# U.S. Nuclear Regulatory Commission

# Site-Specific SRO Written Examination

## Applicant Information

| | |
|---|---|
| Name: | |
| Date: 1/19/15 | Facility/Unit: Susquehanna, LLC |
| Region: I ■ II ☐ III ☐ IV ☐ | Reactor Type: W ☐ CE ☐ BW ☐ GE ■ |
| Start Time: | Finish Time: |

## Instructions

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. To pass the examination you must achieve a final grade of at least 80.00 percent overall, with 70.00 percent or better on the SRO-only items if given in conjunction with the RO exam; SRO-only exams given alone require a final grade of 80.00 percent to pass. You have 8 hours to complete the combined examination, and 3 hours if you are only taking the SRO portion.

## Applicant Certification

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

______________________________
Applicant's Signature

## Results

| | | |
|---|---|---|
| RO/SRO-Only/Total Examination Values | ____ / ____ / ____ | Points |
| Applicant's Scores | ____ / ____ / ____ | Points |
| Applicant's Grade | ____ / ____ / ____ | Percent |

| Exam | RO | Tier | 2 | Group | 1 | Cognitive Level | High | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 263000 K1.02 DC Electrical Distribution | | | | | Importance | 3.2 | | |
| Statement | Knowledge of the physical connections and/or cause-effect relationships between D.C. ELECTRICAL DISTRIBUTION and the following: Battery charger and battery | | | | | | | | |

QUESTION 1

Both Units are operating at 100% power with the following:

- ESS Bus 1B (1A202) de-energizes due to a sustained electrical fault.

Which one of the following describes the effect of this fault on the 125 VDC System?

A. Load Center 1D622 de-energizes.

B. Battery Charger 1D623 remains energized and supplies power to Load Center 1D622.

C. Battery Charger 1D623 de-energizes and Battery 1D620 supplies power to Load Center 1D622.

D. Battery Charger 1D623 de-energizes and Battery Charger 2D623 supplies power to Load Center 1D622.

Proposed Answer: **B**

Applicant References: **None**

Explanation: **Battery Charger 1D623 is normally powered from MCC 0B526. MCC 0B526 is normally powered from ESS Bus 1B. Upon loss of ESS Bus 1B, MCC 0B526 automatically switches to receiving power from ESS Bus 2B due to the action of an automatic transfer switch. Since MCC 0B526 remains energized, Battery Charger 1D623 remains energized and supplying power to Load Center 1D622.**

**A Incorrect – Load Center 1D622 remains energized due to automatic switching of the AC sources to Battery Charger 1D623.**

**B Correct.**

**C Incorrect – Battery 1D620 would supply power to Load Center 1D622 if Battery Charger 1D623 de-energized. However, since Battery Charger 1D623 remains energized due to automatic switching of its AC source, Battery 1D620 remains as the backup source to Load Center 1D622.**

**D Incorrect – A Unit 2 power source does automatically keep Load Center 1D622 energized, however it is through automatic switching of the AC sources to Battery Charger 1D623, not automatic switching of which Battery Charger is connected to Load Center 1D622.**

10CFR55: **41.7**

Technical References: **ON-104-202, TM-OP-002**

Learning Objectives: **TMOP002/1431**

Question Source: **Bank** **Vision SYSID 32125**

Previous NRC Exam: **No**

Comments

| Exam | RO | Tier | 2 | Group | 1 | Cognitive Level | Low | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | **215004 K1.01 Source Range Monitor** | | | | | Importance | | **3.6** | |
| Statement | **Knowledge of the physical connections and/or cause-effect relationships between SOURCE RANGE MONITOR (SRM) SYSTEM and the following: Reactor protection system** | | | | | | | | |

QUESTION 2

A normal plant startup is in progress in accordance with GO-100-102, Plant Startup, Heatup and Power Operation.

Which one of the following describes the status of the SRM shorting links per GO-100-102 and the effect on the ability of the SRMs to cause a Reactor scram?

The SRM shorting links are ___(1)___, therefore the SRMs ___(2)___ cause a Reactor scram.

| | (1) | (2) |
|---|---|---|
| A. | installed | can |
| B. | installed | can NOT |
| C. | removed | can |
| D. | removed | can NOT |

Proposed Answer **B**

Applicant References **None**

Explanation **The SRM shorting links are normally installed, which prevents the SRMs from causing a Reactor scram.**

**A Incorrect – The shorting links are normally installed, but this prevents SRMs from causing a Reactor scram.**

**B Correct.**

**C Incorrect – The SRM shorting links are normally installed. They are only removed during Shutdown Margin Testing.**

**D Incorrect – The SRM shorting links are normally installed. They are only removed during Shutdown Margin Testing.**

10CFR55 **41.7**

Technical References **GO-100-102, TM-OP-078A**

Learning Objectives **TM-OP-078A 1342.a**

Question Source **New**

Previous NRC Exam **No**

Comments

| Exam | RO | Tier | 2 | Group | 1 | Cognitive Level | High | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 259002 A1.02 Reactor Water Level Control | | | | | Importance | 3.6 | | |
| Statement | Ability to predict and/or monitor changes in parameters associated with operating the REACTOR WATER LEVEL CONTROL SYSTEM controls including: Reactor Feedwater flow | | | | | | | | |

QUESTION 3

Unit 1 was operating at 100% power when the following occurred:

- The Reactor was manually scrammed.
- HPCI and RCIC were overridden to prevent Reactor water level from reaching +54".
- Feedwater Level Control (FWLC) Setpoint Setdown initiated.
- Reactor water level is being maintained at +22" by Feedwater Level Control.
- RFP Pump A is injecting.
- Ten minutes have elapsed since Reactor water level was stabilized at +22".

Which one of the following describes the plant response to resetting the FWLC Setpoint Setdown logic with NO additional operator action?

