Reference(s): 06-OP-1P75-M-0002

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-P7500 Obj. 15d

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Level of Difficulty (LOD): 3

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	211000 A3.08
	Importance Rating	4.1

Standby Liquid Control**Ability to monitor automatic operation of the Standby Liquid Control System, including:
System initiation**

Question #44

From its normal standby lineup, Div 1 SLC has been initiated from the Main Control Room.
(By turning C41-C001A key to Start at P601.)

Which one of the following happens ***last*** in the Div 1 SLC initiation sequence?

- A. Pump starts
- B. Squib valve fires
- C. Pump suction valve (F001A) opens
- D. SLC sends a signal to NSSSS to isolate RWCU

Answer: A

Explanation: Normal standby lineup has test tank suction (F031) fully closed and storage tank suction for SLC Pump A (F001A) fully closed. Taking the P601 control switch to START does the following in parallel: 1) immediately fires the 'A' squib; 2) gives an open signal to F001A; sends a signal to NSSSS to isolate RWCU. The pump won't start until it sees F001A fully open; therefore, the pump start is the LAST in the sequence given these answer choices.

- B. Incorrect – The pump starts last. Plausible that the squib would not fire until the pump successfully started.
- C. Incorrect – The pump starts last. Plausible because this valve does open as part of the start sequence and there is an alternate suction (test tank).
- D. Incorrect – The pump starts last. Plausible because RWCU isolation would only be required once SLC is successfully injecting to the Reactor.

Technical Reference(s): 04-1-01-C41-1, GLP-OPS-C4100

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-C4100 Obj 8

Question Source: Bank - 5-2017 Audit #33

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(6)

Level of Difficulty (LOD): 3

Comments:

TRH 11/21/23 – Not sure what to replace C with that's more plausible. Maybe "Test tank outlet valve (F031) closes" (playing on the idea that maybe pump is normally aligned with suction there, pump starts, other suction opens, then this valve closes)? Other options – NMP2 4/23 NRC #33 (squib response and RWCU response), NMP2 2021 NRC #34 (squib current response), JAF 18-1 NRC #33 (indication of proper flow)

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	261000 K6.04
	Importance Rating	3.4

Standby Gas Treatment

Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the Standby Gas Treatment System: Radiation monitoring system

Question #45

The plant is operating at 100% power with the following:

- A surveillance is being conducted on the Fuel Handling Area (FHA) Ventilation Exhaust Radiation Monitors.
- 'A' FHA Rad Monitor is INOP with its function switch in STANDBY.

Then, 'D' FHA Rad Monitor spikes to indicate 4.5 mrem/hr.

Which one of the following describes the resulting status of the Standby Gas Treatment system?

- A. 'A' SBGT train is operating, only
- B. 'B' SBGT train is operating, only
- C. Both SBGT trains are operating
- D. Both SBGT trains have a 1/2 initiation signal, but remain in standby

Answer: A

Explanation: Both the A and D FHA rad monitors input to the initiation logic for the A SBGT train. The INOP condition on A provides a $\frac{1}{2}$ initiation signal to DIV I logic; the D rad monitor being above 3.6 mrem/hr provides the other $\frac{1}{2}$ initiation signal to DIV I. Therefore, A SBGT train A is running. Neither of these rad monitors input to the B SBGT train initiation logic, so B SBGT remains in standby.

- B. Incorrect – B SBGT train remains in standby. Plausible because this would be correct if the rad monitors were changed to B and C.
- C. Incorrect – B SBGT train remains in standby. Plausible because this would be correct for a valid high rad condition.
- D. Incorrect – A SBGT train starts, and B remains in standby with no $\frac{1}{2}$ initiation signal. Plausible because this would be correct for a different combination of rad monitors.

Technical Reference(s): 04-1-02-1H13-P601-19A-B9, 04-1-02-1H13-P601-19A-C9,
04-1-01-T48-1

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-T4801 obj. 8.6

Question Source: Bank - 12-2017 NRC #45

Question History: 12-2017 NRC #45

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(11)

Level of Difficulty (LOD): 3

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	239002 K2.01
	Importance Rating	3.7

Safety Relief Valves**Knowledge of electrical power supplies to the following: SRV solenoids**

Question #46

Which one of the following describes the electrical power supply to the solenoids for the non-ADS SRVs?

- A. 120 VAC ESF power
- B. 125 VDC ESF power
- C. 120 VAC non-ESF power
- D. 125 VDC non-ESF power

Answer: B

Explanation: Each non-ADS SRV has two solenoids, each receiving power from 125 VDC ESF power supplies (11DA and 11DB).

- A. Incorrect – The power supply is 125 VDC ESF. Plausible because 120 VAC ESF supplies power to many solenoids.
- C. Incorrect – The power supply is 125 VDC ESF. Plausible because 120 VAC ESF supplies power to many solenoids. Also plausible that non-ADS SRVs would have a different (non-ESF) power supply from ADS SRVs.
- D. Incorrect – The power supply is 125 VDC ESF. Plausible because the power supply is 125 VDC. Also plausible that non-ADS SRVs would have a different (non-ESF) power supply from ADS SRVs.

Technical Reference(s): GLP-OPS-E2202

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-E2202 obj. 9.1

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Level of Difficulty (LOD): 3

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	262002 2.1.27
	Importance Rating	3.9

Uninterruptable Power Supply (AC/DC)**Knowledge of system purpose and/or function****Question #47**

ESF Inverter 1Y88 has switched to the alternate power source due to an undervoltage condition from their inverter.

Which of the following describes how this inverter is restored to its normal lineup once the undervoltage condition clears?

Once the undervoltage condition clears, the inverter...

- A. auto-transfers back to the inverter supply immediately
- B. auto-transfers back to the inverter supply after a time delay
- C. must be manually transferred back to the inverter supply using controls outside of the Main Control Room
- D. must be manually transferred back to the inverter supply using controls inside the Main Control Room

Answer: C

Explanation: 1Y88 is an ESF Inverter. On loss of inverter voltage, the loads will automatically swap to the alternate power source via the static switch. However, for ESF inverters, the loads do NOT automatically swap back to the inverter after restoration of voltage. The INVERTER TO LOAD pushbutton must be depressed to transfer the loads back to the inverter. This control is located outside of the Main Control Room.

- A. Incorrect – The INVERTER TO LOAD pushbutton must be depressed to transfer the loads back to the inverter. Plausible because the transfer is automatic for some other inverters.
- B. Incorrect – The INVERTER TO LOAD pushbutton must be depressed to transfer the loads back to the inverter. Plausible because the transfer is automatic for some other inverters. Also, plausible that the transfer would only happen after a time delay to prevent rapid cycling of the power supply.
- D. Incorrect – The required control to be manipulated is located outside of the Main Control Room. Plausible because there are alarms associated with 1Y99 in the Main Control Room, as well as many electrical system controls.

Technical Reference(s): 04-1-01-L62-1

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-L6200 obj. 4.1, 4.2, 5.1, 5.2

Question Source: Bank - 2011 Audit #40

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Level of Difficulty (LOD): 3

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	510000 A2.08
	Importance Rating	3.7

Service Water

Ability to (a) predict the impacts of the following on the Service Water System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Abnormal intake water level

Question #48

The plant is operating at 100% power with the following:

- Annunciator P854-1A-A5, PSW RADIAL WELL TROUBLE, is received.
- Radial Well header pressure is 103 psig.
- Radial Well 5 level is 10.8' and lowering.
- Radial Well 5 level is below the continuous operation limit.

Which one of the following describes the required control of Radial Well PSW pump discharge valves, in accordance with 05-1-02-V-11, Loss of Plant Service Water ONEP?

Throttle Radial Well 5 PSW pump discharge valve further...

- A. open. NO adjustment of other Radial Well PSW pump discharge valves is required.
- B. open. Throttle other Radial Well PSW pump discharge valves further closed.
- C. closed. NO adjustment of other Radial Well PSW pump discharge valves is required.
- D. closed. Throttle other Radial Well PSW pump discharge valves further open.

Answer: D

Explanation: With lowering level in Radial Well 5, the associated PSW pump discharge valve must be throttled further closed in an attempt to allow level in the well to recover. This lowers PSW pressure. To restore pressure, the other Radial Well discharge valves must be throttle further open.

- A. Incorrect – Radial Well 5 discharge valve must be throttled further open. Plausible because often to raise level in a system a discharge valve would need to be throttled further open. Other Radial Well PSW pump discharge valves must be throttled further open. Plausible because this is not directly related to raising Radial Well 5 level, but a secondary effect of the adjustment of Radial Well 5 PSW pump discharge valve.
- B. Incorrect – Radial Well 5 discharge valve must be throttled further open. Plausible because often to raise level in a system a discharge valve would need to be throttled further open.
- C. Incorrect – Other Radial Well PSW pump discharge valves must be throttled further open. Plausible because this is not directly related to raising Radial Well 5 level, but a secondary effect of the adjustment of Radial Well 5 PSW pump discharge valve.

Technical Reference(s): 05-1-02-V-11

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-P4447 Obj 26.1

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Level of Difficulty (LOD): 3

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	259002 A1.01
	Importance Rating	4.3

Reactor Water Level Control

Ability to predict and/or monitor changes in parameters associated with operation of the Reactor Water Level Control System, including: Reactor water level

Question #49

The plant is operating at 100% power with the following:

- Feedwater Level Control (FWLC) is in a normal Automatic lineup.
- Reactor water level is 36" and stable.

Then, the steam flow signal from transmitter C34-N030A to Feedwater level control fails instantaneously full downscale ("hard" failure).

Which one of the following describes the response of Reactor water level?

Reactor water level initially...

- A. rises significantly
- B. lowers significantly
- C. remains approximately steady with FWLC remaining in three element control
- D. remains approximately steady with FWLC automatically swapping to single element control

Answer: D

Explanation: The Feedwater level control system receives 4 steam flow signals. When operating in a normal Automatic lineup at 100% power, the system is controlling in 3-element control. With a hard failure of a steam flow signal, the system automatically swaps to single element control. This prevents the failed signal from causing a Reactor water level perturbation. Therefore, Reactor water level remains steady.

- A. Incorrect – Reactor water level remains steady. Plausible because this would be correct for a slowly lowering Feedwater flow signal.
- B. Incorrect – Reactor water level remains steady. Plausible because this would be correct if the steam flow signal failed down slowly.
- C. Incorrect – FWLC swaps to single element control. Plausible that the system would ignore the failed instrument and control just based on the other 3 steam flow signals.

Technical Reference(s): 04-1-02-1H13-P680-2A-C9, GLP-OPS-C3400

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-C3400 5.3

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(4)

Level of Difficulty (LOD): 3

Comments:

TRH 11/19/23 – Changed to lower cognitive level based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	209001 A1.07
	Importance Rating	3.9

Low-Pressure Core Spray

Ability to predict and/or monitor changes in parameters associated with operation of the Low-Pressure Core Spray System, including: Emergency generator loading

Question #50

The plant has experienced a loss of offsite power and loss of coolant accident with the following:

- All Diesel Generators (DGs) have loaded onto their respective buses.
- LPCS is running on minimum flow.
- Reactor pressure is 50 psig.

Then, an operator fully opens E21-F005, LPCS INJ SHUTOFF VLV.

Which one of the following describes the impact on DG loading?

Loading...

- A. rises on DG A
- B. rises on DG B
- C. lowers on DG A
- D. lowers on DG B

Answer: A

Explanation: With LPCS operating on minimum flow at a Reactor pressure of 50 psig, opening E21-F005 raises LPCS flow. This raises current draw of the LPCS pump. The LPCS pumps receives power from 15AA, which is currently powered by DG A. Therefore, loading rises on DG A.

- B. Incorrect – DG A is affected, not B. Plausible that LPCS would be powered from the similar 16AB.
- C. Incorrect – Loading rises, not lowers. Plausible because this would be correct for other manipulations, such as lowering flow or stopping the pump. Also plausible that the pump would draw higher current at low flow, when it is pumping against higher resistance.
- D. Incorrect – Loading rises, not lowers. Plausible because this would be correct for other manipulations, such as lowering flow or stopping the pump. Also plausible that the pump would draw higher current at low flow, when it is pumping against higher resistance. DG A is affected, not B. Plausible that LPCS would be powered from the similar 16AB.

Technical Reference(s): 04-1-01-E21-1

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-E2100 Obj. 10.4

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Level of Difficulty (LOD): 2

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	203000 K4.06
	Importance Rating	3.5

RHR/LPCI: Injection Mode

Knowledge of RHR/LPCI: Injection Mode design features and/or interlocks that provide for the following: No-suction path pump trip

Question #51

The plant was operating at 100% power when a steam leak in the Drywell occurred with the following:

- Drywell pressure is 2.5 psig and rising slowly.
- Reactor water level is 34" and steady.
- All RHR pumps are running.

Then, E12-F004A, RHR PMP A SUCT FM SUPP POOL, spuriously closes.

Which one of the following describes the resulting status of RHR pump A?

- A. Trips on a low flow signal
- B. Trips on a valve position signal
- C. Trips on a low suction pressure signal
- D. Continues to run, but has NO suction path

Answer: B

Explanation: RHR pump A takes suction on the Suppression Pool through valve E12-F004A. When this valve begins to close, it begins to lower RHR pump A suction pressure and flow. The pump is provided with an automatic trip based on E12-F004A position to prevent it from continuing to run as suction pressure and flow continue to degrade.

- A. Incorrect – The pump trips on a valve position signal. Plausible because flow also lowers and some pumps do trip on low flow signals.
- C. Incorrect – The pump trips on a valve position signal. Plausible because suction pressure also lower and some pumps do trip on low suction pressure signals.
- D. Incorrect – The pump trips on a valve position signal. Plausible because this would be correct if an automatic trip was not provided, as is the case with some pumps.

Technical Reference(s): 04-1-01-E12-1, Electrical Drawing 1181-067

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-E1200 Obj. 7.6

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Level of Difficulty (LOD): 3

Comments:

TRH 11/19/23 – Added that plant was initially operating at 100% power based on NRC comment (ADHR would not be in service at 100% power, so it rules out an abnormal position for RHR A ADHRS MODE TRIP ENABLE HS-M649A).

Examination Outline Cross-Reference: Level RO
 Tier # 2
 Group # 1
 K/A # 218000 A4.05
 Importance Rating 4.0

Automatic Depressurization**Ability to manually operate and/or monitor in the control room: ADS timer reset**

Question #52

The plant experiences a loss of high-pressure injection and loss of coolant accident with the following sequence of events:

Time (mm:ss)	Conditions
00:00	The ADS timers begin timing due to valid ADS actuation signals.
01:00	Reactor water level rises above -150.3" and remains above this level for the remainder of the transient.
02:00	Current time.

- Operators have NOT inhibited ADS.
- Reactor pressure has been below 1000 psig throughout this timeline.

Which one of the following describes the status of ADS during this timeline?

ADS valves...

- A. opened and remain open.
- B. opened, but then re-closed.
- C. remained closed throughout this timeline. At time 02:00, the ADS timers are timing.
- D. remained closed throughout this timeline. At time 02:00, the ADS timers are NOT timing.

Answer: D

Explanation: The ADS timers begin timing when Drywell pressure exceeds 1.39 psig and Reactor water level lowers below -150.3" (with a confirmatory Reactor water level signal of <+11.4"). The timer will open ADS valves after 105 seconds. However, the low Reactor water level signals do NOT seal-in. At time 00:00, conditions are met for ADS timers to start timing. At time 01:00, the low Reactor water level signal clears and the ADS timers stop timing. Since the ADS timers only timed for 60 seconds, they did not open ADS valves. At time 02:00, Reactor water level is still above -150.3", therefore the ADS timers are not timing.

- A. Incorrect – ADS valves never opened. Plausible because the timers did get an initiation signal and this would be correct if the Reactor water level signals sealed-in.
- B. Incorrect – ADS valves never opened. Plausible because the timers did get an initiation signal and this would be correct if the timers were less than 60 seconds and did not seal-in.
- C. Incorrect – At time 02:00, the ADS timers are NOT timing. Plausible because the timers did get an initiation signal and there is a 9.2 minute timer also associated with ADS.