The FWLC system will...

A. automatically restore Reactor water level to +35" by raising RFPT A speed in flow control mode.

B. automatically restore Reactor water level to +35" by opening FW LOW LOAD VALVE LV-10641.

C. continue to maintain Reactor water level at +22" by maintaining RFPT A speed in flow control mode.

D. continue to maintain Reactor water level at +22" by controlling the position of FW LOW LOAD VALVE LV-10641.

Proposed Answer: **D**

Applicant References: **None**

Explanation: **The FWLC system will automatically align the Feedwater system to startup level control via LV-10641 with one Feedwater pump in discharge pressure control mode following a scram. The low load controller LIC-C32-iR602 setpoint will remain at 22" following reset of Setpoint Setdown until operators return the setpoint to +35".**

**A Incorrect – The level setpoint will remain at +22" until manually adjusted. The FWLC system automatically aligns to startup level control mode, therefore LV-10641 will control flow, not RFPT A in flow control mode.**

**B Incorrect – The level setpoint will remain at +22" until manually adjusted.**

**C Incorrect – The FWLC system automatically aligns to startup level control mode, therefore LV-10641 will control flow, not RFPT A in flow control mode.**

**D Correct.**

10CFR55: **41.4**

Technical References: **OP-145-001, TM-OP-045I**

| | |
|---|---|
| Learning Objectives | **TMOP045I / 120001** |
| Question Source | **Bank** **LOC25 Cert #6** |
| Previous NRC Exam | **No** |
| Comments | |

| Exam | RO | Tier | 2 | Group | 1 | Cognitive Level | Low | Level of Difficulty | 2.5 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | | 239002 K2.01 SRVs | | | | Importance | | 2.8 | |
| Statement | | Knowledge of electrical power supplies to the following: SRV solenoids | | | | | | | |

QUESTION 4

Which one of the following describes the electrical power supply to the SRV solenoids that provide the pressure relief function?

A. One 125 VDC distribution panel supplies all 16 SRVs.

B. One 120 VAC distribution panel supplies all 16 SRVs.

C. Two 125 VDC distribution panels each supply half of the 16 SRVs.

D. Two 120 VAC distribution panels each supply half of the 16 SRVs.

Proposed Answer **A**

Applicant References **None**

Explanation **125 VDC distribution panel 1D614 supplies electrical power to the pressure relief solenoids for all 16 SRVs.**

**A Correct.**

**B Incorrect – The electrical power source is 125 VDC, not 120 VAC.**

**C Incorrect – Two 125 VDC distribution panels provide electrical power for the ADS solenoids, however a single 125 VDC distribution panel provides electrical power for all pressure relief solenoids.**

**D Incorrect - Two 125 VDC distribution panels provide electrical power for the ADS solenoids, however a single 125 VDC distribution panel provides electrical power for all pressure relief solenoids.**

10CFR55 **41.3**

Technical References **TM-OP-83E**

Learning Objectives **1661.R**

Question Source **New**

Previous NRC Exam **No**

Comments

| Exam | RO | Tier | 2 | Group | 1 | Cognitive Level | High | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 262002 K3.07 UPS (AC/DC) | | | | | Importance | 2.6 | | |
| Statement | Knowledge of the effect that a loss or malfunction of the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) will have on following: Movement of control rods: Plant-Specific | | | | | | | | |

QUESTION 5

Unit 1 is operating at 100% power with the following:

- A malfunction occurs with the Vital UPS.
- Bus 1Y629 de-energizes.

Which one of the following describes an effect of this malfunction?

A. Control rod movement with RMCS is unavailable.

B. The RTIME Computer System is removed from service.

C. Standby Liquid Control System B squib valve is disabled.

D. Recirc MG set scoop tubes lock and runback signals initiate.

Proposed Answer **A**

Applicant References **None**

Explanation **Loss of Bus 1Y629 causes a loss of power to the Rod Drive Control Cabinet which causes CRD insert and withdraw valves to fail closed. This results in a loss of the ability to move control rods with RMCS.**

**A Correct.**

**B Incorrect – Loss of 1Y629 does not result in loss of RTIME. Loss of 1Y619 does result in loss of RTIME.**

**C Incorrect – Loss of 1Y629 causes loss of SBLC tank level alarms, but does not disable the B squib valve. Loss of 1Y216 does disable the B squib valve.**

**D Incorrect – Loss of 1Y629 does not result in Recirc MG set scoop tube lock or a runback signal. Loss of 1Y128 does cause Recirc MG set scoop tubes to lock and runback signals to initiate.**

10CFR55 **41.7**

Technical References **ON-117-001**

Learning Objectives **2472.C**

Question Source **New**

Previous NRC Exam **No**

Comments

| Exam | RO | Tier | 2 | Group | 1 | Cognitive Level | High | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 261000 K3.06 SGTS | | | | | Importance | 3.0 | | |
| Statement | Knowledge of the effect that a loss or malfunction of the STANDBY GAS TREATMENT SYSTEM will have on following: Primary containment oxygen content: Mark-I&II | | | | | | | | |

QUESTION 6

A plant startup is in progress on Unit 1 with the following:

- Containment inerting is in progress.
- Standby Gas Treatment (SGTS) fan A is running and aligned to take suction from the Unit 1 Primary Containment only.
- Primary Containment oxygen concentration is 15%, down slow.

Then, a malfunction in the Unit 1 PCIS logic causes a RB Zone 3 Isolation Signal on a -38" Reactor water level signal to be initiated.

Which one of the following describes the effect of this malfunction on Primary Containment oxygen concentration?

Primary Containment oxygen concentration will...