Technical Reference(s): GLP-OPS-E2202

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-E2202 obj. 15

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Level of Difficulty (LOD): 4

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	219000 K3.03
	Importance Rating	3.9

RHR/LPCI: Torus/Suppression Pool Cooling Mode

Knowledge of the effect that a loss or malfunction of the RHR/LPCI: Torus/Suppression Pool Cooling Mode will have on the following systems or system parameters: Primary containment

Question #53

The plant has experienced an accident with the following:

- Reactor pressure is 0 psig.
- Reactor water level is 25" and stable with LPCS injecting.
- RHR A is operating in Suppression Pool Cooling.

It is discovered that valve E12-F024A, RHR A TEST RTN TO SUPP POOL, is stuck in its current position.

Then, Containment pressure rises and causes an automatic Containment Spray initiation signal.

Which one of the following describes the response of RHR A?

E12-F028A, CTMT SPR A SPARGER INJ VLV,...

- A. opens. Flow through the Suppression Pool Cooling lineup continues.
- B. opens. Flow through the Suppression Pool Cooling lineup is terminated.
- C. remains closed. Flow through the Suppression Pool Cooling lineup continues.
- D. remains closed. Flow through the Suppression Pool Cooling lineup is terminated.

Answer: A

Explanation: E12-F024A is opened to place RHR A in the Suppression Pool Cooling lineup. E12-F024A receives a close signal upon automatic initiation of Containment Spray. With the valve stuck in the open position, it will be unable to close on this signal. E12-F028A receives an open signal upon the automatic initiation of Containment Spray. This valve does not need a permissive signal from E12-F024A closing prior to opening. Therefore, E12-F028A does open. With E12-F024A failed open, there is no other valve closure that terminates the Suppression Pool Cooling lineup. Therefore, flow through the Suppression Pool Cooling lineup continues.

- B. Incorrect – Flow through the Suppression Pool Cooling lineup continues. Plausible that some other valve would receive a close signal and would terminate the Suppression Pool Cooling lineup.
- C. Incorrect – E12-F028A opens. Plausible that it would need a permissive signal from E12-F024A closing before it opens to prevent these lineups being simultaneously aligned.
- D. Incorrect – E12-F028A opens. Plausible that it would need a permissive signal from E12-F024A closing before it opens to prevent these lineups being simultaneously aligned. Flow through the Suppression Pool Cooling lineup continues. Plausible that some other valve would receive a close signal and would terminate the Suppression Pool Cooling lineup.

Technical Reference(s): 04-1-01-E12-1, E1181-41

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-E1200 Obj. 7.11

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Level of Difficulty (LOD): 3

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	268000 A4.02
	Importance Rating	3.1

Radwaste**Ability to manually operate and/or monitor in the control room: Primary containment sump pumps**

Question #54

The plant is operating at 100% power with the following:

- Containment floor drain sump pump mode handswitch is in AUTO B/STANDBY A.

Then, the following annunciators are received:

- P680-8A1-C6, CTMT FLOOR DR SMP LVL HI-HI
- P680-8A1-D6, CTMT FLOOR DR SMP TEMP HI

Which one of the following describes the resulting status of Containment floor drain sump pumps?

	A	B
A.	NOT running	NOT running
B.	NOT running	Running
C.	Running	NOT running
D.	Running	Running

Answer: D

Explanation: With the Containment floor drain sump pump mode handswitch is in AUTO B/STANDBY A, pump B performs normal sump pump downs and pump A starts on hi-hi sump level. With annunciator P680-8A1-C6, CTMT FLOOR DR SMP LVL HI-HI, in alarm, both pumps are running.

- A. Incorrect – Both pumps are running. Plausible because this would be correct if either the hi-hi level or hi temp conditions caused an isolation of this sump.
- B. Incorrect – Both pumps are running. Plausible because this would be correct if level was high but below the hi-hi alarm setpoint.
- C. Incorrect – Both pumps are running. Plausible because this would be correct if level was high but below the hi-hi alarm setpoint and the mode handswitch was in AUTO A/STANDBY B.

Technical Reference(s): ARI P680-8A1-C6, ARI P680-8A1-D6

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-P4500 Obj. 7.1

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(9)

Level of Difficulty (LOD): 3

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	290001 A3.01
	Importance Rating	4.1

Secondary Containment**Ability to monitor automatic operation of the Secondary Containment, including:
Secondary containment isolation**

Question #55

The plant is operating at 100% power with the following:

- A loss of Feedwater occurs.
- The Reactor scrams.
- HPCS and RCIC automatically initiate.
- Reactor water level reaches a low of -45".

Which one of the following identifies an isolation valve whose position indication has changed status (from OPEN to CLOSED) on the Isolation Valve Status Board?

- A. B21-F028A, MSL A CTMT OUTBOARD ISOL
- B. P11-F062, CST WATER SUPPLY HEADER TO AUX BLDG
- C. B21-F019, INBOARD MSL DRAINS OUTBOARD DRAIN VALVE
- D. P11-F047, REFUEL WATER TRANSFER PUMP SUCTION FROM RADWASTE

Answer: B

Explanation: P11-F062 is a normally open Auxiliary Building (Secondary Containment) isolation valve that closes at Level 2 (-41.6").

- A. Incorrect – This is a Group 1 PCIV. Group 1 valves don't close until level lowers to Level 1 (-150.3"). Plausible because this valve would close if Reactor water level had lowered further.
- C. Incorrect – This is a Group 1 PCIV. Group 1 valves don't close until level lowers to Level 1 (-150.3"). Plausible because this valve would close if Reactor water level had lowered further.
- D. Incorrect – This valve is initially closed at 100% power. Plausible because this valve does get an isolation signal based on Reactor water level <-41.6".

Technical Reference(s): 05-1-02-III-5

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-P5100 obj. 23

Question Source: Bank – 2012 NRC #59

Question History: 2012 NRC #59

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Level of Difficulty (LOD): 3

Comments:

TRH 11/19/23 – Revised A and D based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	290002 2.1.28
	Importance Rating	4.1

Reactor Vessel Internals**Knowledge of the purpose and function of major system components and controls**

Question #56

Which one of the following RPV internal components has the purpose of separating the water that is already flowing upward through the core from the water that is flowing downward through the downcomer annulus?

- A. Jet Pump
- B. Baffle Plate
- C. Core Shroud
- D. Lower Plenum

Answer: C

Explanation: The core shroud has the purpose of separating the water that is already flowing upward through the core from the water that is flowing downward through the downcomer annulus

- A. Incorrect – The jet pump is located outside of the core shroud and does not separate these 2 flow paths. Plausible because this is an RPV internal component located in the annulus and involved in directing flow.
- B. Incorrect – The baffle plate separates the downcomer annulus from the below core plate area, but does not provide separation between the two flows in this question. Plausible because this is an RPV internal component involved in separating flow paths.
- D. Incorrect – The lower plenum redirects the downcomer water to flow upward through the core plate and into the core, but does not provide separation between the two flows. Plausible because this is an RPV internal component involved in directing flow paths.

Technical Reference(s): GLP-OPS-B1300

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-B1300 obj. 4.1

Question Source: Bank - 2010 NRC #37

Question History: 2010 NRC #37

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(3)

Level of Difficulty (LOD): 2

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	241000 K1.39
	Importance Rating	3.3

Reactor/Turbine Pressure Regulating

Knowledge of the physical connections and/or cause and effect relationships between the Reactor/Turbine Pressure Regulating System and the following systems: Main and reheat steam system

Question #57

A plant power ascension is in progress.

Reactor power is raised from 70% to 100%.

Which one of the following describes how Turbine Control Valves and Reactor pressure respond to this power change?

	<u>Turbine Control Valves</u>	<u>Reactor Pressure</u>
A.	Throttle further open	Remains steady
B.	Throttle further open	Rises
C.	Throttle further closed	Remains steady
D.	Throttle further closed	Rises

Answer: B

Explanation: Over the course of this power rise, EHC throttles Turbine Control Valves further open to control Main Turbine inlet pressure. Steam flow through the Main Steam Lines rises as this occurs, resulting in higher head loss from the Reactor to the Main Turbine inlet. This results in Reactor pressure rising over the course of this power rise.

- A. Incorrect – Reactor pressure rises. Plausible that EHC would operate to maintain Reactor pressure steady over this power range.
- C. Incorrect – TCVs throttle further open. Plausible that the power rise would cause Reactor pressure to rise, which would require TCVs to throttle closed to stabilize the flow of steam to the Turbine. Reactor pressure rises. Plausible that EHC would operate to maintain Reactor pressure steady over this power range.
- D. Incorrect – TCVs throttle further open. Plausible that the power rise would cause Reactor pressure to rise, which would require TCVs to throttle closed to stabilize the flow of steam to the Turbine.

Technical Reference(s): GLP-OPS-N3202

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-N3202 Obj. 4.7, 9.1

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Level of Difficulty (LOD): 3

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	215001 K4.03
	Importance Rating	2.9

Traversing In-Core Probe

Knowledge of Traversing In-Core Probe design features and/or interlocks that provide for the following: Radiation shielding

Question #58

Which one of the following describes where the Traversing In-Core Probe (TIP) detectors are stored immediately following TIP operations?

Immediately following TIP operations, the TIP detectors are stored...

- A. inside the Reactor vessel below the core bottom
- B. in the storage location within the Containment Wall
- C. in the storage location between the Weir Wall and Drywell wall
- D. inside the Reactor vessel inside a TIP tube of a peripheral fuel cell

Answer: A

Explanation: Because they become highly radioactive during operation, the TIP detectors pose a significant radiation hazard and must be shielded to minimize their contribution to local radiation levels. Following operation, short term radiation protection is provided by storing the detectors inside the reactor vessel (below the core) to allow radiation levels to decay. For the long term, the detectors may be withdrawn to their storage positions inside the drywell wall or left in the reactor vessel below the core. Current operating practice is to leave the detectors stored inside the reactor vessel below the core bottom.

- B. Incorrect – The TIP detectors are stored in the Reactor vessel below the core bottom. Plausible because they do have a designated storage location which is within the Drywell wall not the Containment wall.
- C. Incorrect – The TIP detectors are stored in the Reactor vessel below the core bottom. Plausible because they do have a designated storage location which is within the Drywell wall not between the Weir Wall and Drywell wall.
- D. Incorrect – The TIP detectors are stored in the Reactor vessel below the core bottom. Plausible because the TIPs are stored inside the RPV but not in a fuel cell anywhere.

Technical Reference(s): GLP-OPS-C5105

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-C5105 obj. 4.2, 4.3, 4.4

Question Source: Bank – 2022 NRC #57

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Level of Difficulty (LOD): 3

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	233000 K2.01
	Importance Rating	3.1

Fuel Pool Cooling/Cleanup**Knowledge of electrical power supplies to the following: Fuel pool cooling pumps**

Question #59

Which one of the following is the power supply to Fuel Pool Cooling Pump A?

- A. 11B13
- B. 12B52
- C. 15BA3
- D. 16BB1

Answer: C

Explanation: The power supply to FPCC pump A is 15BA3.

- A. Incorrect – The power supply is 15BA3. Plausible because this is the power supply to the FPCC precoat pump.
- B. Incorrect – The power supply is 15BA3. Plausible because this is the power supply to FPCC holding pump A.
- D. Incorrect – The power supply is 15BA3. Plausible because this is the power supply to FPCC pump B.

Technical Reference(s): 04-1-01-G41-1

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-G4146 Obj. 8.1

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(13)

Level of Difficulty (LOD): 3

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	510001 A2.04
	Importance Rating	3.2

Circulating Water

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

Question #60

The plant is operating at 100% power with the following:

- Flooding is occurring in the Condenser Bay area.
- 05-1-02-VI-1, Flooding ONEP, has been entered.

Which one of the following:

- alarms would confirm that the flooding is from the Circulating Water system
and
- is the required operator action if the flooding is from the Circulating Water system,

in accordance with 05-1-02-VI-1?

	Alarm	Required Action
A.	P680-1A-D3, CNDS DMIN EFL CNDCT HI	Trip Circulating Water pumps and verify N71-F002A & B, Circ Water Pump Discharge, close
B.	P680-1A-D3, CNDS DMIN EFL CNDCT HI	Close N71-F037A & B, PSW Supply to Circ Water Pit, and leave Circulating Water pumps running
C.	P680-10A-D12, CLG TWR LOOP A FLO LO	Trip Circulating Water pumps and verify N71-F002A & B, Circ Water Pump Discharge, close
D.	P680-10A-D12, CLG TWR LOOP A FLO LO	Close N71-F037A & B, PSW Supply to Circ Water Pit, and leave Circulating Water pumps running

Answer: C

Explanation: Flooding from Circulating Water into the Condenser Bay area would be confirmed by receipt of annunciator P680-10A-D12, CLG TWR LOOP A FLO LO, due to diversion of Circulating Water flow to the cooling tower through the break. Given flooding from the Circulating Water system, 05-1-02-VI-1 requires tripping Circulating water pumps and verifying their discharge valves close.

- A. Incorrect – This annunciator would not confirm flooding from Circulating Water into the Condenser Bay area. Plausible because this annunciator would be received if the leak from Circulating Water was through the Condenser tubes into the Hotwell.
- B. Incorrect – This annunciator would not confirm flooding from Circulating Water into the Condenser Bay area. Plausible because this annunciator would be received if the leak from Circulating Water was through the Condenser tubes into the Hotwell. The required action includes tripping Circulating Water pumps. Plausible because closing N71-F037A & B is another action in the procedure for flooding from Circulating Water system. Also plausible because tripping of Circulating Water pumps is a major action that has a significant impact on continued power operation.
- D. Incorrect – The required action includes tripping Circulating Water pumps. Plausible because closing N71-F037A & B is another action in the procedure for flooding from Circulating Water system. Also plausible because tripping of Circulating Water pumps is a major action that has a significant impact on continued power operation.

Technical Reference(s): 05-1-02-VI-1

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-N7100 Obj. 17

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Level of Difficulty (LOD): 3

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	256000 K6.09
	Importance Rating	2.6

Condensate**Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the Condensate System: Offgas system****Question #61**

The plant is operating at 100% power when both Offgas Post Treatment Radiation Monitors FAIL to an INOP condition.

Which one of the following describes the impact on Offgas flow rate and Main Condenser vacuum?

	<u>Offgas Flow Rate</u>	<u>Main Condenser Vacuum</u>
A.	Rises	Improves
B.	Rises	Degrades
C.	Lowers	Improves
D.	Lowers	Degrades

Answer: D

Explanation: The Main condenser is a component of the Condensate System. When both Offgas post treatment rad monitors fail to an INOP condition causes an N64-F060, Offgas Discharge to Vent Vlv, isolation. Offgas no longer has flow for removal of air and non-condensable gases. Main Condenser Vacuum will begin to degrade (vacuum will lower).

- A. Incorrect – Offgas flow rate lowers. Plausible that the INOP failure would cause some other response, such as higher flow to provide more filtration/gas removal. Main Condenser vacuum degrades. Plausible because some changes in Offgas system performance would cause Main Condenser vacuum to improve.
- B. Incorrect – Offgas flow rate lowers. Plausible that the INOP failure would cause some other response, such as higher flow to provide more filtration/gas removal.
- C. Incorrect – Main Condenser vacuum degrades. Plausible because some changes in Offgas system performance would cause Main Condenser vacuum to improve.

Technical Reference(s): 04-1-02-1H13-P601-19A-C8, 04-1-01-N64-1

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-N1900 obj. 22

Question Source: Bank – 2022 NRC #62

Question History: 2022 NRC #62

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Level of Difficulty (LOD): 3

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	216000 A1.07
	Importance Rating	3.8

Nuclear Boiler Instrumentation

Ability to predict and/or monitor changes in parameters associated with operation of the Nuclear Boiler Instrumentation, including: RPV level

Question #62

Given the following:

- Reactor coolant temperature is 120°F.
- Drywell temperature is 135°F.
- Actual Reactor water level is +35".
- Recirculation pumps A and B are running in slow speed.

Which one of the following describes how these conditions affect Narrow Range Reactor water level instruments?