A. rise.

B. remain approximately constant.

C. continue to lower, but at a slower rate.

D. continue to lower at approximately the same rate.

Proposed Answer **B**

Applicant References **None**

Explanation **The RB Zone 3 Isolation Signal on -38" Reactor water level causes the SGTS suction on the Primary Containment to isolate and the nitrogen supply to the Primary Containment to isolate. With no flow entering or exiting Primary Containment, oxygen concentration will remain approximately constant.**

**A Incorrect – Both the SGTS suction and the N2 supply to the Primary Containment will isolate, causing Primary Containment oxygen concentration to remain approximately constant. Plausible if applicant believed plant response caused air to enter Primary Containment.**

**B Correct.**

**C Incorrect – Both the SGTS suction and the N2 supply to the Primary Containment will isolate, causing Primary Containment oxygen concentration to remain approximately constant. Plausible if applicant believed plant response isolated SGTS suction but kept N2 addition going.**

**D** **Incorrect – Both the SGTS suction and the N2 supply to the Primary Containment will isolate, causing Primary Containment oxygen concentration to remain approximately constant. Plausible if applicant believed this specific isolation signal kept both SGTS suction and N2 addition going.**

| | | |
|---|---|---|
| 10CFR55 | **41.9** | |
| Technical References | **TM-OP-070, OP-173-001, ON-159-002** | |
| Learning Objectives | **1985.j / 1991.a** | |
| Question Source | **Bank** | **LOC23 NRC #13** |
| Previous NRC Exam | **Yes** | **LOC23 NRC #13** |
| Comments | | |

| Exam | RO | Tier | 2 | Group | 1 | Cognitive Level | High | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 400000 K4.01 Component Cooling Water | | | | | Importance | 3.4 | | |
| Statement | Knowledge of CCWS design feature(s) and or interlocks which provide for the following: Automatic start of standby pump | | | | | | | | |

QUESTION 7

Unit 1 is operating at 100% power with the following:

- RBCCW pump 1A is running.
- RBCCW pump 1B is in standby.
- RBCCW pump discharge header pressure is 90 psig, steady.
- Then, RBCCW pump 1A breaker trips on overcurrent.
- RBCCW pump discharge header pressure is 55 psig, down slow.

Which one of the following describes the status of RBCCW pump 1B?

RBCCW pump 1B...

A. auto-starts due to a RBCCW pump 1A breaker trip signal.

B. auto-starts due to a low discharge header pressure signal.

C. remains in standby until discharge header pressure lowers further.

D. remains in standby until Operators manually start it.

Proposed Answer: **B**

Applicant References: **None**

Explanation: **The standby RBCCW pump will auto-start on trip of the running pump based on a low discharge header pressure signal (<61 psig).**

**A Incorrect – RBCCW pump 1B does auto-start, but there is no RBCCW pump auto-start due to a breaker trip signal from the other RBCCW pump.**

**B Correct.**

**C Incorrect – Low discharge header pressure is required to auto-start RBCCW pump 1B, however pressure is already below the setpoint of 61 psig.**

**D Incorrect – The standby RBCCW pump will auto-start due to low discharge header pressure.**

10CFR55: **41.4**

Technical References: **AR-123-E03, TM-OP-014**

Learning Objectives: **1694.a**

Question Source: **Bank JAF 9/12 NRC #53**

Previous NRC Exam: **Yes JAF 9/12 NRC #53**

Comments

| Exam | RO | Tier | 2 | Group | 1 | Cognitive Level | High | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 209001 K4.05 LPCS | | | | | Importance | 2.6 | | |
| Statement | Knowledge of LOW PRESSURE CORE SPRAY SYSTEM design feature(s) and/or interlocks which provide for the following: Pump minimum flow | | | | | | | | |

QUESTION 8

Unit 1 is operating at 100% power with the following:

- Flow verification testing is in progress on Core Spray loop A.
- HV-152-F015A, Core Spray Loop A Test to Supp Pool, is full open.

Then, a loss of coolant accident results in the following:

- Reactor water level is -140", down slow.
- Reactor pressure is 475 psig, down slow.
- Drywell pressure is 15 psig, up slow.
- Core Spray loop A valves are aligned as shown in the picture on the next page.