Narrow Range Reactor water level instruments are indicating...

- A. lower than actual Reactor water level due to Drywell temperature
- B. higher than actual Reactor water level due to Drywell temperature
- C. lower than actual Reactor water level due to Reactor coolant temperature
- D. higher than actual Reactor water level due to Reactor coolant temperature

Answer: D

Explanation: The Narrow Range Reactor water level instruments are calibrated for Reactor pressure of 1025 psig and Drywell temperature of 135°F. With Reactor coolant temperature at 120°F, Reactor pressure is 0 psig. This creates an error in the Narrow Range Reactor water level indication. The lower Reactor coolant temperature means the density of the fluid on the variable leg is higher than expected. This higher density creates higher pressure on the variable leg side of the D/P sensor. This results in the instrument indicating higher than actual Reactor water level.

- A. Incorrect – Drywell temperature is not inducing an error because it is at calibration condition. Plausible because Drywell temperature would also induce an error if it deviated from calibration conditions.
- B. Incorrect – Drywell temperature is not inducing an error because it is at calibration condition. Plausible because Drywell temperature would also induce an error if it deviated from calibration conditions.
- C. Incorrect – Narrow Range Reactor water level instruments indicate higher than actual Reactor water level. Plausible because they would indicate lower than actual if variable leg density was lower or if reference leg density was higher than for calibration conditions.

Technical Reference(s): GLP-OPS-B2101

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-B2101

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(2)

Level of Difficulty (LOD): 3

Comments:

TRH 11/19/23 – Re-wrote question based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	290003 K5.01
	Importance Rating	3.6

Control Room Ventilation

Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the Control Room Ventilation: Control of airborne contamination (e.g., radiological, toxic gas, smoke)

Question #63

The plant is operating at 100% power when an accident results in the following sequence of events:

Time (minutes)	Conditions
0	All 4 Control Room Vent Radiation Monitors are indicating 4.5 mRem/hr and rising slowly.
10	- Reactor water level is -41.6" and lowering slowly. - HPCS and RCIC auto initiate and begin to restore level.

Which one of the following describes the operation of Control Room Standby Fresh Air Ventilation system Fresh Air Inlet valves, Z51-F007 and Z51-F016?

These valves first receive an isolation signal at time (1) and can then first be manually opened at time (2).

	(1)	(2)
A.	0	0
B.	0	10
C.	10	10
D.	10	20

Answer: B

Explanation: On the receipt of a control room isolation signal of -41.6" RPV level or 1.23 psig D/W pressure or >4.5 mrem/hr on Control Room vent rad monitor, the SBFA unit will auto start in the Recirc mode. Fresh air inlet valves F007 and F016 will auto close and be interlocked closed for 10 minutes. Any subsequent initiation signal during the 10 min time delay has no effect.

- A. Incorrect – The valves are interlocked closed for 10 minutes. Plausible that manual action could be taken to reopen the valves if necessary immediately after the isolation has completed.
- C. Incorrect – The valves close at 0 minutes. Plausible because another isolation signal is received at time 10 minutes. The valves are interlocked closed for 10 minutes. Plausible that manual action could be taken to reopen the valves if necessary immediately after the isolation has completed.
- D. Incorrect – The valves close at 0 minutes. Plausible because another isolation signal is received at time 10 minutes.

Technical Reference(s): 04-S-01-Z51-1, GLP-OPS-Z5100, E-0131 SH 3

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-Z5100 obj. 8, 9 & 11

Question Source: Bank - 12-2017 NRC #65

Question History: 12-2017 NRC #65

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(11)

Level of Difficulty (LOD): 3

Comments:

TRH 11/19/23 – Revised stem conditions to exactly match setpoints based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.2.43
	Importance Rating	3.0

Knowledge of the process used to track inoperable alarms

Question #64

The plant is operating at 100% power with the following:

- One input to a multiple-input annunciator has failed.
- This single input to the annunciator has been bypassed.
- The rest of the inputs to the annunciator are functioning properly.

Which one of the following describes how the associated annunciator is required to be identified, in accordance with 02-S-01-25, Deficient Equipment Identification?

- A. NO identification is required for the alarm window
- B. Place two vertical lengths of red tape on the alarm window
- C. Place a length of red tape diagonally across the alarm window
- D. Place two lengths of red tape diagonally across the alarm window to form an 'X'

Answer: B

Explanation: Per 02-S-01-25, If an annunciator input is bypassed, the Annunciator window is marked as follows: Two vertical lengths of red tape are applied to distinguish an annunciator with a bypassed input.

- A. Incorrect – Two vertical lengths of red tape are applied to distinguish an annunciator with a bypassed input. Plausible because all other inputs to this annunciator are working correctly and the nuisance input is no longer causing spurious alarms.
- C. Incorrect – Two vertical lengths of red tape are applied to distinguish an annunciator with a bypassed input. Plausible because this would be correct for annunciators that are NOT functioning properly due to faulty alarm circuitry.
- D. Incorrect – Two vertical lengths of red tape are applied to distinguish an annunciator with a bypassed input. Plausible because this would be correct if the annunciator card was removed.

Technical Reference(s): 02-S-01-25

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-PROC Objective 33.4

Question Source: Bank - 2-2020 Audit #67

Question History: 2013 NRC #71

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Level of Difficulty (LOD): 3

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.1.4
	Importance Rating	3.3

Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, “no-solo” operation, maintenance of active license status, 10 CFR Part 55

Question #65

Which one of the following is the minimum number of twelve (12) hour shifts a licensed operator is required to complete in a calendar quarter to maintain an active license, in accordance with 02-S-01-39, Maintaining Watchstanding Proficiency?

- A. 3
- B. 5
- C. 7
- D. 9

Answer: B

Explanation: 02-S-01-39 requires a licensed operator to work a minimum of five (5) twelve (12) hour shifts each calendar quarter to maintain active license status.

- A. Incorrect – Five (5) twelve (12) hour shifts is the minimum. Plausible because this would be equivalent to one (1) shift per month and would be nearly 40 hours, which is the time requirement for reactivating a license.
- C. Incorrect – Five (5) twelve (12) hour shifts is the minimum. Plausible because 7 is the minimum number of eight (8) hour shifts.
- D. Incorrect – Five (5) twelve (12) hour shifts is the minimum. Plausible because this would be equivalent to three (3) shifts per month and is well within the number of shifts a typical operator would work in a quarter.

Technical Reference(s): 02-S-01-39

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-PROC Objective 6.1

Question Source: Bank – NMP1 2023 NRC #64

Question History: NMP1 2023 NRC #64

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Level of Difficulty (LOD): 3

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.3.12
	Importance Rating	3.2

Knowledge of radiological safety principles and procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, or alignment of filters

Question #66

A plant startup is in progress with the following:

- Reactor power is 14%.
- Personnel are preparing to enter the Drywell to investigate a problem.
- NO emergency is in progress.

Which one of the following describes the ability to make this Drywell entry, in accordance with 01-S-06-7, Containment and Drywell Access Control?

- A. All control rods must be fully inserted prior to entry
- B. The Reactor must be made subcritical prior to entry, but all control rods are NOT required to be fully inserted
- C. Reactor power must be lowered to 10% or less prior to entry, but the Reactor does NOT need to be made subcritical
- D. The entry may be performed at the current Reactor power level

Answer: C

Explanation: 01-S-06-7 requires Reactor power to be less than or equal to 10% to allow personnel entry to the Drywell. There is no requirement to take the Reactor subcritical or fully insert all control rods.

- A. Incorrect – There is no requirement to fully insert all control rods. Plausible because there is a requirement to lower Reactor power and this would lower Reactor power as low as possible.
- B. Incorrect – There is no requirement to make the Reactor subcritical. Plausible because there is a requirement to lower Reactor power and this would lower Reactor power significantly.
- D. Incorrect – Reactor power must be lowered. Plausible because Reactor power is already relatively low.

Technical Reference(s): 01-S-06-7

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-PROC Obj. 15.4

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Level of Difficulty (LOD): 3

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.1.37
	Importance Rating	4.3

Knowledge of procedures, guidelines, or limitations associated with reactivity management

Question #67

A plant startup is in progress in accordance with 03-1-01-1, Cold Shutdown to Generator Carrying Minimum Load.

Which one of the following describes:

- the SRM count rate at which individual notch control rod withdrawal first becomes required
- and
- when this restriction is first released,

in accordance with 03-1-01-1?

	<u>When Individual Notch Control Rod Withdrawal First Becomes Required</u>	<u>When This Restriction Is First Released</u>
A.	3x the initial SRM count rate	Criticality is achieved
B.	3x the initial SRM count rate	Point of Adding Heat (POAH) is achieved
C.	10x the initial SRM count rate	Criticality is achieved
D.	10x the initial SRM count rate	Point of Adding Heat (POAH) is achieved

Answer: D

Explanation: 03-1-01-1 step 2.1.23 states, "Single notching is to begin at 10 times the initial SRM count rate. All further rod withdrawals are to be performed in individual notch withdrawal until Point of Adding Heat (POAH) is achieved."

- A. Incorrect – Individual notch withdrawal first becomes required at 10x the initial SRM count rate. Plausible because this is a significant amount of rise in SRM count rate. Also plausible because this limitation is often 3 count rate doublings at BWRs. The restriction is first released at the POAH. Plausible because criticality is a major milestone in the startup and operation of the Reactor.
- B. Incorrect – Individual notch withdrawal first becomes required at 10x the initial SRM count rate. Plausible because this is a significant amount of rise in SRM count rate. Also plausible because this limitation is often 3 count rate doublings at BWRs.
- C. Incorrect – The restriction is first released at the POAH. Plausible because criticality is a major milestone in the startup and operation of the Reactor.

Technical Reference(s): 03-1-01-1

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-IOI01 Obj. 3.5

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Level of Difficulty (LOD): 3

Comments:

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Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.2.22
	Importance Rating	4.0

Knowledge of limiting conditions for operation and safety limits

Question #68

The plant has experienced a transient with the following:

- Peak Reactor pressure was 1355 psig.
- Lowest Reactor water level was -105".

Which one of the following describes whether Reactor pressure and Reactor water level exceeded Safety Limits (SLs), in accordance with Technical Specifications?

	<u>Reactor Pressure</u>	<u>Reactor Water Level</u>
A.	Exceeded a SL	Exceeded a SL
B.	Exceeded a SL	Did NOT exceed a SL
C.	Did NOT exceed a SL	Exceeded a SL
D.	Did NOT exceed a SL	Did NOT exceed a SL

Answer: B

Explanation: The peak Reactor pressure remained below the Safety Limit of 1325 psig. The lowest Reactor water level remained above the Safety Limit of above the top of active irradiated fuel (-166").

- A. Incorrect – Reactor water level did not exceed a Safety Limit. Plausible because it did go very low, including below Level 2 (-41.6").
- C. Incorrect – Reactor pressure did exceed a Safety Limit. Plausible because this would be correct if Reactor pressure rose slightly less. Reactor water level did not exceed a Safety Limit. Plausible because it did go very low, including below Level 2 (-41.6").
- D. Incorrect – Reactor pressure did exceed a Safety Limit. Plausible because this would be correct if Reactor pressure rose slightly less.

Technical Reference(s): Technical Specifications 2.1.1.3 and 2.1.2

References to be provided to applicants during examination: None

Learning Objective:

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Level of Difficulty (LOD): 3

Comments:

TRH 11/19/23 – Revised to have Reactor pressure exceeding the Safety Limit and reclassified to lower cognitive level based on NRC comments.

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.4.35
	Importance Rating	3.8

Knowledge of non-licensed operator responsibilities during an emergency

Question #69

A Site Area Emergency has been declared with the following:

- All Emergency Response Facilities have been activated and are fully staffed.
- All extra non-licensed operators on shift have been assigned to assist in a Search and Rescue operation.

Which one of the following identifies the facility where these non-licensed operators need to report if assigned to a Search and Rescue team, in accordance with 10-S-01-18, Personnel Search and Rescue?

- A. Main Control Room
- B. Technical Support Center (TSC)
- C. Operations Support Center (OSC)
- D. Emergency Operations Facility (EOF)

Answer: C

Explanation: Once the ERFs are activated and staffed, the Operations Support Center is the location where on-site support personnel, including extra non-licensed operators, report during an emergency for forming Search and Rescue teams.

- A. Incorrect – The Operations Support Center is the location for forming Search and Rescue teams. Plausible because extra non-licensed operators would often be dispatched for field actions from the Main Control Room if an emergency was not in progress.
- B. Incorrect – The Operations Support Center is the location for forming Search and Rescue teams. Plausible because the Technical Support Center is involved in the decision-making process regarding Search and Rescue activities.
- D. Incorrect – The Operations Support Center is the location for forming Search and Rescue teams. Plausible because the Emergency Operations Facility is involved in overall oversight of the emergency response.

Technical Reference(s): 10-S-01-18

References to be provided to applicants during examination: None

Learning Objective: GLP-EP-EPTS26_011-1 Obj. 5.2

Question Source: Bank - NMP2 2023 NRC #69

Question History: NMP2 2023 NRC #69

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Level of Difficulty (LOD): 3

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	4
	Group #	
	K/A #	293006 K1.05
	Importance Rating	3.3

Fluid Statics and Dynamics**Explain operational implications of fluid/water hammer**

Question #70

Which one of the following is the operational concern with starting an RHR pump with the loop NOT full of water and adequately vented?

- A. Cavitation
- B. Overheating
- C. Thermal shock
- D. Water hammer

Answer: D

Explanation: The concern with starting an RHR pumps with the loops not full of water and adequately vented is water hammer.

- A. Incorrect – The concern is water hammer. Plausible because cavitation is a concern if there is inadequate Suppression Pool water level.
- B. Incorrect – The concern is water hammer. Plausible because overheating is a concern related to pump motor start limits.
- C. Incorrect – The concern is water hammer. Plausible because the concern is a pressure shock. Also, plausible because thermal shock is a concern in other applications in the plant.

Technical Reference(s): IN 87-10 Potential for Water Hammer During Restart of RHR Pumps
GLP-OPS-E1200

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-E1200 Obj. 17

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(14)

Level of Difficulty (LOD): 2

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	4
	Group #	
	K/A #	292004 K1.11
	Importance Rating	2.6

Reactivity Coefficients

Describe the effect on the magnitude of void coefficient from changes in the following:
Core void fraction

Question #71

Grand Gulf is operating at 60% power with the core coolant flow consisting of 80% water by volume and 20% steam by volume.

In this condition, the core void fraction is (1) %.

If the core void fraction increases by 5%, the void coefficient of reactivity will become (2) negative.

	(1)	(2)
A.	20	less
B.	20	more
C.	25	less
D.	25	more

Answer: B

Explanation: Void Fraction = (Volume of Steam)/(Volume of Steam and Water)

Therefore, Void Fraction = $0.2 / 1.0 = 20\%$.

Void coefficient becomes more negative as void fraction increases.

- A. Incorrect – Void coefficient becomes more negative. Plausible because this is correct for other plant parameter changes.
- C. Incorrect – Void fraction is 20%. Plausible because 25% is derived by dividing 20% by 80%. Void coefficient becomes more negative. Plausible because this is correct for other plant parameter changes.
- D. Incorrect – Void fraction is 20%. Plausible because 25% is derived by dividing 20% by 80%.

Technical Reference(s): Academic Fundamentals, General Electric, September 1984, Revision 1, pp. 4-16a and 4-21a.

References to be provided to applicants during examination: None

Learning Objective: BR04Ir4_Coefficients 8

Question Source: Bank – NRC GFES bank B7717

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(1)

Level of Difficulty (LOD): 4

Comments:

TRH 11/19/23 – Added bank question number based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	4
	Group #	
	K/A #	293004 K1.14
	Importance Rating	2.7

Thermodynamic Processes**Explain the condensing process**

Question #72

The plant is operating at 100% power with the following:

- Main Condenser vacuum is 26.5" Hg.
- Circulating Water supply temperature is 75°F.

Which one of the following describes the effect on Main Condenser vacuum if Circulating Water supply temperature lowers to 65°F?

Main Condenser vacuum...

- A. rises (improves) due to lower saturation pressure in the condenser
- B. rises (improves) due to better performance of steam jet air ejectors
- C. lowers (degrades) due to higher saturation pressure in the condenser
- D. lowers (degrades) due to reduced performance of steam jet air ejectors

Answer: A

Explanation: A lower Circulating Water supply temperature results in a lower Condenser inlet temperature. A lower Condenser inlet temperature results in more energy being extracted from the steam in the Main Condenser, which results in more complete condensation, and a rise (improvement) in Main Condenser vacuum.

- B. Incorrect – A Circulating Water supply temperature change does not cause an appreciable change in steam jet air ejector performance. Plausible because steam jet air ejector performance does affect Main Condenser vacuum, and other parameters, such as steam supply pressure, do affect steam jet air ejector performance.
- C. Incorrect – Main Condenser vacuum rises (improves). Plausible because vacuum would lower (degrade) given other changes, such as rising Condenser inlet temperature or lower Circulating Water flow rate.
- D. Incorrect – Main Condenser vacuum rises (improves). Plausible because vacuum would lower (degrade) given other changes, such as rising Condenser inlet temperature or lower Circulating Water flow rate. A Circulating Water supply temperature change does not cause an appreciable change in steam jet air ejector performance. Plausible because steam jet air ejector performance does affect Main Condenser vacuum, and other parameters, such as steam supply pressure, do affect steam jet air ejector performance.

Technical Reference(s): BT04Ir4_Processes

References to be provided to applicants during examination: None

Learning Objective: BT04Ir4_Processes 17

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(14)

Level of Difficulty (LOD): 3

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	4
	Group #	
	K/A #	292005 K1.10
	Importance Rating	3.3

Control Rods**State the purpose of flux shaping**

Question #73

Which one of the following states a primary reason for establishing specific control rod patterns to shape neutron flux?

- A. Equalize burn up of control rods
- B. Minimize local power peaks in the core
- C. Prevent conversion of uranium into plutonium in the core
- D. Reduce excessive localized Reactor vessel neutron irradiation

Answer: B

Explanation: One of the primary reasons for flux shaping is to minimize local power peaks in the core. This aids in satisfying thermal limits, preventing damage to the fuel, and burning up the fuel more evenly.

- A. Incorrect – Equalizing burn up of control rods is not a primary reason for flux shaping. At 100% power, most control rods are fully withdrawn. Even with flux shaping, the flux profile is weighted towards the bottom center of the core, resulting in higher control rod burnup for the rods in this area, which are typically the rods that are partially inserted to shape flux. Plausible because having a more even flux profile would cause more even burn up of control rods than if the flux profile was more concentrated.
- C. Incorrect – Part of the control rod sequencing strategy is to create conditions early in cycle that actually promote conversion of uranium into plutonium in the core. Plausible because there are generally restrictions on plutonium production due to nuclear weapons proliferation concerns.
- D. Incorrect – Reactor vessel neutron irradiation is not a primary reason for flux shaping. Plausible because Reactor vessel neutron irradiation is a long term concern for the Reactor vessel's metal strength/embrittlement.

Technical Reference(s): BR0SIr4_Control_Rods

References to be provided to applicants during examination: None

Learning Objective: BR0SIr4_Control_Rods 10

Question Source: Bank – NRC GFES B1557 & B1656

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(1)

Level of Difficulty (LOD): 3

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	4
	Group #	
	K/A #	292008 K1.24
	Importance Rating	3.2

Reactor Operational Physics**Describe the parameters to be monitored and controlled during rod pattern exchanges**

Question #74

EN-RE-206, Control Rod Sequence Exchange, contains the following limitation for withdrawing control rods during a control rod pattern exchange:

“All interior control rods (control rods that are more than two rows from the periphery of the core) shall be shadowed by two other adjacent control rods that are inserted at least two notches beyond its position before it is withdrawn from the core.”

Which one of the following describes the reason for this limitation while withdrawing control rods?

- A. Provide increased control rod worth
- B. Provide margin to the rod line boundary
- C. Prevent localized transition boiling in adjacent bundles
- D. Prevent challenging thermal limits in adjacent bundles due to control rods “crossing tips”

Answer: D

Explanation: 17-S-02-400, CONTROL ROD SEQUENCES AND MOVEMENT CONTROL, states:

“When operating the Recirc pumps on FAST SPEED, control rod tips should not be pulled past adjacent partially inserted control rod tips unless specifically evaluated by a qualified Reactor Engineer and approved by the Reactor Engineering Supervisor. “

Therefore, the limitation in the question from EN-RE-206 describes the method used in the rod movement sequence to prevent crossing control rod tips. Crossing control rod tips results in a power spike when the control rod tips align during control rod withdrawal. This could challenge thermal limits in the affected bundles.

- A. Incorrect – The reason is to prevent a power spike from crossing tips. Plausible because rod worth does change from shadowing rods, however, it reduces rod worth, not increase rod worth.
- B. Incorrect – The reason is to prevent a power spike from crossing tips. Plausible because by having adjacent rods inserted beyond the rod being withdrawn, it does provide additional margin to the rod line boundary.
- C. Incorrect – The reason is to prevent a power spike from crossing tips. Plausible if it is not understood that transition boiling is associated with critical power ratio which is based on bundle power and not local power.

Technical Reference(s): EN-RE-206, 17-S-02-400

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-IOI02 Obj 2.11

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(1)

Level of Difficulty (LOD): 4

Comments:

TRH 12/15/23 – Revised wording of D slightly based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	4
	Group #	
	K/A #	293008 K1.04
	Importance Rating	2.7

Thermal-Hydraulics**Describe means by which boiling improves convection heat transfer**

Question #75

Which one of the following describes the effect of nucleate boiling on the heat transfer from the fuel rods to the Reactor coolant?

Nucleate boiling occurring at the surface of the fuel rods ____ (1) ____ the convective heat transfer from the fuel rod to the Reactor coolant because ____ (2) ____.

- | | (1) | (2) |
|----|--------|--|
| A. | raises | the motion of the steam bubbles
aids with mixing of the coolant |
| B. | raises | steam has a greater thermal
conductivity than water |
| C. | lowers | the motion of the steam bubbles
interferes with mixing of the coolant |
| D. | lowers | steam has a smaller thermal
conductivity than water |

Answer: A

Explanation: Nucleate boiling raises convective heat transfer because the motion of the steam bubbles aids in mixing. This carries the heat generated in the fuel into the bulk coolant more effectively than if nucleate boiling was not present.

- B. Incorrect – Steam does not have a greater thermal conductivity than water. Plausible because there is a difference in thermal conductivity between steam and water, and this drives a difference in conductive heat transfer between the two fluids.
- C. Incorrect – Nucleate boiling raises convective heat transfer. Plausible because other forms of boiling, such as transition boiling, lowers heat transfer.
- D. Incorrect – Nucleate boiling raises convective heat transfer. Plausible because other forms of boiling, such as transition boiling, lowers heat transfer.

Technical Reference(s): BT08lr4_Thermal_Hydraulics

References to be provided to applicants during examination: None

Learning Objective: BT08lr4_Thermal_Hydraulics 3

Question Source: Bank – NRC GFES Bank B486 & B1086

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(14)

Level of Difficulty (LOD): 3

Comments:

TRH 11/19/23 – Re-classified as Bank question based on NRC comment.

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	700000 2.4.30
	Importance Rating	4.1

Generator Voltage and Electric Grid Disturbances

Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator

Question #76

The plant is operating at 100% power with the following:

- Grid instabilities are occurring.
- 05-1-02-I-8, Loss of BOP AC Power / Grid Instability ONEP, is entered.
- The Main Generator voltage regulator is transferred from AUTO to MAN per SOI 04-1-01-N40-1, Main Generator and Auxiliaries, due to oscillations while in AUTO.
- Main Generator reactive load is adjusted to the maximum allowable.
- 500 KV System voltage is stabilized at 495 KV.

Which one of the following is the most restrictive time requirement, if any, for notifying the Transmission Control Center (TCC) and/or Systems Operations Center (SOC) based on these conditions, in accordance with plant procedures?

Notification is...

- A. NOT required
- B. required within a maximum of 30 minutes
- C. required within a maximum of 1 hour
- D. required within a maximum of 4 hours

Answer: B

Explanation: Notification to TCC is required because the Main Generator voltage regulator is transferred from AUTO to MAN and 500 KV System voltage is not within the required band of 505-515 KV. The notification must be made within a maximum of 30 minutes.

- A. Incorrect – Notification to TCC is required because the Main Generator voltage regulator is transferred from AUTO to MAN and 500 KV System voltage is not within the required band of 505-515 KV. Plausible because 500 KV System voltage has been stabilized at near the nominal value.
- C. Incorrect – The notification must be made within a maximum of 30 minutes. Plausible because 1 hour is a common notification time limit, such as in the emergency plan.
- D. Incorrect – The notification must be made within a maximum of 30 minutes. Plausible because 4 hours is a common notification time limit, such as in 10 CFR 50.72.

Technical Reference(s): 05-1-02-I-8, 04-1-01-N40-1

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-ONEP OBJ. 7

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(5)

Level of Difficulty (LOD): 3

Comments:

TRH 12/15/23 – Resampled generic K/A and replaced question based on NRC comment.

TRH 1/9/24 – Rearranged answer choices based on NRC comment.

Examination Outline Cross Reference	Level	RO
700000 (APE 25) Generator Voltage and Electric Grid Disturbances / 6	Tier	1
	Group #	1
AK3. Knowledge of the reasons for the following responses as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: (CFR: 41.4, 41.5, 41.7, 41.10 / 45.8)	K/A	700000
	Rating	3.6
AK3.02 Actions contained in abnormal operating procedure for voltage and grid disturbances	Revision	1
Revision Statement:		
NRC Submittal comments:		
Rev 1: Changed Question Source to NRC 12/2017 Q# 20 as bank.		

NRC 12/2017 Q# 20

Question: 12

Loss of AC Power ONEP states for grid instability, "Maintain Main Generator MVARs between +297 and -253 MVARs."

What is the reason for this step?

- I. Prevent a reverse power condition
- J. Prevent Generator stator bars from overheating
- K. Ensure ESF bus voltages remain within Tech Spec limits
- L. Ensure Generator output voltage is maintained at 22.5 KV

Answer: B
Explanation:
Per ONEP, section 3.4.3 NOTE: Maintaining Generator MVARs within these limits <u>Should</u> prevent Generator stator bars from overheating <u>AND</u> ensure the reverse power relay recognizes a reverse power condition.
Distracters:
A is wrong, but plausible. Per the above explanation, this will ensure the reverse power relay recognizes a reverse power condition not prevent one.
C is wrong, but plausible. Maintaining MVARs on the Generator has no control of Grid parameters.

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295025 EA2.01
	Importance Rating	4.3

High Reactor Pressure

Ability to determine and/or interpret the following as they apply to High Reactor Pressure: Reactor pressure

Question #77

The plant has experienced a failure to scram with the following:

- Reactor power is 40%.
- Reactor pressure is 1045 psig and steady.
- Two SRVs are currently open.
- RHR 'A' and RHR 'B' systems have been placed in Suppression Pool Cooling.
- Suppression pool level is 18' 5" and steady.
- Suppression pool temperature is 175°F and rising.

Which of the following describes the required control of Reactor pressure, in accordance with EP-2A, ATWS RPV Control?

- A. Immediately Emergency Depressurize the Reactor
- B. Lower Reactor pressure to attempt to prevent entering unsafe zone of HCTL
- C. Anticipate an Emergency Depressurization and fully open the Main Turbine Bypass valves
- D. Maintain Reactor pressure unless HCTL is exceeded, then Emergency Depressurize the Reactor

Answer: B

Explanation: Reactor power exceeds the capacity of the Turbine Bypass Valves, therefore SRVs will continue to open and Suppression pool temperature will continue to rise. HCTL is being approached, but is not yet violated. In the pressure branch of EP-2A, it states "If SP temp cannot be maintained in the Safe Zone of the HCTL THEN, maintain RPV pressure in the Safe Zone of the HCTL."

- A. Incorrect – An Emergency Depressurization is not yet required because HCTL has not been violated. Plausible because conditions are challenging HCTL.
- C. Incorrect – Anticipating Emergency Depressurization is not allowed while in EP-2A. Plausible because anticipating Emergency Depressurization is appropriate in other conditions where limits are being approached with Emergency Depressurization would be required.
- D. Incorrect – Reactor pressure must be lowered to attempt to prevent entering the unsafe zone of HCTL. Plausible because Reactor depressurization during a high powered ATWS is avoided if possible to prevent positive reactivity addition.

Technical Reference(s): EP-2A

References to be provided to applicants during examination: Heat Capacity Temperature Limit (HCTL)

Learning Objective: GLP-OPS-EP2A Obj 9

Question Source: Bank – 2-2020 NRC #78

Question History: 2-2020 NRC #78

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Level of Difficulty (LOD): 3

Comments:

TRH 11/20/23 – Moved HCTL from being an embedded graphic to being a standalone provided reference based on NRC comment.

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295016 AA2.03
	Importance Rating	4.2

Control Room Abandonment

Ability to determine and/or interpret the following as they apply to Control Room Abandonment: Reactor pressure

Question #78

The plant is operating at 100% power with the following:

- A fire in the Control room requires an evacuation.
- 05-1-02-II-1, Shutdown from the Remote Shutdown Panel ONEP, is being performed.
- Control has been established at the Remote Shutdown Panel.
- Reactor pressure is 1000 psig and slowly rising.

Which one of the following describes the required control of Reactor pressure/temperature, in accordance with 05-1-02-II-1?

- A. Stabilize Reactor pressure below 1060 psig, but do NOT commence a cooldown.
- B. Commence a cooldown. The cooldown rate is limited to no more than 20°F/hr.
- C. Commence a cooldown. The cooldown rate is limited to no more than 90°F/hr.
- D. Rapidly depressurize the Reactor irrespective of cooldown rate.

Answer: C

Explanation: Given evacuation due to a fire in the Control Room, 05-1-02-II-1 requires initiating a Reactor depressurization using SRVs within 23 minutes. A cooldown rate of 90°F/hr applies.

- A. Incorrect – Depressurization/cooldown is required. Plausible that the Reactor would be maintained in a stable hot shutdown condition while operating from the Remote Shutdown Panel due to limited equipment availability/control or to minimize the transient.
- B. Incorrect – The cooldown rate limit is 90°F/hr. Plausible because this is a limit given in the PTLR for other conditions.
- D. Incorrect – A cooldown rate of 90°F/hr applies. Plausible because this type of rapid depressurization does apply in numerous other emergency situations and would allow for injection with LPCI quicker.

Technical Reference(s): 05-1-02-II-1

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-ONEP Obj 6

Question Source: Bank - JAF 17-2 NRC #77

Question History: JAF 17-2 NRC #77

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(5)

Level of Difficulty (LOD): 3

Comments:

TRH 12/15/23 – Revised wording in B and C to “is limited to no more than” based on NRC comment.

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295003 AA2.03
	Importance Rating	3.9

Partial or Complete Loss of AC Power**Ability to determine and/or interpret the following as they apply to Partial or Complete Loss of AC Power: Battery status**

Question #79

The plant is operating at 50% power with the following:

- Bus 17AC was de-energized for 2 days due to an electrical fault.
- Bus 17AC is now re-energized.
- Battery Charger 1C4 has NOT yet been re-energized.
- Division 3 Battery 1C3 pilot cell voltage is now 2.11 VDC.

Which one of the following describes the OPERABILITY of Division 3 Battery 1C3?

Battery 1C3...

- A. must be declared INOPERABLE
- B. may be considered OPERABLE for 24 hours only if average voltage for each connected cell is a minimum of 2.07 VDC
- C. may be considered OPERABLE for 24 hours only if average voltage for each connected cell is a minimum of 2.13 VDC
- D. may be considered OPERABLE for 31 days for stated conditions if charger 1C5 is in service

Answer: A

Explanation: Technical Specification 3.8.4 requires the Division 3 Battery to have a charger connected to consider the Division 3 electrical power subsystem to be operable. With Battery Charger 1C4 not in-service, Technical Specification 3.8.4 Condition A must be entered. The Required Action requires verifying battery cell parameters meet Table 3.8.6-1 Category A limits within 1 hour. With pilot cell voltage <2.13 V, the battery does not meet the associated Category A limit in Table 3.8.6-1. Therefore, Technical Specification 3.8.4 Condition B must be entered, which requires declaring the battery inoperable immediately.