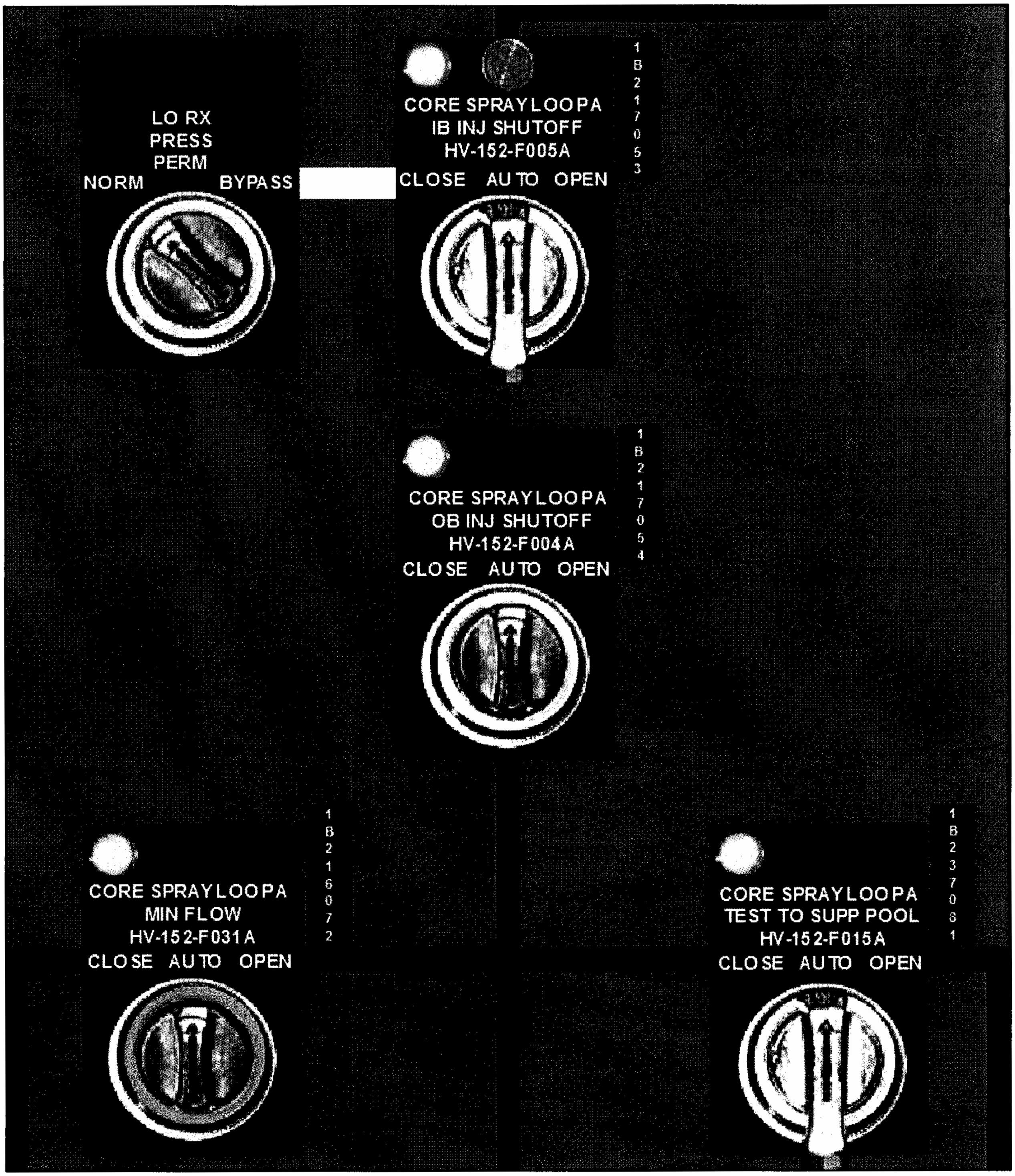
Which one of the following describes the status of Core Spray loop A valves?

A. Core Spray loop A valves have operated as designed.

B. HV-152-F015A, Core Spray Loop A Test to Supp Pool, is closed, but should be open.

C. HV-152-F005A, Core Spray Loop A IB Inj Shutoff, is closed, but should be open.

D. HV-152-F031A, Core Spray Loop A Min Flow, is closed, but should be open.


LO RX
PRESS
PERM
NORM BYPASS
CORE SPRAY LOOP A
IB INJ SHUTOFF
HV-152-F005A
CLOSE AUTO OPEN
1B217053
CORE SPRAY LOOP A
OB INJ SHUTOFF
HV-152-F004A
CLOSE AUTO OPEN
1B217054
CORE SPRAY LOOP A
MIN FLOW
HV-152-F031A
CLOSE AUTO OPEN
1B216072
CORE SPRAY LOOP A
TEST TO SUPP POOL
HV-152-F015A
CLOSE AUTO OPEN
1B237081

| | |
|---|---|
| Proposed Answer | **D** |
| Applicant References | **None** |
| Explanation | **With flow verification testing initially in progress and HV-152F015A full open, Core Spray loop A injection valve (F005A) was initially closed and the min flow valve (F031A) was initially closed. When Reactor water level went below -129", Core Spray received an automatic initiation signal. This signal causes the Core Spray test valve (F015A) to immediately close. As this valve closes and Core Spray flow lowers, the min flow valve (F031A) should have automatically opened. The picture shows the min flow valve closed.**<br>**A Incorrect – The min flow valve (F031A) should be open, but indicates closed.**<br>**B Incorrect – Although the test valve (F015A) was initially open, it automatically closes as soon as Reactor water level reaches -129". This valve does not wait to re-position until Reactor pressure reaches 420 psig, such as the injection valves do.**<br>**C Incorrect – The injection valves (F004A/F005A) do not automatically open until Reactor pressure reaches 420 psig.**<br>**D Correct.** |
| 10CFR55 | **41.8** |
| Technical References | **OP-151-001** |
| Learning Objectives | **2080.d, 2080.g** |
| Question Source | **Modified Bank JAF 4/14 NRC #15** |
| Previous NRC Exam | **Yes JAF 4/14 NRC #15** |
| Comments | |

| Exam | RO | Tier | 2 | Group | 1 | Cognitive Level | High | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 262001 K5.02 AC Electrical Distribution | | | | | Importance | | 2.6 | |
| Statement | Knowledge of the operational implications of the following concepts as they apply to A.C. ELECTRICAL DISTRIBUTION: Breaker control | | | | | | | | |

QUESTION 9

Unit 1 is operating at 100% power with the following:

- Division II RHR is in Suppression Pool Cooling using RHR pump 1D and RHRSW pump 1B.
- An electrical transient results in loss of 125 VDC ESS Channel B distribution panel 1D624.
- One minute later, a Bus Undervoltage condition occurs on 4KV ESS bus 1A204.

Which one of the following describes the response of the RHRSW pump 1B breaker?

RHRSW pump 1B breaker __(1)__ because __(2)__.

| | (1) | (2) |
|---|---|---|
| A. | remains closed | its control power is NOT available |
| B. | remains closed | it is NOT affected by the undervoltage condition |
| C. | trips | Its control power is NOT affected by loss of 1D624 |
| D. | trips | its control power automatically transferred to 1D644 |

Proposed Answer **D**

Applicant References **None**

Explanation **RHRSW pump 1B loses its normal breaker control power upon loss of 1D624, however control power to the trip circuit automatically transfers to 1D644. The breaker then trips on the undervoltage signal on bus 1A204.**

**A Incorrect – RHRSW pump 1B breaker trips on the Bus Undervoltage condition with control power automatically supplied from 1D644.**

**B Incorrect – RHRSW pump 1B breaker trips on the Bus Undervoltage condition with control power automatically supplied from 1D644.**

**C Incorrect – RHRSW pump 1B loses its normal breaker control power upon loss of 1D624, however control power to the trip circuit automatically transfers to 1D644.**

**D Correct.**

10CFR55 **41.7**

Technical References **ON-102-620, TM-OP-016**

Learning Objectives **2058.c and 2058.d**

Question Source **Bank** **LOC24 Cert #9**

Previous NRC Exam **No**

Comments

| Exam | RO | Tier | 2 | Group | 1 | Cognitive Level | High | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 217000 K5.02 RCIC | | | | | Importance | 3.1 | | |
| Statement | Knowledge of the operational implications of the following concepts as they apply to REACTOR CORE ISOLATION COOLING SYSTEM (RCIC): Flow indication | | | | | | | | |

QUESTION 10

Unit 1 is shutdown following an automatic Reactor scram with the following:

- Reactor pressure is 900 psig, steady.
- Reactor water level is +25", up slow.
- RCIC is injecting to the Reactor at 400 gpm with the RCIC flow controller in AUTOMATIC.

Then, the RCIC pump discharge flow element fails low.

Which one of the following describes the response of actual RCIC pump flow and the ability to control RCIC flow with the flow controller in MANUAL?

Actual RCIC pump flow ___(1)___. The RCIC flow controller ___(2)___ function properly in MANUAL.

| | (1) | (2) |
|---|---|---|
| A. | lowers | will |
| B. | lowers | will NOT |
| C. | rises | will |
| D. | rises | will NOT |

Proposed Answer C

Applicant References None

Explanation

With the RCIC pump discharge flow element failed low, both the flow input signal to the flow controller and the control room flow indication will lower. The flow controller will sense that flow is below demanded flow of 400 gpm and further open the RCIC throttle valve. RCIC speed and actual flow will rise. Taking the flow controller to MANUAL gets rid of the flow feedback signal to the RCIC throttle valve. The flow element failure will have no adverse affect on control of RCIC flow in MANUAL.

A Incorrect – While RCIC flow indication will lower, actual RCIC flow will rise.

B Incorrect – While RCIC flow indication will lower, actual RCIC flow will rise.

C Correct.

D Incorrect – Actual RCIC pump flow will rise, however the flow controller will function completely normally in MANUAL.

10CFR55 41.8

Technical References TM-OP-050

| | |
|---|---|
| Learning Objectives | **2008.c, 2015.i** |
| Question Source | **New** |
| Previous NRC Exam | **No** |
| Comments | |

| Exam | RO | Tier | 2 | Group | 1 | Cognitive Level | High | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 205000 K6.05 Shutdown Cooling | | | | | Importance | 3.2 | | |
| Statement | Knowledge of the effect that a loss or malfunction of the following will have on the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE): Component cooling water systems | | | | | | | | |

QUESTION 11

A Unit 1 shutdown is in progress with the following:

- RHR loop A has been placed in Shutdown Cooling.
- RHR pump 1A is running.
- RHRSW pump 1A is running.
- Reactor pressure is 39 psig, steady.

Then, RHRSW pump 1A trips on overcurrent. Reactor pressure is 41 psig, up slow.

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

A. RHR loop A Shutdown Cooling lineup automatically isolates based on a signal from RHRSW pump 1A breaker position.

B. RHR loop A remains in the Shutdown Cooling lineup, but RHRSW flow to the heat exchanger CANNOT be restored.

C. RHR loop A remains in the Shutdown Cooling lineup and RHRSW flow to the heat exchanger can be restored using RHRSW pump 2A.

D. RHR loop A remains in the Shutdown Cooling lineup and RHRSW flow to the heat exchanger can be restored using RHRSW pump 1B.

Proposed Answer: **C**

Applicant References: **None**

Explanation: **When RHRSW pump 1A trips, cooling water flow to RHR heat exchanger 1A is lost. AR-109-G02 and OP-116-001 provide guidance for aligning RHRSW pump 2A to supply cooling water to the Unit 1 RHR loop A heat exchanger.**

- **A Incorrect – RHR loop A will isolate if Reactor pressure reaches 98 psig, but does not automatically isolate based on trip of RHRSW pump 1A.**
- **B Incorrect – RHR loop A does remain in the SDC lineup, but RHRSW flow can be restored to the heat exchanger using RHRSW pump 2A.**
- **C Correct.**
- **D Incorrect – RHR loop A does remain in the SDC lineup and can have RHRSW flow restored to the heat exchanger from an alternate pump. However, it is RHRSW pump 2A, not 1B, that can be cross-connected.**

10CFR55: **41.3**

Technical References: **OP-149-002, TM-OP-016, TM-OP-049, OP-116-001, AR-109-G02**

Learning Objectives: **2058.a**

Question Source: **New**

Previous NRC Exam **No**

Comments

| Exam | RO | Tier | 2 | Group | 1 | Cognitive Level | High | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 218000 K6.05 ADS | | | | | Importance | 3.0 | | |
| Statement | Knowledge of the effect that a loss or malfunction of the following will have on the AUTOMATIC DEPRESSURIZATION SYSTEM: A.C. power: Plant-Specific | | | | | | | | |

QUESTION 12

Unit 1 has experienced a loss of coolant accident with the following:

- All offsite power has been lost.
- EDGs C and D have failed to start.
- HPCI and RCIC have tripped.
- Drywell pressure is 15 psig, up slow.
- Reactor pressure is 800 psig, down slow.
- Reactor water level is -129", down slow.

Which one of the following describes the response of the Automatic Depressurization System (ADS) logic timers?