- B. Incorrect – The battery must be declared inoperable. Plausible because this is based on Technical Specification 3.8.6 Required Action A.2 and Table 3.8.6-1 Category C.
- C. Incorrect – The battery must be declared inoperable. Plausible because this is based on Technical Specification 3.8.6 Required Action A.2 and Table 3.8.6-1 Category B.
- D. Incorrect – The battery must be declared inoperable. Plausible because charger 1C5 is a related battery charger, and this would be correct if battery charger 1C4 was in service.

Technical Reference(s): Technical Specifications 3.8.4 and 3.8.6 and associated bases

References to be provided to applicants during examination: Technical Specifications 3.8.4 and 3.8.6

Learning Objective: GLP-OPS-TS001, OBJ. 40

Question Source: Bank - 2023 LOR biennial exam (Id0326)

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Level of Difficulty (LOD): 4

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295026 2.4.23
	Importance Rating	4.4

Suppression Pool High Water Temperature**Knowledge of the bases for prioritizing emergency operating procedures implementation****Question #80**

The plant has experienced a spurious opening of an SRV with the following:

- The Reactor has been manually scrammed.
- The SRV has been closed.
- EP-3, Primary Containment Control, has been entered due to high Suppression Pool temperature and level, only.

Which one of the following describes how to implement EP-3?

- A. All legs are entered. There IS a specified order for which leg to execute first.
- B. All legs are entered. There IS NOT a specified order for which leg to execute first.
- C. Only the SP/T and SP/L legs are entered. There IS a specified order for which leg to execute first.
- D. Only the SP/T and SP/L legs are entered. There IS NOT a specified order for which leg to execute first.

Answer: B

Explanation: When an EP-3 entry condition is met, the procedure is entered and all legs are entered concurrently. There is no specified prioritization between the legs. Prioritization is left up to the SRO based on the specifics of the event in progress.

- A. Incorrect – There is no specified order between legs. Plausible that a pre-determined order would exist to ensure consistent implementation.
- C. Incorrect – All legs are entered concurrently. Plausible because not all legs have an associated entry condition met. There is no specified order between legs. Plausible that a pre-determined order would exist to ensure consistent implementation.
- D. Incorrect – All legs are entered concurrently. Plausible because not all legs have an associated entry condition met.

Technical Reference(s): EP-3

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-EP3_EMERGENCY PROCEDURE - EP-3 Obj 2

Question Source: Bank – NMP1 2020 NRC #99

Question History: NMP1 2020 NRC #99

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(5)

Level of Difficulty (LOD): 2

Comments:

TRH 11/20/23 – Capitalized IS and IS NOT in 2nd part of each answer choice based on NRC comment.

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295006 AA2.05
	Importance Rating	4.5

Scram

Ability to determine and/or interpret the following as they apply to SCRAM: Whether a reactor SCRAM has occurred

Question #81

The plant has been scrammed with the following:

- APRMs indicate downscale.
- IRMs are fully inserted and indicate downscale on Range 1.
- SRMs are fully inserted and trending down with a negative period.
- A cluster of ten (10) adjacent control rods remain withdrawn at position 04.
- All other control rods are fully inserted.
- EP-2, RPV Control, has been entered due to low Reactor water level.
- NO steps have yet been performed in the Emergency Procedures (EPs).
- Reactor pressure is 940 psig and steady.

Which one of the following describes the required procedure execution and control of Reactor pressure, in accordance with the EPs?

	<u>Required Procedure Execution</u>	<u>Control of Reactor Pressure</u>
A.	Remain in EP-2	Depressurization is allowed
B.	Remain in EP-2	Depressurization is NOT allowed
C.	Exit EP-2 and enter EP-2A, ATWS RPV Control	Depressurization is allowed
D.	Exit EP-2 and enter EP-2A, ATWS RPV Control	Depressurization is NOT allowed

Answer: C

Explanation: EP-2 must be exited and EP-2A must be entered because it cannot yet be determined that the Reactor will remain shutdown under all conditions without boron based on the TSG 3.10 inset in EP-2. This determination is made based on control rod positions, not Reactor power indications. However, the given Reactor power indications do show that the Reactor is currently subcritical. With the Reactor subcritical and no boron injected, the Reactor may be depressurized while in EP-2A.

- A. Incorrect – EP-2 must be exited and EP-2A must be entered. Plausible because Reactor power indications do show that the Reactor is shutdown under current conditions.
- B. Incorrect – EP-2 must be exited and EP-2A must be entered. Plausible because Reactor power indications do show that the Reactor is shutdown under current conditions. A Reactor depressurization is allowed. Plausible because Reactor depressurization/cooldown has the potential to affect Reactor power and is limited under some failure to scram conditions.
- D. Incorrect – A Reactor depressurization is allowed. Plausible because Reactor depressurization/cooldown has the potential to affect Reactor power and is limited under some failure to scram conditions.

Technical Reference(s): EP-2, EP-2A

References to be provided to applicants during examination: None

Learning Objective GLP-OPS-EP02 Obj 4.3

Question Source: Bank – Susquehanna 2015 NRC #76

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Level of Difficulty (LOD): 3

Comments:

TRH 11/20/23 – Added that the 10 control rods are clustered and adjacent based on NRC comment.

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295019 2.4.45
	Importance Rating	4.3

Partial or Complete Loss of Instrument Air**Ability to prioritize and interpret the significance of each annunciator or alarm**

Question #82

The plant is operating at 100% power with the following:

- Plant Air Compressor 'A' is running.
- Plant Air Compressor 'B' is running.
- Plant Air Compressor 'C' is in standby.
- An RO reports the following alarms and indication:
 - P870-7A-F3, PLANT AIR TROUBLE
 - P870-7A-D3, PLANT INSTR AIR HEADER PRESS HI/LO
 - P854-1A-C4, PLANT AIR HEADER PRESS HI/LO
- Instrument Air Header pressure on P870 indicates 100 psig
- The crew enters 05-1-02-V-9, Loss of Instrument Air ONEP, Section B, Instrument Air Pressure Lowering.
- An air leak is reported from the field.

Which one of the following describes:

(1) the action the CRS should direct the RO to perform

and

(2) the threshold for Instrument Air Header Pressure that would require the CRS to enter section 'A' of the ONEP, Complete Loss of Instrument Air?

- A. (1) Ensure all Hiller Air Operated Valves close
(2) 60 psig
- B. (1) Ensure all Hiller Air Operated Valves close
(2) 80 psig
- C. (1) Ensure Standby Air Compressor auto starts
(2) 60 psig
- D. (1) Ensure Standby Air Compressor auto starts
(2) 80 psig

Answer: D

Explanation: Per listed ARI, P870-7A-D3, PLANT INSTR AIR HEADER PRESS HI/LO,
POSSIBLE CAUSES: Plant Air receiver pressure < 90 psig

AUTOMATIC ACTIONS: Plant Air Compressor A (C) (B) auto starts (IF in Standby) IF Plant Air Median pressure drops to <115 psig for 5 seconds. If this alarm is received, then receiver pressure is <115 psig. Therefore, auto actions should be to ensure standby air compressor auto starts.

Per ONEP, "A Complete Loss of Instrument Air is defined as Instrument Air Header Pressure CANNOT be MAINTAINED GREATER THAN 80 psig."

- A. Incorrect – Hiller Air Operated valves should not be closed yet. Plausible because they do close if Instrument Air pressure lowers further. The threshold is 80 psig. Plausible because 60 psig is called out in the ONEP as the pressure at which Hiller Air Operated valves closed.
- B. Incorrect – Hiller Air Operated valves should not be closed yet. Plausible because they do close if Instrument Air pressure lowers further.
- C. Incorrect – The threshold is 80 psig. Plausible because 60 psig is called out in the ONEP as the pressure at which Hiller Air Operated valves closed.

Technical Reference(s): 05-1-02-V-9, 04-1-02-1H13-P870-7A-D3

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-ONEP Obj 4

Question Source: Bank – 2022 NRC #79

Question History: 2022 NRC #79

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Level of Difficulty (LOD): 3

Comments:

TRH 11/17/23 – Revised to have 2 PACs initially running based on normal plant lineup and NRC comment on Question #2.

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	2
	K/A #	295020 2.4.51
	Importance Rating	4.0

Inadvertent Containment Isolation

Knowledge of emergency operating procedure exit conditions (e.g., emergency condition no longer exists or severe accident guideline entry is required)

Question #83

The plant is operating at 100% power with the following:

- A spurious isolation signal causes all MSIVs to close.
- The Reactor scrams.
- All control rods insert.
- EP-2, RPV Control, is entered.
- Reactor water level is 35" and steady with Feedwater controlling.
- Reactor pressure is 900 psig and slowly lowering with one SRV open.
- Reactor pressure is being lowered to a band of 450-600 psig.

Which one of the following describes when EP-2 can be exited?

EP-2 can be exited...

- A. at this time
- B. only once the plant is brought to Mode 4 or 5
- C. only once Reactor pressure is in the band of 450-600 psig
- D. only once the cause of the spurious isolation signal is identified and corrected

Answer: A

Explanation: Currently, there are no EP-2 entry conditions met or exceeded. The absence of entry conditions alone does not require/allow EP exit. EP-2 can be exited as soon as it is determined an emergency no longer exists. The given conditions show that the plant is in a controlled state. Therefore, EP-2 can be exited at this time.

- B. Incorrect – EP-2 can be exited at this time. Plausible because the version of EP-2 that is currently being used is only applicable in Modes 1-3 and EP-2 does contain guidance that cools the plant down towards Mode 4.
- C. Incorrect – EP-2 can be exited at this time. Plausible because Reactor pressure is currently being lowered (is not yet stabilized in the ordered band) and this is a Reactor pressure band describes in 02-S-01-43, Transient Mitigation Strategy.
- D. Incorrect – EP-2 can be exited at this time. Plausible because this is the initial cause of the transient that led to the need for EP entry.

Technical Reference(s): EP-2

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-EP02 Obj 6

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Level of Difficulty (LOD): 3

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	2
	K/A #	295002 AA2.07
	Importance Rating	3.4

Loss of Main Condenser Vacuum

Ability to determine and/or interpret the following as they apply to Loss of Main Condenser Vacuum: Turbine limitations

Question #84

The plant is operating at 100% power with the following:

- A loss of Main Condenser vacuum occurs.
- 05-1-02-V-8, Loss of Condenser Vacuum ONEP, is being executed.
- A manual Reactor scram is inserted.
- Main Condenser vacuum is 19" Hgv and lowering slowly.
- Reactor pressure is 935 psig and steady on Turbine Bypass Valves.
- Reactor water level is 35" and stable.
- Feedwater pumps are injecting.

Which one of the following describes:

- the current operation of the Feedwater pumps
and
- the associated procedural direction regarding the Feedwater pump low vacuum trip,

in accordance with plant procedures?

	<u>Feedwater pumps have...</u>	<u>Associated Procedural Direction</u>
A.	failed to trip on low vacuum	Feedwater pumps must be tripped now per the RFP TRIP ARIs
B.	failed to trip on low vacuum	Feedwater pumps may be left in service and low vacuum trips bypassed per 05-1-02-V-8
C.	NOT reached their low vacuum trip yet	Low vacuum trips may be bypassed now per 05-1-02-V-8
D.	NOT reached their low vacuum trip yet	If Feedwater pumps trip on low vacuum and vacuum is above the MSIV low vacuum trip setpoint, then low vacuum trips may be bypassed per RFP TRIP ARIs or 04- 1-01-N21-1, Feedwater System SOI

Answer: D

Explanation: Main Condenser vacuum remains above the Feedwater pump trip setpoint of 16" Hgv, therefore the Feedwater pumps have not yet reached their low vacuum trip. If Main Condenser vacuum lowers to 16" Hgv, the associated Alarm Response Instructions for RFP Trip provide an allowance to bypass the trip to restore injection using Feedwater pumps, as long as vacuum remains above the MSIV low vacuum trip setpoint.

- A. Incorrect – Feedwater pumps have not yet reached their low vacuum trip. Plausible because this would be correct if vacuum was <16" Hgv.
- B. Incorrect – Feedwater pumps have not yet reached their low vacuum trip. Plausible because this would be correct if vacuum was <16" Hgv.
- C. Incorrect – If Feedwater pumps trip on low vacuum, the low vacuum trips may be bypassed and this guidance is found in the associated RFP TRIP ARIs. There is no allowance in 05-1-02-V-8 to bypass the trips now (pre-emptively). Plausible that action would be taken ahead of time to avoid temporary loss of the Feedwater pumps.

Technical Reference(s): 05-1-02-V-8, ARI P680-2A-A2, 04-1-01-N21-1 Feedwater System Att VI

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-N2100 Obj 12&15

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Level of Difficulty (LOD): 4

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	2
	K/A #	295029 EA2.03
	Importance Rating	3.6

High Suppression Pool Water Level

Ability to determine and/or interpret the following as they apply to High Suppression Pool Water Level: Drywell/containment water level

Question #85

The plant has experienced a loss of coolant accident with the following:

- All Suppression Pool and Containment water level instruments have malfunctioned.
- The last observed Suppression Pool water level was 23' and slowly rising.
- Reactor pressure is 700 psig and slowly lowering.
- Reactor water level is 30" and stable.
- Containment pressure is 10 psig and slowly rising.

Which one of the following describes the ability to determine Containment water level?

Containment water level...

- A. can be determined by directing use of a Field Support Guideline (FSG)
- B. can be determined by directing use of an attachment in EP-1, Emergency/Severe Accident Procedure Support Documents
- C. CANNOT be determined, therefore direct terminating all RPV injection from external sources
- D. CANNOT be determined, therefore direct performing an Emergency Depressurization

Answer: B

Explanation: With a loss of Suppression Pool and Containment water level indications, Containment water level can still be determined using the difference in RCIC suction pressure (when aligned to Suppression Pool) and Containment pressure. The guidance for this determination is found in EP-1 attachment 29.

- A. Incorrect – The guidance for this determination is found in EP-1 attachment 29. Plausible because the FSGs are another type of procedure that provide alternate guidance while executing the EPs.
- C. Incorrect – Containment water level can be determined using EP-1 attachment 29. Plausible because all installed instrumentation has malfunctioned. Also plausible because terminating all RPV injection from external sources is an action related to high Suppression Pool water level.
- D. Incorrect – Containment water level can be determined using EP-1 attachment 29. Plausible because all installed instrumentation has malfunctioned. Also plausible because performing an Emergency Depressurization is an action related to high Suppression Pool water level.

Technical Reference(s): EP-1, EP-2, EP-3

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-EP01 Obj 7

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(5)

Level of Difficulty (LOD): 4

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	259002 A2.03
	Importance Rating	3.8

Reactor Water Level Control

Ability to (a) predict the impacts of the following on the Reactor Water Level Control System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Loss of reactor water level input

Question #86

The plant is operating at 100% power with the following:

- Annunciator P680-2A-C9, DFCS TROUBLE, alarms.
- Annunciator P680-4A2-A2, RX WTR LVL SIG FAIL HI/LO, alarms.
- Investigation reveals the following:
 - Narrow range Reactor water level instrument 1C34-N004A has failed downscale.
 - The associated trip unit, 1C34-K624A, REACTOR VESSEL LEVEL A (LVL 9 MT/RFPT TRIP), indicates a hard failure.

Which one of the following identifies the earliest Required Action completion time, if any, in accordance with Technical Specifications (TS) and/or the Technical Requirements Manual (TRM), based on this failure?