A. NEITHER division ADS logic timer has initiated.

B. Division I ADS logic timer has initiated, but division II has NOT.

C. Division II ADS logic timer has initiated, but division I has NOT.

D. Both division ADS logic timers have initiated.

Proposed Answer: **D**

Applicant References: **None**

Explanation: **Loss of all offsite power, combined with failure of EDGs C and D to start, results in no power to RHR pumps C and D and Core Spray pumps C and D. RHR pumps A and B and Core Spray pumps A and B are running. This is a sufficient number of ECCS pumps to satisfy both divisions of the ADS logic. Both divisions of ADS logic remain energized by their DC source. Therefore, when Reactor water level reaches -129" with Drywell pressure >1.72 psig, both division ADS logic timers have initiated.**

**A Incorrect – With RHR pumps A and B running, a sufficient number of running ECCS pumps is available to initiate both division ADS logic timers.**

**B Incorrect – With RHR pumps A and B running, a sufficient number of running ECCS pumps is available to initiate both division ADS logic timers.**

**C Incorrect – With RHR pumps A and B running, a sufficient number of running ECCS pumps is available to initiate both division ADS logic timers.**

**D Correct.**

10CFR55: **41.8**

Technical References: **TM-OP-83E, AR-110-A01**

Learning Objectives: **2105.c, 2103.e**

Question Source: **Modified Bank LOC23 Cert #25**

Previous NRC Exam: **No**

Comments

| Exam | RO | Tier | 2 | Group | 1 | Cognitive Level | High | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | **223002 A1.02 PCIS/Nuclear Steam Supply Shutoff** | | | | | Importance | 3.7 | | |
| Statement | **Ability to predict and/or monitor changes in parameters associated with operating the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF controls including: Valve closures** | | | | | | | | |

QUESTION 13

Following a small break LOCA on Unit 1, HPCI is being used to control reactor water level. Conditions are:

- 42 inches RPV water level
- 950 psig Reactor pressure
- 4.2 psig Drywell pressure

WHICH ONE of the following describes the HPCI System response if the PCO depresses the HPCI manual isolation pushbutton on 1C601?

A. HPCI will continue to inject.

B. ONLY the Inboard Isolation Valves will close.

C. ONLY the Outboard Isolation Valves will close.

D. Both the Inboard and Outboard Isolation Valves will close.

Proposed Answer **C**

Applicant References **None**

Explanation **Note: The K/A requires testing the candidate's ability to predict and/or monitor changes in valve closures (parameter) associated with operating PCIS/NSSSS controls. The question meets the K/A by providing various answer choices of valve closures that occur when operating an NSSSS control, specifically, the HPCI manual isolation pushbutton. This requires the candidate to first recognize that a HPCI initiation signal exists and serves as a permissive for the manual isolation to function. If this permissive is not known, candidate may incorrectly believe that an initiation signal will override a manual isolation signal which supports plausibility of distractor A. The candidate must also determine which isolation valve(s) will close which supports plausibility of distractors B & D.**

**A** **Incorrect – With an initiation signal present, the manual isolation pushbutton isolates Division 2 (F003 and F042).**

**B** **Incorrect – Only Division 2 is effected by the HPCI manual Isolation logic. Division 1 closes F002 and F100.**

**C** **Correct.**

**D** **Incorrect – With an initiation signal present, the manual isolation pushbutton only isolates Division 2 (F003 and F042).**

10CFR55 **41.7**

Technical References **TM-OP-052; OP-152-001, Section 2.11**

Learning Objectives **TMOP-OP-059B 10481.a**

Question Source **Bank** **SSES LOR Bank 5/5/2010 TMOP052/2038/ 006**

| | |
|---|---|
| Previous NRC Exam | **No** |
| Comments | **RTB 1/9/15 – Replaced question based on NRC comment for K/A match.** |

| Exam | RO | Tier | 2 | Group | 1 | Cognitive Level | High | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 211000 A1.08 SLC | | | | | Importance | | 3.7 | |
| Statement | Ability to predict and/or monitor changes in parameters associated with operating the STANDBY LIQUID CONTROL SYSTEM controls including: RWCU system lineup | | | | | | | | |

QUESTION 14

Unit 1 is operating at 50% with the following:

- Reactor Water Cleanup (RWCU) is in service.
- A loss of Feedwater occurs.
- The Reactor is manually scrammed.
- Reactor water level reaches a low of -25".
- Control Rod Drive (CRD) flow is maximized for injection.
- Standby Liquid Control (SBLC) pump 1A is started for injection.

Which one of the following describes the response of the RWCU inlet isolation valves?

| | HV-144-F001, RWCU Inlet Inboard Isolation Valve | HV-144-F004, RWCU Inlet Outboard Isolation Valve |
|---|---|---|
| A. | Remains open | Remains open |
| B. | Remains open | Closes |
| C. | Closes | Remains open |
| D. | Closes | Closes |

Proposed Answer B

Applicant References None

Explanation RWCU inlet outboard isolation valve (F004) automatically closes when SBLC pump 1A is started. RWCU inlet inboard isolation valve (F001) does NOT close on this same signal. Both valves would also receive an isolation signal if Reactor water level lowered to -38". Since Reactor water level reached a low of only -28", this isolation signal was NOT received. Therefore, RWCU inlet inboard isolation valve (F001) remains open.

A Incorrect – Even though SBLC pump 1A is being started for level control and not specifically for boron injection, the RWCU isolation still occurs. This causes F004 to close.

B Correct.

C Incorrect – Start of SBLC pump 1A causes F004 to close, not F001.

D Incorrect – Start of SBLC pump 1A only closes F004. Reactor water level did not reach the low level isolation setpoint that would also close F001.

10CFR55 41.6

Technical References OP-153-001, ON-159-002

Learning Objectives 10095.g

Question Source Bank Vision SYSID 33499

| | |
|---|---|
| Previous NRC Exam | **No** |
| Comments | |

| Exam | RO | Tier | 2 | Group | 1 | Cognitive Level | Low | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | | 215005 A2.03 APRM / LPRM | | | | | Importance | | 3.6 |
| Statement | | 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 conditions or operations: Inoperative trip (all causes) | | | | | | | |

QUESTION 15

Unit 1 is operating at 100% power with the following:

- APRM 1 is bypassed while a manual gain adjustment is being performed.
- APRM 2 indicates a critical self-test fault.

Which one of the following describes the response of the Voters and the ability to bypass APRM 2 in accordance with OP-178-002, Power Range Neutron Monitoring System (PRNMS)?

The Voters...

A. receive a vote. APRM 2 may be bypassed while APRM 1 is still bypassed.

B. receive a vote. APRM 1 must be un-bypassed before bypassing APRM 2.

C. do NOT receive a vote. APRM 2 may be bypassed while APRM 1 is still bypassed.

D. do NOT receive a vote. APRM 1 must be un-bypassed before bypassing APRM 2.

Proposed Answer: **B**

Applicant References: **None**

Explanation: **A critical self-test fault causes the associated APRM to generate an INOP signal. An INOP signal sends one vote to all four of the Voters, just as if the APRM were generating an upscale signal. OP-178-002 cautions that "any attempt to bypass more than one APRM will result in all APRMs being un bypassed". Therefore APRM 1 must be un-bypassed before APRM 2 may be bypassed.**

**A Incorrect – The Voters do receive a vote due to an APRM upscale trip, however APRMs 1 and 2 may NOT be simultaneously bypassed.**

**B Correct.**

**C Incorrect – A critical self-test fault causes the associated APRM to generate an INOP signal. An INOP signal sends one vote to all four of the Voters, just as if the APRM were generating an upscale signal. APRMs 1 and 2 may NOT be simultaneously bypassed.**

**D Incorrect - A critical self-test fault causes the associated APRM to generate an INOP signal. An INOP signal sends one vote to all four of the Voters, just as if the APRM were generating an upscale signal.**

10CFR55: **41.7**

Technical References: **OP-178-002, AR-103-001, TM-OP-078D**

Learning Objectives: **15716.e**

Question Source: **New**

Previous NRC Exam: **No**

Comments

| Exam | RO | Tier | 2 | Group | 1 | Cognitive Level | Low | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | | 264000 A2.05 EDGs | | | | Importance | | 3.6 | |
| Statement | | Ability to (a) predict the impacts of the following on the EMERGENCY GENERATORS (DIESEL/JET); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Synchronization of the emergency generator with other electrical supplies | | | | | | | |

QUESTION 16

Unit 1 is operating at 100% power with the following:

- Diesel Generator (DG) A is running.
- DG A is being synchronized to 4.16KV Bus 1A per SO-024-001A.DG A Gov Mode Sel switch is selected to ISOCHRONOUS.
- The DG A to Bus 1A Sync Sel switch is selected to ON.
- The associated synchroscope is rotating in the FAST (clockwise) direction at 1 revolution per 60 seconds.

Which one of the following describes the acceptability of these conditions for synchronizing DG A to Bus 1A, in accordance with SO-024-001A?

A. These conditions are all acceptable for synchronizing DG A to Bus 1A.

B. The DG A Gov Mode Sel switch position is unacceptable and must be taken to DROOP. Synchroscope indication is acceptable.

C. The synchroscope indication is unacceptable and must be sped up. The DG A Gov Mode Sel switch position is acceptable.

D. Both DG A Gov Mode Sel switch position and synchroscope indication are unacceptable. The DG A Gov Mode Sel switch must be taken to DROOP and the synchroscope must be sped up.

Proposed Answer **B**

Applicant References **None**

Explanation **DG Gov Mode Sel switch position and synchroscope rotation speed are important during synchronization with an energized bus to ensure proper load sharing once the DG output breaker is closed. SO-024-001A requires the DG A Gov Mode Sel switch to be in DROOP, not ISOCH, to prevent unstable load sharing and a potential DG trip. SO-024-001A requires the synchroscope to be rotating in the FAST direction ~1 revolution per 60 seconds to ensure proper synchronization and initial loading of the DG. Therefore, the given synchroscope rotation is acceptable, but the DG A Gov Mode Sel switch must be taken to DROOP.**

**A** **Incorrect - The DG A Gov Mode Sel switch must be taken to DROOP to ensure proper synchronization to an energized bus.**

**B** **Correct.**

**C** **Incorrect – Synchroscope rotation is in accordance with the requirement in SO-024-001A. The DG A Gov Mode Sel switch must be taken to DROOP to ensure proper synchronization to an energized bus.**

| | |
|---|---|
| | **D Incorrect – Synchroscope rotation is in accordance with the requirement in SO-024-001A.** |
| 10CFR55 | **41.10** |
| Technical References | **SO-024-001A** |
| Learning Objectives | **2254** |
| Question Source | **New** |
| Previous NRC Exam | **No** |
| Comments | |

| Exam | RO | Tier | 2 | Group | 1 | Cognitive Level | High | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 203000 A3.05 RHR/LPCI: Injection Mode | | | | | Importance | 4.4 | | |
| Statement | Ability to monitor automatic operations of the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) including: Reactor water level | | | | | | | | |

QUESTION 17

Unit 1 has experienced a loss of coolant accident with the following:

- Reactor water level is -110", down slow.
- Reactor pressure is 400 psig, down slow.
- Drywell pressure is 7.5 psig, up slow.

Which one of the following describes the response of RHR?

RHR...