- A. Restore the instrument within a maximum of 24 hours
- B. Restore the instrument within a maximum of 72 hours
- C. Restore the instrument within a maximum of 7 days
- D. NO TS or TRM condition entry is required

Answer: C

Explanation: Narrow range Reactor water level instrument 1C34-N004A provides a signal to Digital Feedwater Level Control for control of Reactor water level and to initiate a high level trip of the Main Turbine and Feedwater pumps. When the instrument fails downscale, it is automatically removed from the Reactor water level control algorithm and Digital Feedwater Level Control continues to control Reactor water level with the remaining two level inputs. However, with the instrument failed downscale, it is not functional for the high level trip. TRM 6.3.7 requires all 3 Reactor water level instruments to be functional for this trip. Therefore, TRM 6.3.7 Condition A must be entered (7 days). The remaining 2 Reactor water level instruments maintain trip capability, therefore Condition B entry is not required (72 hours).

- A. Incorrect – 7 days are allowed. Plausible because 24 hours is a time limitation in TS 3.3.5.2 and would apply for other failed Reactor water level instruments.
- B. Incorrect – 7 days are allowed. Plausible because this would be correct if trip capability was not maintained with this instrument non-functional (Condition B).
- D. Incorrect – TRM 6.3.7 Condition A entry is required. Plausible because entry would not be required for an upscale failure, since the instrument would be providing the required trip signal. Also plausible because some instrumentation specifications require less than all of the associated instruments to be operable/functional (eg. 4 instruments, only 3 required).

Technical Reference(s): 04-1-01-B21 Attachment V, ARIs P680-2A-C9 and P680-4A2-A2, Technical Specification 3.3.5.2, TRM 6.3.7

References to be provided to applicants during examination: Technical Specification 3.3.5.2, TRM 6.3.7

Learning Objective: GLP-OPS-TS001 TECHNICAL SPECIFICATIONS Obj 40

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Level of Difficulty (LOD): 3

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	218000 2.4.12
	Importance Rating	4.3

Automatic Depressurization**Knowledge of operating crew responsibilities during emergency and abnormal operations**

Question #87

A plant has experienced a loss of coolant accident with the following:

- Reactor water level is -155" and steady.
- Reactor pressure is 450 psig and lowering slowly.
- Drywell pressure is 10 psig and rising slowly.
- All RHR pumps have failed to start.
- HPCS has failed to start.
- LPCS is operating properly.
- ADS has NOT been inhibited.

Which one of the following describes:

- (1) the response of the ADS valves if the current conditions continue for two (2) minutes, and
- (2) the required control of ADS under these conditions, in accordance with EP-2, RPV Control?

	(1) If current conditions continue, ADS valves will...	(2) EP-2 requires...
A.	open	allowing ADS valves to open
B.	open	inhibiting ADS
C.	remain closed	manually opening ADS valves
D.	remain closed	inhibiting ADS

Answer: B

Explanation: Reactor water level is below the ADS actuation setpoint of -150.3" and Drywell pressure is above 1.39 psig. Multiple injection systems have failed to operate properly, including RHR A, which inputs into the ADS logic. However, LPCS running alone is enough to satisfy the ADS logic. Therefore, ADS valve will open after 105 seconds. EP-2 requires inhibiting ADS in this situation. ADS valves are only manually opened if Reactor water level continues to lower to below -166".

- A. Incorrect – EP-2 requires inhibiting ADS in this situation. Plausible because ADS is designed to open valves in this situation and ADS valves are manually opened if Reactor water level continues to lower to below -166".
- C. Incorrect – ADS will open valves after 105 seconds with Reactor water level <17.8" and LPCS running. Plausible because the RHR failure does prevent part of the ADS logic from being satisfied. EP-2 requires inhibiting ADS in this situation. Plausible because ADS is designed to open valves in this situation and ADS valves are manually opened if Reactor water level continues to lower to below -166".
- D. Incorrect – ADS will open valves after 105 seconds with Reactor water level <-150.3", Drywell pressure above 1.39 psig, and LPCS running. Plausible because the RHR failure does prevent part of the ADS logic from being satisfied.

Technical Reference(s): ARI P601-18A-B3(B4)(C2)(E2), EP-2

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-EP02_003-1 Obj 7.1

Question Source: Bank – NMP2 2019 NRC #88

Question History: NMP2 2019 NRC #88

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Level of Difficulty (LOD): 3

Comments:

TRH 1/9/24 – Rearranged answer choices based on NRC comment.

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	262002 A2.05
	Importance Rating	3.9

Uninterruptable Power Supply (AC/DC)

Ability to (a) predict the impacts of the following on the Uninterruptable Power Supply (AC/DC) and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Loss of UPS

Question #88

The plant is operating at 100% power with the following:

- Inverter 1Y98 DC Feeder breaker has tripped.
- P807-3A-G3, STATIC INVRTR 1Y98 TROUBLE, alarm is in.
- The inverter is indicating no voltage.

Which one of the following describes:

- the impact on the plant

AND

- the required procedure to use to restore the inverter?

	Impact on the Plant	Required Procedure
A.	Div 2 Half Scram, Div 2 Auxiliary Building Isolation	ONEP 05-1-02-V-19, Loss of 125 VDC
B.	Div 2 Half Scram, Div 2 Auxiliary Building Isolation	SOI 04-1-01-L62-1, Static Inverters System
C.	RC&IS INOP	ONEP 05-1-02-V-19, Loss of 125 VDC
D.	RC&IS INOP	SOI 04-1-01-L62-1, Static Inverters System

Answer: D

Explanation: A loss of power to the 1Y98 Inverter will cause the following:

Lose Rod Control & Information System:

- 1) RGDS power (P653)
- 2) BJM power
- 3) Accumulator fault data
- 4) Rod position data on P680 (still available at RACS panels)

Lose various BOP computer multiplexers

- 1) Lose FPCCU indications and recorder at P001 (Area 9, 166')
- 2) FPCCU Makeup Valves G41-F002 and G41-F047 close

Lose MS/R A & B control cabinet power

Lose various plant radio repeaters

Per ARI for P807-3A-G3:

3.2 Refer to SOI 04-1-01-L62-1 and manually transfer to alt. pwr. supply, if AC output voltage, current or frequency on respective inverter indicates inverter failure or erratic inverter operation, and the static switch has not already transferred power.

- A. Incorrect – These actions would occur if 1Y88 inverter was to trip. Plausible due to the similar numbers. Even though the breaker that tripped was a DC breaker, it was solely for the 1Y98 inverter. This does not meet the criteria for entering the Loss of DC ONEP. Plausible due to a DC breaker tripped.
- B. Incorrect – These actions would occur if 1Y88 inverter was to trip. Plausible due to the similar numbers
- C. Incorrect – Even though the breaker that tripped was a DC breaker, it was solely for the 1Y98 inverter. This does not meet the criteria for entering the Loss of DC ONEP. Plausible due to a DC breaker tripped.

Technical Reference(s): 04-1-02-1H13-P807-3A-G3, 04-1-01-L62-1, ONEP 05-1-02-V-19

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-L6200, Rev 19, Obj 3.8.1

Question Source: Bank - 10-2020 NRC #87

Question History: 10-2020 NRC #87

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Level of Difficulty (LOD): 3

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	215005 2.4.6
	Importance Rating	4.7

Average Power Range Monitor/Local Power Range Monitor**Knowledge of emergency and abnormal operating procedures major action categories**

Question #89

The plant has experienced a failure to scram with the following:

- APRMs indicate 9% and steady.
- Reactor water level is -20" and steady.
- The level leg of EP-2A, ATWS RPV Control, is being executed.

Which one of the following describes the required control of Reactor water level, in accordance with EP-2A?

Reactor water level...

- A. may be restored to a band of 11.4" to 53.5"
- B. must be stabilized at or near the current value
- C. must be intentionally lowered to at least -70"
- D. must be intentionally lowered to at least -166"

Answer: C

Explanation: The level leg of EP-2A uses APRM indication to determine the required control of Reactor water level. With APRMs indicating above 5% and Reactor water level above -70", EP-2A requires intentionally lowering Reactor water level by terminating and preventing all injection except SLC, CRD, and RCIC. The procedure requires lowering level to at least -70".

- A. Incorrect – Reactor water level must be lowered. Plausible because this would be correct if APRMs indicated $\leq 5\%$.
- B. Incorrect – Reactor water level must be lowered. Plausible because Reactor power is relatively low and it is currently stable. Maintaining Reactor water level at or near the current value would prevent inducing more transient while efforts are made to insert control rods.
- D. Incorrect – The minimum threshold for lowering Reactor water level is -70". Plausible because -166" is the lower end of a water level band in EP-2A and is used as a threshold Reactor water level for other decisions related to APRM power level.

Technical Reference(s): EP-2A

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-EP2A Obj 6

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Level of Difficulty (LOD): 3

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	211000 A2.01
	Importance Rating	3.8

Standby Liquid Control

Ability to (a) predict the impacts of the following on the Standby Liquid Control System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Pump trip

Question #90

Standby Liquid Control (SLC) pump 'B' trips during surveillance testing.

An ATWS occurs with Reactor power at 15%.

The RO performs immediate operator actions.

Electrical bus 15AA has lost power and indicates a fault on the bus.

Which one of the following describes:

- the availability of a SLC pump
and
- the required procedure to execute in response to these conditions?

	SLC Pump	Required Procedure Execution
A.	NONE available	EP-1 Attachment 28, Alternate SLC Injection
B.	NONE available	04-1-01-C41-1, Standby Liquid Control System
C.	'A' is available	EP-1 Attachment 28, Alternate SLC Injection
D.	'A' is available	04-1-01-C41-1, Standby Liquid Control System

Answer: A

Explanation: With a loss of 15AA, power is lost to SLC pump 'A' therefore no SLC pumps are available due to 'B' pump already being tripped. In these conditions, the direction in 04-1-01-C41-1 for injecting SLC does not work. EP-2A calls for Attachment 28 when boron cannot be injected.

- B. Incorrect – With neither SLC pump available, the direction in 04-1-01-C41-1 for injecting SLC does not work. Plausible because this would be correct if SLC pump 'A' was available. Also plausible that this procedure would have alternate direction boron injection without an SLC pump.
- C. Incorrect – 'A' SLC pump is not available due to the loss of bus 15AA. Plausible that this pump would be powered by a different source.
- D. Incorrect – 'A' SLC pump is not available due to the loss of bus 15AA. Plausible that this pump would be powered by a different source. With neither SLC pump available, the direction in 04-1-01-C41-1 for injecting SLC does not work. Plausible because this would be correct if SLC pump 'A' was available. Also plausible that this procedure would have alternate direction boron injection without an SLC pump.

Technical Reference(s): 04-1-01-C41-1, 05-1-02-EP-1 Attachment 28

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-EP2A, OBJ. 9

Question Source: Bank – 2-2020 NRC #86

Question History: 2-2020 NRC #86

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Level of Difficulty (LOD): 3

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	2
	K/A #	271000 A2.06
	Importance Rating	3.3

Offgas

Ability to (a) predict the impacts of the following on the Offgas System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Offgas system holdup volume explosion / fire

Question #91

The plant is operating at 100% power with the following:

- A hydrogen explosion occurs in the Offgas system.
- 05-1-02-V-13, Hydrogen Explosion in Offgas ONEP, is entered.
- Annunciator P601-19A-E8, OG POST-TREAT RAD HI, alarms.
- Annunciator P601-19A-D8, OG POST-TREAT RAD HI-HI, alarms.
- Annunciator P601-19A-C8, OG POST-TREAT RAD HI-HI-HI/INOP, alarms.

Which one of the following describes the required action, in accordance with 05-1-02-V-13?

- A. Scram the Reactor.
- B. Reduce core flow to 70 Mlbm/hr. Further power reduction is NOT yet required.
- C. Initiate an air purge on the recombiner train. A power reduction is NOT yet required.
- D. Reduce core flow to 70 Mlbm/hr and then insert CRAM rods. A Reactor scram is NOT yet required.

Answer: A

Explanation: 05-1-02-V-13 directs monitoring Offgas radiation monitor indications. With Annunciator P601-19A-C8 in alarm, Offgas post-treatment radiation has risen to the auto-isolation setpoint. 05-1-02-V-13 requires a manual Reactor scram based on Offgas post-treatment radiation exceeding the auto-isolation setpoint.

Note: Avoided explicitly testing the first half of the K/A (such as in a 2-column format) to prevent double jeopardy with Question #61. Per NUREG 1021 ES-4.2 C.2, focused the question instead on the aspect of the K/A statement requiring the highest cognitive level, the (b) portion of the A.2 K/A statement. This portion of the K/A statement is also most applicable to the SRO job function. The first half is implicitly tested, since knowledge of the Offgas radiation monitor auto-isolation setpoint is needed to predict the impact of these conditions in order to determine the required action.

- B. Incorrect – A Reactor scram is required. Plausible because this would be correct if Offgas rad level were rising but still below auto-isolation setpoint.
- C. Incorrect – A Reactor scram is required. Plausible because this would be correct if Offgas rad levels were not yet an issue.
- D. Incorrect – A Reactor scram is required. Plausible because lowering core flow to 70Mlbm/hr would be correct if Offgas rad level were rising but still below auto-isolation setpoint. Also plausible because additionally inserting CRAM rods may be performed to further lower power based on Condenser vacuum response or need to complete plant shutdown.

Technical Reference(s): ARI P601-19A-C8, 05-1-02-V-13

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-ONEP Obj 6
GLP-OPS-N6465 Obj 13

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Level of Difficulty (LOD): 3

Comments:

TRH 11/20/23 – Added LOD rating based on NRC comment.

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	2
	K/A #	234000 K4.02
	Importance Rating	3.8

Fuel Handling Equipment

Knowledge of Fuel Handling System design features and/or interlocks that provide for the following: Prevention of control rod movement during core alterations

Question #92

A refueling outage is in progress with the following:

- The Reactor Mode Switch is in REFUEL.
- All control rods are fully inserted.
- The next fuel move will transfer a fuel bundle to the Reactor.

Which one of the following describes when a control rod block will **first** be received and the basis for this interlock, in accordance with Technical Specifications?

	<u>A control rod block will first be received when the Refuel Bridge is...</u>	<u>Basis for Interlock</u>
A.	about to move over the core	Prevent inadvertent criticality
B.	about to move over the core	Limit radiation levels on Refuel Bridge
C.	over the core and the main hoist begins to lower	Prevent inadvertent criticality
D.	over the core and the main hoist begins to lower	Limit radiation levels on Refuel Bridge

Answer: A

Explanation: At the beginning of this fuel move, the fuel bundle will be loaded onto the main hoist, with the main hoist in the full up position. A limit switch detects when the Refuel Bridge is about to move over the core. The combination of the main hoist load cell detecting the loaded fuel bundle and the limit switch detecting the Refuel Bridge about to move over the core initiates the control rod block. Technical Specification 3.9.1 bases state that the basis for this refueling interlock is to prevent inadvertent criticality.

- B. Incorrect – The basis for this refueling interlock is to prevent inadvertent criticality. Plausible because withdrawing a control rod would cause some additional rise in radiation levels on the Refuel Bridge while it is over the core.
- C. Incorrect – The control rod block is first received earlier, when the Refuel Bridge is about to move over the core. Plausible because as long as the fuel bundle is not lowered, a control rod could still be moved without having two simultaneous reactivity additions.
- D. Incorrect – The control rod block is first received earlier, when the Refuel Bridge is about to move over the core. Plausible because as long as the fuel bundle is not lowered, a control rod could still be moved without having two simultaneous reactivity additions. The basis for this refueling interlock is to prevent inadvertent criticality. Plausible because withdrawing a control rod would cause some additional rise in radiation levels on the Refuel Bridge while it is over the core.

Technical Reference(s): 04-1-01-F11-1, Technical Specification 3.9.1 bases

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-TS001_018-1 Obj 40

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(7)

Level of Difficulty (LOD): 3

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	2
	K/A #	288000 2.1.32
	Importance Rating	4.0

Plant Ventilation**Ability to explain and apply system precautions, limitations, notes, or cautions****Question #93**

The plant is shutdown with the following:

- A fire has occurred in the Turbine Building.
- The fire has been extinguished.
- It is now desired to remove smoke from the Turbine Building.
- Turbine Building Ventilation is currently secured and available for restart.

Which one of the following describes:

- the preferred method of removing smoke from the Turbine Building
and
- the associated reason,

in accordance with 04-1-01-U41-1, Turbine Building Ventilation System?

Preferentially remove smoke by using...

- A. normal Turbine Building Ventilation. This is based on preventing system or component damage.
- B. normal Turbine Building Ventilation. This is based on ability to monitor radioactive release to the environment.
- C. Turbine Building Smoke Removal fans. This is based on preventing system or component damage.
- D. Turbine Building Smoke Removal fans. This is based on ability to monitor radioactive release to the environment.

Answer: B

Explanation: 04-1-01-U41-1 Precaution and Limitation 3.1 and Notes in section 5.2, Smoke Removal, discuss use of Turbine Building Smoke Removal fans. If possible, normal Turbine Building Ventilation fans are to be used for smoke removal, not Turbine Building Smoke Removal fans. Use of Turbine Building Smoke Removal fans requires additional actions by Chemistry to determine the release to the environment.

- C. Incorrect – If possible, normal Turbine Building Ventilation fans are to be used for smoke removal, not Turbine Building Smoke Removal fans. Plausible because Turbine Building Smoke Removal fans are specifically designed for this purpose. This preference is based on ability to determine release to the environment. Plausible that the concern would be component/system damage due to presence of smoke/heat/etc. following the fire.
- D. Incorrect – If possible, normal Turbine Building Ventilation fans are to be used for smoke removal, not Turbine Building Smoke Removal fans. Plausible because Turbine Building Smoke Removal fans are specifically designed for this purpose.
- A. Incorrect – This preference is based on ability to determine release to the environment. Plausible that the concern would be component/system damage due to presence of smoke/heat/etc. following the fire.

Technical Reference(s): 04-1-01-U41-1

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-U4100 Obj 4.6

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(5)

Level of Difficulty (LOD): 4

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.4.16
	Importance Rating	4.4

Knowledge of emergency and abnormal operating procedures implementation hierarchy and coordination with other support procedures or guidelines such as, operating procedures, abnormal operating procedures, or severe accident management guidelines

Question #94

The plant has experienced a loss of coolant accident with the following:

- EP-2, RPV Control, and EP-3, Primary Containment Control, are being implemented.
- Emergency Depressurization has been performed due to low Reactor water level.
- LPCS is injecting 4000 gpm to the Reactor.
- NO other injection systems are available.
- Reactor pressure is 50 psig and stable.
- Reactor water level is -220" and stable.
- Drywell pressure is 11 psig and rising slowly.

Which one of the following describes the required procedure implementation?

- A. Remain in EP-2 and EP-3.
- B. Exit EP-2 and remain in EP-3.
Enter SAP-1, RPV Control. Entry into SAP-2, Containment and Radioactive Release Control, is NOT required.
- C. Exit EP-3 and remain in EP-2.
Enter SAP-2, Containment and Radioactive Release Control. Entry into SAP-1, RPV Control, is NOT required.
- D. Exit EP-2 and EP-3.
Enter SAP-1, RPV Control, and SAP-2, Containment and Radioactive Release Control.

Answer: D

Explanation: Since the Reactor has already been depressurized, injection sources have been maximized, LPCS flow is below 7000 gpm, no other injection exists, and Reactor water level cannot be raised above -191", EP-2 requires exiting all EPs and entering both SAPs.

- A. Incorrect – EP-2 requires exiting all EPs and entering both SAPs. Plausible because this would be correct if either LPCS flow were higher or Reactor water level was higher.
- B. Incorrect – EP-2 requires exiting all EPs and entering both SAPs. Plausible that SAPs would be entered individually, such as EPs.
- C. Incorrect – EP-2 requires exiting all EPs and entering both SAPs. Plausible that SAPs would be entered individually, such as EPs. Also plausible that the given conditions would satisfy adequate core cooling, but not Containment requirements.

Technical Reference(s): EP-2

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-EP3 Obj 6
GLP-OPS-EP2 Obj 6

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Level of Difficulty (LOD): 3

Comments:

TRH 11/20/23 – Revised layout of B and C to combine EPs into first line and SAPs into second line (except for wrapping based on length) based on NRC comment.

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.3.6
	Importance Rating	3.8

Ability to approve liquid or gaseous release permits

Question #95

Per 01-S-08-11, Radioactive Discharge Controls, who is required to sign the "Release Authorization" block of a Batch Liquid Radioactive Release Permit?

- A. Shift Manager, only
- B. Operations Manager, only
- C. Shift Manager and Chemistry Technician
- D. Operations Manager and Chemistry Technician

Answer: A

Explanation: 01-S-08-11 "Release Authorization", Part 4, signature is only the Shift Manager. The responsibilities sections states that the SM is responsible for all releases during his shift.

- B. Incorrect – Only the Shift Manager signs the "Release Authorization" block. Plausible because Operations Supervision, which would include the Operations Manager, signs other places on the Batch Liquid Radioactive Release Permit.
- C. Incorrect – Only the Shift Manager signs the "Release Authorization" block. Plausible because the Radiochemist signs other places on the Batch Liquid Radioactive Release Permit.
- D. Incorrect – Only the Shift Manager signs the "Release Authorization" block. Plausible because Operations Supervision, which would include the Operations Manager, signs other places on the Batch Liquid Radioactive Release Permit. Also plausible because the Radiochemist signs other places on the Batch Liquid Radioactive Release Permit.

Technical Reference(s): 01-S-08-11

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-PROC Objective 53.3 (SRO only)

Question Source: Bank – 5-2017 NRC #97

Question History: 5-2017 NRC #97

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(4)

Level of Difficulty (LOD): 4

Comments:

TRH 11/20/23 – Revised D to not have Shift Manager in three choices, based on NRC comment.

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.4.40
	Importance Rating	4.5

Knowledge of SRO responsibilities in emergency plan implementing procedures

Question #96

The plant has experienced an accident with the following:

- Conditions warrant emergency classification.
- You are assuming the role of the Emergency Director.
- The Emergency Response Organization has NOT yet been activated.

Which one of the following is a responsibility that you ARE allowed to delegate, in accordance with EN-EP-609, Emergency Operations Facility (EOF) Operation?

- A. Ordering local area/building evacuation
- B. Classifying and declaring the emergency
- C. Emergency notifications of the accident to off-site agencies
- D. Making Protective Action Recommendations (PARs) to off-site agencies

Answer: A

Explanation: EN-EP-609 lists the following responsibilities that the Emergency Director cannot delegate (non-delegable):

- CLASSIFY Emergency Conditions
- MAKE Protective Action Recommendations (PARs)
- Initial Notification of Emergency Classification and/or PARs to Offsite Agencies (States, Locals, NRC)

Of the given answer choices, only ordering local area/building evacuation is not on this list.

- B. Incorrect – This cannot be delegated. Plausible because this responsibility eventually gets transferred to another individual in the EOF.
- C. Incorrect – This cannot be delegated. Plausible because this responsibility eventually gets transferred to another individual in the EOF.
- D. Incorrect – This cannot be delegated. Plausible because this responsibility eventually gets transferred to another individual in the EOF.

Technical Reference(s): EN-EP-609, Emergency Operations Facility (EOF)
Operations Att 1

References to be provided to applicants during examination: None

Learning Objective: FCBT-EP-EROROLES Obj 1

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(1)

Level of Difficulty (LOD): 2

Comments:

TRH 11/20/23 – Replaced choice A based on NRC comment.

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.2.37
	Importance Rating	4.6

Ability to determine operability or availability of safety-related equipment

Question #97

The plant is operating at 100% power with the following:

- It is discovered that a surveillance has NOT been completed on time for a Technical Specification required system.
- The surveillance frequency is quarterly (92 days).
- The surveillance was last performed 120 days ago.
- A risk evaluation has been performed and the associated impact is being managed.
- It is expected that the system will pass the surveillance when it is performed.

Which one of the following describes the requirement for this missed surveillance, in accordance with Technical Specifications?

- A. The associated LCO must be declared NOT met at this time
- B. Complete the surveillance within a maximum of 24 hours from the time of discovery or then the associated LCO must be declared NOT met
- C. Complete the surveillance within a maximum of 23 days from the time of discovery or then the associated LCO must be declared NOT met
- D. Complete the surveillance within a maximum of 92 days from the time of discovery or then the associated LCO must be declared NOT met

Answer: D

Explanation: Surveillance Requirement (SR) 3.0.3 applies given discovery of a missed surveillance after the required frequency has elapsed. The requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified frequency (in this case, 92 days), whichever is greater. Therefore, up to 92 days are allowed to perform the missed surveillance before being required to declare the LCO not met.

- A. Incorrect – SR 3.0.3 allows a delay time to perform the missed surveillance before being required to declare the LCO not met. Plausible because the surveillance is currently over the 125% limit of SR 3.0.2.
- B. Incorrect – SR 3.0.3 allows the longer of 24 hours or the specified frequency (92 days). Plausible because in some cases 24 hours would be the limit for a missed surveillance.
- C. Incorrect – SR 3.0.3 allows the longer of 24 hours or the specified frequency (92 days). 23 days is based on the grace period (25% of 92 days) allowed by SR 3.0.2.

Technical Reference(s): Technical Specification Surveillance Requirement 3.0.3

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-TS001 Obj 15

Question Source: Bank – NMP2 April 2023 NRC #97

Question History: NMP2 April 2023 NRC #97

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Level of Difficulty (LOD): 3

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.2.25
	Importance Rating	4.2

Knowledge of the bases in technical specifications for limiting conditions for operation and safety limits

Question #98

Technical Specification 3.4.8, RCS Specific Activity, limits the allowable specific activity in the Reactor coolant.

Which one of the following describes the limitation and the basis behind this limitation, in accordance with Technical Specifications?

Limits specific activity of dose equivalent (1) in the Reactor coolant to prevent exceeding 10 CFR limits to an individual (2).

- | | <u>(1)</u> | <u>(2)</u> |
|----|------------|--|
| A. | Cesium 137 | in the Control Room following a Recirc loop rupture |
| B. | Cesium 137 | at the site boundary following a Main Steam line break |
| C. | Iodine 131 | in the Control Room following a Recirc loop rupture |
| D. | Iodine 131 | at the site boundary following a Main Steam line break |

Answer: D

Explanation: Technical Specification 3.4.8 limits the dose equivalent Iodine 131. The basis is to prevent exceeding dose limits to an individual at the site boundary following a Main Steam line break.

- A. Incorrect – Technical Specification 3.4.8 limits the dose equivalent Iodine 131. Plausible because Cesium 137 is a significant radioisotope referenced in the ODCM. The basis behind the limit is an individual at the site boundary following a Main Steam line break. Plausible because dose to Control Room personnel and a Recirc loop rupture are the basis for other requirements.
- B. Incorrect – Technical Specification 3.4.8 limits the dose equivalent Iodine 131. Plausible because Cesium 137 is a significant radioisotope referenced in the ODCM.
- C. Incorrect – The basis behind the limit is an individual at the site boundary following a Main Steam line break. Plausible because dose to Control Room personnel and a Recirc loop rupture are the basis for other requirements.

Technical Reference(s): Technical Specification 3.4.8 and bases

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-TS001 Obj 40

Question Source: Bank – NMP1 2019 NRC #97

Question History: NMP1 2019 NRC #97

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(2)

Level of Difficulty (LOD): 3

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.1.15
	Importance Rating	3.4

Knowledge of administrative requirements for temporary management direction, such as standing orders, night orders, or operations memoranda

Question #99

The plant is operating at 100% power with the following:

- A degraded plant condition requires issuing temporary written guidance to Operations personnel, in accordance with EN-OP-112, Night and Standby Orders.
- It is desired for the guidance to remain in effect for approximately one month and until formally cancelled.

Which one of the following identifies:

- the type of order to be used
- and
- who is required to approved this order,

in accordance with EN-OP-112?

	Type of Order	Required Approval
A.	Night Order	Shift Manager
B.	Night Order	Director, Operations
C.	Standing Order	Shift Manager
D.	Standing Order	Director, Operations

Answer: D

Explanation: Per EN-OP-112, Night Orders are only effective until the next set of Night Orders is issued. For more long-term guidance, Standing Orders are used. They remain in effect until formally cancelled. Standing Orders require the approval of the Director, Operations.

- A. Incorrect – Night Orders are only effective until the next set of Night Orders is issued. Plausible that one month would be a short enough time to fit within the more temporary Night Order process versus the longer-term Standing Order process. Also plausible that a Night Order would require formal cancellation. Approval is required from the Director, Operations. Plausible because many approvals are at the Shift Manager level.
- B. Incorrect – Night Orders are only effective until the next set of Night Orders is issued. Plausible that one month would be a short enough time to fit within the more temporary Night Order process versus the longer-term Standing Order process. Also plausible that a Night Order would require formal cancellation.
- C. Incorrect – Approval is required from the Director, Operations. Plausible because many approvals are at the Shift Manager level.

Technical Reference(s): EN-OP-112

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-PROC Objective 20

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(5)

Level of Difficulty (LOD): 3

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.1.42
	Importance Rating	3.4

Knowledge of new and spent fuel movement procedures

Question #100

Given the following evolutions:

- (1) Replacement of an SRM detector in the Reactor during an outage
- (2) Loading of a new fuel bundle into the Reactor during refueling
- (3) Removal of an irradiated fuel bundle from the Reactor during refueling

Which one of the following identifies which of these evolutions must be observed and directly supervised by an SRO?

- A. (2) only
- B. (3) only
- C. (2) and (3) only
- D. (1), (2), and (3)

Answer: C

Explanation: 03-1-01-5, Refueling IOI, requires all core alterations to be observed and directly supervised by an SRO. Technical Specifications define the movement of any fuel within the Reactor as a core alteration. Evolutions (2) and (3) meet this definition, and therefore must be observed and directly supervised by an SRO. Evolution (1) does not meet this definition and is a specific exception listed in Technical Specifications.

- A. Incorrect – Evolution (3) does need to be observed and directly supervised by an SRO. Plausible because this is lowering the reactivity of the core, which lessens the chance of criticality.
- B. Incorrect – Evolution (2) does need to be observed and directly supervised by an SRO. Plausible because the radiation risk of moving a new fuel bundle is significantly lower than that of moving irradiated fuel.
- D. Incorrect – Evolution (1) does not need to be observed and directly supervised by an SRO. Plausible because it is movement of a component inside the Reactor and it is a specific exception listed in Technical Specifications.

Technical Reference(s): 03-1-01-5, Technical Specification Definitions

References to be provided to applicants during examination: None

Learning Objective: GLP-OPS-IOI05 obj 2.11

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(6)

Level of Difficulty (LOD): 3

Comments:

TRH 11/20/23 – Replaced (1) and revised combinations in A and B based on NRC comments.

INITIAL LICENSED OPERATOR WRITTEN EXAMINATION
OPEN-REFERENCES TABLE OF CONTENTS

2024 NRC SRO Exam

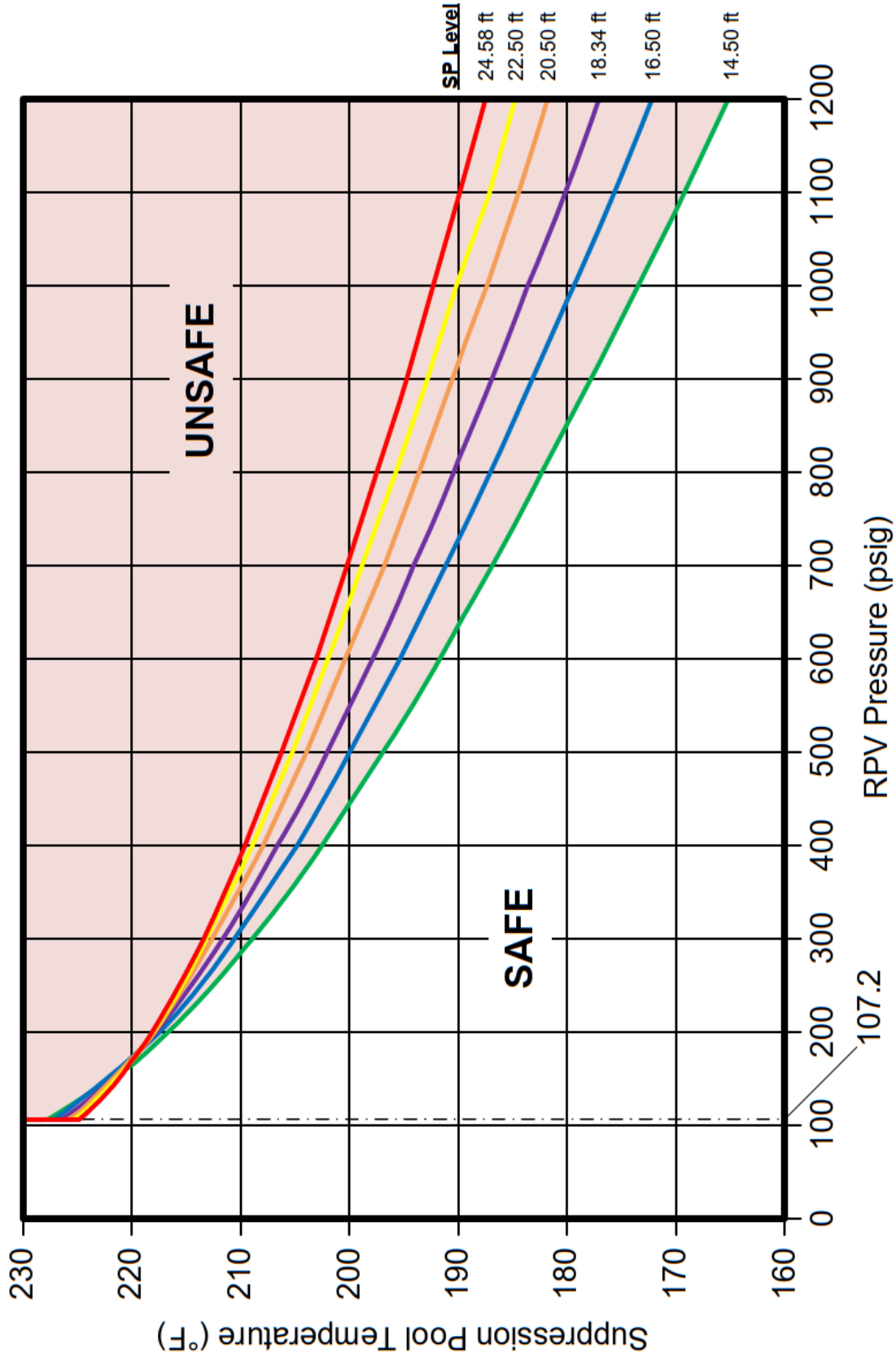
TAB

PROVIDED HANDOUTS

- 1 HCTL, Heat Capacity Temperature Limit, Figure 1, EP-3
- 2 Technical Specifications 3.8.4 and 3.8.6
- 3 Technical Specification 3.3.5.2, TRM 6.3.7

TAB 1

Figure 1
Heat Capacity Temperature Limit



TAB 2

3.8 ELECTRICAL POWER SYSTEMS

3.8.4 DC Sources—Operating

LCO 3.8.4 The Division 1, Division 2, and Division 3 DC electrical power subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required battery charger inoperable.	-----NOTE----- Entry into MODE 1, 2 or 3 is not allowed, except during power reductions. ----- A.1 Verify battery cell parameters meet Table 3.8.6-1 Category A limits.	1 hour <u>AND</u> Once per 8 hours thereafter
B. Required Action and associated Completion Time of Condition A not met.	B.1 Declare associated battery inoperable.	Immediately
C. Division 1 or 2 DC electrical power subsystem inoperable for reasons other than Condition A.	C.1 Restore Division 1 and 2 DC electrical power subsystems to OPERABLE status.	2 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time for Division 1 or 2 DC electrical power subsystem for condition A, B, or C not met.	D.1 -----NOTE ----- LCO 3.0.4.a is not applicable when entering MODE 3. ----- Be in MODE 3.	12 hours
E. Division 3 DC electrical power subsystem inoperable for reasons other than Condition A.	E.1 Declare High Pressure Core Spray System inoperable.	Immediately
F. Required Action and associated Completion Time for Division 3 DC electrical power subsystem for Condition A, B or E not met.	F.1 Be in MODE 3. <u>AND</u> F.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.8.4.1 Verify battery terminal voltage is ≥ 129 V on float charge.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.4.2	Verify no visible corrosion at battery terminals and connectors. <u>OR</u> Verify battery connection resistance is $\leq 1.5 \text{ E-4 ohm}$ for inter-cell connections, $\leq 1.5 \text{ E-4 ohm}$ for inter-rack connections, $\leq 1.5 \text{ E-4 ohm}$ for inter-tier connections, and $\leq 1.5 \text{ E-4 ohm}$ for terminal connections.	In accordance with the Surveillance Frequency Control Program
SR 3.8.4.3	Verify battery cells, cell plates, and racks show no visual indication of physical damage or abnormal deterioration that could degrade battery performance.	In accordance with the Surveillance Frequency Control Program
SR 3.8.4.4	Remove visible corrosion and verify battery cell to cell and terminal connections are coated with anti-corrosion material.	In accordance with the Surveillance Frequency Control Program
SR 3.8.4.5	Verify battery connection resistance is $\leq 1.5 \text{ E-4 ohm}$ for inter-cell connections, $\leq 1.5 \text{ E-4 ohm}$ for inter-rack connections, $\leq 1.5 \text{ E-4 ohm}$ for inter-tier connections, and $\leq 1.5 \text{ E-4 ohm}$ for terminal connections.	In accordance with the Surveillance Frequency Control Program
SR 3.8.4.6	Verify each Division 1 and 2 required battery charger supplies ≥ 400 amps at $\geq 125 \text{ V}$ for ≥ 10 hours; and the Division 3 battery charger supplies ≥ 50 amps at $\geq 125 \text{ V}$ for ≥ 4 hours.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.8.4.7 -----NOTES----- 1. SR 3.8.4.8 may be performed in lieu of SR 3.8.4.7 once per 60 months. 2. This Surveillance shall not be performed in MODE 1, 2, or 3 (not applicable to Division 3). However, credit may be taken for unplanned events that satisfy this SR. ----- Verify battery capacity is adequate to supply, and maintain in OPERABLE status, the required emergency loads for the design duty cycle when subjected to a battery service test.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.8.4.8 -----NOTE----- This Surveillance shall not be performed in MODE 1, 2, or 3 (not applicable to Division 3). However, credit may be taken for unplanned events that satisfy this SR. ----- Verify battery capacity is $\geq 80\%$ of the manufacturer's rating when subjected to a performance discharge test.	In accordance with the Surveillance Frequency Control Program <u>AND</u> 12 months when battery shows degradation, or has reached 85% of the expected life with capacity < 100% of manufacturer's rating <u>AND</u> 24 months when battery has reached 85% of the expected life with capacity $\geq 100\%$ of manufacturer's rating

3.8.6 Battery Cell Parameters

LC0 3.8.6 Battery cell parameters for the Division 1, 2, and 3 batteries shall be within limits.

APPLICABILITY: When associated DC electrical power subsystems are required to be OPERABLE.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each battery.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more batteries with one or more battery cell parameters not within Table 3.8.6-1 Category A or B limits.	A.1 Verify pilot cell's electrolyte level and float voltage meet Table 3.8.6-1 Category C limits.	1 hour
	<u>AND</u>	
	A.2 Verify battery cell parameters meet Table 3.8.6-1 Category C limits.	24 hours
	<u>AND</u>	Once per 7 days thereafter
	<u>AND</u>	
	A.3 Restore battery cell parameters to Table 3.8.6-1 Category A and B limits.	31 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> One or more batteries with average electrolyte temperature of the representative cells < 60°F. <u>OR</u> One or more batteries with one or more battery cell parameters not within Table 3.8.6-1 Category C limits.	B.1 Declare associated battery inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.8.6.1 Verify battery cell parameters meet Table 3.8.6-1 Category A limits.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.6.2	Verify battery cell parameters meet Table 3.8.6-1 Category B limits.	In accordance with the Surveillance Frequency Control Program <u>AND</u> Once within 72 hours after battery overcharge > 150 V
SR 3.8.6.3	Verify average electrolyte temperature of representative cells is $\geq 60^{\circ}\text{F}$.	In accordance with the Surveillance Frequency Control Program

Table 3.8.6-1 (page 1 of 1)
Battery Cell Parameter Requirements

PARAMETER	CATEGORY A: LIMITS FOR EACH DESIGNATED PILOT CELL	CATEGORY B: LIMITS FOR EACH CONNECTED CELL	CATEGORY C: LIMITS FOR EACH CONNECTED CELL
Electrolyte Level	> Minimum level indication mark, and $\leq 1/4$ inch above maximum level indication mark(a)	> Minimum level indication mark, and $\leq 1/4$ inch above maximum level indication mark(a)	Above top of plates, and not overflowing
Float Voltage	≥ 2.13 V	≥ 2.13 V	> 2.07 V
Specific Gravity(b)(c)	≥ 1.195	≥ 1.190 <u>AND</u> Average of all connected cells ≥ 1.200	Not more than 0.020 below average of all connected cells <u>AND</u> Average of all connected cells ≥ 1.190

- (a) It is acceptable for the electrolyte level to temporarily increase above the specified maximum level during equalizing charges provided it is not overflowing.
- (b) Corrected for electrolyte temperature and level. Level correction is not required, however, when battery charging is < 2 amps when on float charge.
- (c) A battery charging current of < 2 amps when on float charge is acceptable for meeting specific gravity limits following a battery recharge, for a maximum of 7 days. When charging current is used to satisfy specific gravity requirements, specific gravity of each connected cell shall be measured prior to expiration of the 7 day allowance.

TAB 3

3.3 INSTRUMENTATION

3.3.5.2 Reactor Pressure Vessel (RPV) Water Inventory Control Instrumentation

LCO 3.3.5.2 The RPV Water Inventory Control instrumentation for each Function in Table 3.3.5.2-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.5.2-1.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more channels inoperable.	A.1 Enter the Condition referenced in Table 3.3.5.2-1 for the channel.	Immediately
B. As required by Required Action A.1 and referenced in Table 3.3.5.2-1.	B.1 Declare associated penetration flow path(s) incapable of automatic isolation.	Immediately
	<u>AND</u> B.2 Calculate DRAIN TIME.	Immediately
C. As required by Required Action A.1 and referenced in Table 3.3.5.2-1.	C.1 Place channel in trip.	1 hour

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. As required by Required Action A.1 and referenced in Table 3.3.5.2-1.	D.1 Declare HPCS System inoperable.	1 hour
	<u>OR</u> D.2 Align the HPCS pump suction to the suppression pool.	1 hour
E. As required by Required Action A.1 and referenced in Table 3.3.5.2-1.	E.1 Restore channel to OPERABLE status.	24 hours
F. Required Action and associated Completion Time of Condition C, D, or E not met.	F.1 Declare associated ECCS injection/spray subsystem inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

-----NOTE-----
Refer to Table 3.3.5.2-1 to determine which SRs apply for each ECCS Function.

SURVEILLANCE		FREQUENCY
SR 3.3.5.2.1	Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program
SR 3.3.5.2.2	Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program
SR 3.3.5.2.3	Perform LOGIC SYSTEM FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program

Table 3.3.5.2-1 (page 1 of 2)
RPV Water Inventory Control Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Low Pressure Coolant Injection-A (LPCI) and Low Pressure Core Spray (LPCS) Subsystems					
a. Reactor Vessel Pressure - Low (Injection Permissive)	4,5	3 ^(a)	C	SR 3.3.5.2.1 SR 3.3.5.2.2	≤ 534 psig
b. LPCS Pump Discharge Flow – Low (Bypass)	4,5	1 ^(a)	E	SR 3.3.5.2.1 SR 3.3.5.2.2	≥ 1285 gpm
c. LPCI Pump A Discharge Flow – Low (Bypass)	4,5	1 ^(a)	E	SR 3.3.5.2.1 SR 3.3.5.2.2	≥ 1133 gpm
d. Manual Initiation	4,5	1 ^(a)	E	SR 3.3.5.2.3	NA
2. LPCI B and LPCI C Subsystems					
a. Reactor Vessel Pressure - Low (Injection Permissive)	4,5	3 ^(a)	C	SR 3.3.5.2.1 SR 3.3.5.2.2	≤ 534 psig
b. LPCI Pump B and LPCI Pump C Discharge Flow – Low (Bypass)	4,5	1 per pump ^(a)	E	SR 3.3.5.2.1 SR 3.3.5.2.2	≥ 1133 gpm
c. Manual Initiation	4,5	1 ^(a)	E	SR 3.3.5.2.3	NA

(continued)

- (a) Associated with an ECCS subsystem required to be OPERABLE by LCO 3.5.2, "Reactor Pressure Vessel (RPV) Water Inventory Control."

Table 3.3.5.2-1 (page 2 of 2)
RPV Water Inventory Control Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3. High Pressure Core Spray (HPCS) System					
a. Condensate Storage Tank Level - Low	4 ^(b) , 5 ^(b)	1 ^(a)	D	SR 3.3.5.2.1 SR 3.3.5.2.2	≥ 4.7 ft
b. HPCS Pump Discharge Pressure - High (Bypass)	4, 5	1 ^(a)	E	SR 3.3.5.2.1 SR 3.3.5.2.2	≥ 108 psig and ≤ 1282 psig
c. HPCS System Flow Rate - Low (Bypass)	4, 5	1 ^(a)	E	SR 3.3.5.2.1 SR 3.3.5.2.2	≥ 1124 gpm and ≤ 1327 gpm
4. RHR System Isolation					
a. Reactor Vessel Water Level – Low, Level 3	(c)	2 in one trip system	B	SR 3.3.5.2.1 SR 3.3.5.2.2	≥ 10.8 inches
5. Reactor Water Cleanup (RWCU) System Isolation					
a. Reactor Vessel Water Level – Low, Level 2	(c)	2 in one trip system	B	SR 3.3.5.2.1 SR 3.3.5.2.2	≥ -43.8 inches

- (a) Associated with an ECCS subsystem required to be OPERABLE by LCO 3.5.2, "Reactor Pressure Vessel (RPV) Water Inventory Control."
- (b) When HPCS is OPERABLE for compliance with LCO 3.5.2, "Reactor Pressure Vessel (RPV) Water Inventory Control," and aligned to the condensate storage tank.
- (c) When automatic isolation of the associated penetration flow path(s) is credited in calculation DRAIN TIME.

6.3 INSTRUMENTATION

6.3.7 FEEDWATER/MAIN TURBINE TRIP INSTRUMENTATION

LCO 6.3.7 The feedwater/main turbine trip Function shall have 3 channels of reactor vessel water level - high, level 9, instrumentation functional.

APPLICABILITY: MODE 1

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each Channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channel(s) nonfunctional.	A.1 Restore channel(s) to FUNCTIONAL status.	7 days
B. The feedwater/main turbine trip Function not maintained.	B.1 Restore the feedwater/main turbine trip capability.	72 hours
C. Required Action and associated Completion Time not met.	C.1 Enter LCO 6.0.1	Immediately

SURVEILLANCE REQUIREMENTS

NOTE

When a channel is placed in a nonfunctional status solely for performance of required surveillances, entry into associated Conditions and Required Actions may be delayed for up to 2 hours, provided that the trip Function capability is maintained.

SURVEILLANCE		FREQUENCY
SR 6.3.7.1	Perform a CHANNEL CHECK.	12 hours
SR 6.3.7.2	Perform a CHANNEL FUNCTIONAL TEST.	24 months
SR 6.3.7.3	Perform a CHANNEL CALIBRATION. The Allowable Value shall be ≤ 58.7 inches and ≥ 57.3 inches. The Trip Setpoint shall be 58.0 inches, nominal. The Trip Output Relay Time Delay shall be ≤ 1.1 seconds and ≥ 0.90 seconds with a nominal setting of 1.0 second.	24 months
SR 6.3.7.4	Perform a LOGIC SYSTEM FUNCTIONAL TEST and simulated automatic operation of all channels.	24 months