A. is currently injecting to the Reactor.

B. remains in a standby lineup until Reactor water level reaches -129".

C. pumps have automatically started, but will NOT inject until Reactor water level reaches -129".

D. pumps have automatically started, but will NOT inject until Reactor pressure lowers further.

Proposed Answer **D**

Applicant References **None**

Explanation **Reactor water level has not yet reached the LPCI initiation setpoint of -129". However, LPCI also initiates on the combination of high Drywell pressure (>1.72 psig) and low Reactor pressure (<420 psig). With both of these parameters met, RHR pumps have started. With Reactor pressure <420 psig, the LPCI injection valves have also opened. RHR is not yet injecting to the Reactor because Reactor pressure is above the shutoff head of the pumps. Once Reactor pressure lowers below approximately 300 psig, RHR will begin to inject, regardless of current Reactor water level.**

**A Incorrect – RHR is operating in LPCI mode with the injection valves open, however Reactor pressure is too high for the system to be currently injecting.**

**B Incorrect – Although Reactor water level <-129" is a LPCI initiation signal, RHR has already initiated in the LPCI mode due to the combination of high Drywell pressure and low Reactor pressure.**

**C Incorrect - Although Reactor water level <-129" is a LPCI initiation signal, the combination of high Drywell pressure and low Reactor pressure alone can cause RHR to inject in the LPCI mode.**

**D Correct.**

10CFR55 **41.8**

Technical References **OP-149-001, TM-OP-049**

Learning Objectives **181.a**

| | | |
|---|---|---|
| Question Source | **Modified Bank** | **LOC23 NRC #14** |
| Previous NRC Exam | **No** | |
| Comments | | |

| Exam | RO | Tier | 2 | Group | 1 | Cognitive Level | High | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | **215003 A3.03 IRM** | | | | | Importance | | 3.7 | |
| Statement | **Ability to monitor automatic operations of the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM including: RPS status** | | | | | | | | |

QUESTION 18

A startup is in progress on Unit 1 with the following:

- All Intermediate Range Monitors (IRMs) are on range 4.
- IRM A is INOP due to loss of high voltage.
- IRM A is NOT yet bypassed.
- All other IRMs indicate as follows:

| IRM | Indication |
|---|---|
| B | 75 |
| C | 125 |
| D | 105 |
| E | 90 |
| F | 85 |
| G | 100 |
| H | 80 |

Which one of the following describes the status of control rod blocks and the Reactor Protection System (RPS) based on IRMs?

These IRM conditions result in...

A. NEITHER a control rod block NOR a half scram.

B. a control rod block, but NO half scram.

C. a control rod block and a half scram, but NO full scram.

D. a control rod block and a full scram.

Proposed Answer **C**

Applicant References **None**

Explanation **Loss of high voltage to IRM A causes an INOP trip. This causes both a control rod block and half scram on RPS channel A. These protective actions are still being enforced since IRM A has not yet been bypassed. IRM C is indicating above the upscale trip setpoint of 120. This is also causing a half scram on RPS channel A. All of the RPS channel B IRMs (B, D, F, H) are below their trip setpoints.**

**A Incorrect – IRMs A and C both cause a control rod block and a half scram on RPS channel A.**

**B Incorrect - IRMs A and C both cause a control rod block and a half scram on RPS channel A.**

**C Correct.**

D Incorrect – Two IRMs are causing a half scram, but both input to RPS channel A. No IRMs are causing a half scram on RPS channel B.

| | |
|---|---|
| 10CFR55 | 41.7 |
| Technical References | AR-104-001 A05 and A06 |
| Learning Objectives | 2347.c |
| Question Source | Modified Bank LOC23 Cert #13 |
| Previous NRC Exam | No |
| Comments | |

| Exam | RO | Tier | 2 | Group | 1 | Cognitive Level | High | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 206000 A4.06 HPCI | | | | | Importance | 4.3 | | |
| Statement | Ability to manually operate and/or monitor in the control room: Reactor pressure: BWR-2,3,4 | | | | | | | | |

QUESTION 19

Unit 1 has experienced a scram with the following conditions:

- Reactor water level is +35", steady, with RCIC injecting.
- HPCI has been started for pressure control per OP-152-001, HPCI System.
- Reactor pressure is 1040 psig, steady.
- HPCI discharge pressure is 625 psig, steady.
- HPCI speed is 3350 rpm, steady.
- HPCI flow is 3500 gpm, steady, with FC-E41-1R600, HPCI TURBINE FLOW CONTROL, in AUTO.
- HV-155-F008, HPCI TEST LINE TO CST ISO, is throttled partially open.

Which one of the following describes the actions required to maximize the Reactor cooldown rate with HPCI in accordance with OP-152-001?

A. Raise the FC-E41-1R600 setpoint and throttle HV-155-F008 further open.

B. Raise the FC-E41-1R600 setpoint and throttle HV-155-F008 further closed.

C. Lower the FC-E41-1R600 setpoint and throttle HV-155-F008 further open.

D. Lower the FC-E41-1R600 setpoint and throttle HV-155-F008 further closed.

Proposed Answer **B**

Applicant References **None**

Explanation **The Reactor cooldown rate is maximized by raising HPCI flow while also raising pump discharge pressure. This is accomplished by raising the setpoint of FC-E41-1R600 and throttling HV-155-F008 further closed.**

**A Incorrect – Raising the flow controller setpoint is correct, however HV-155-F008 must be throttled further closed to raise pump discharge pressure.**

**B Correct.**

**C Incorrect – Lowering the flow controller setpoint and throttling HV-155-F008 further open will lower the amount of work the HPCI turbine must performed, lowering steam draw to the HPCI turbine and the Reactor cooldown rate.**

**D Incorrect – HV-155-F008 does need to be throttled further closed to raise pump discharge pressure, however to maximize cooldown the flow controller setpoint must also be raised.**

10CFR55 **41.5**

Technical References **OP-152-001**

Learning Objectives **2035.b**

Question Source **Bank** **Vision SYSID 33565**

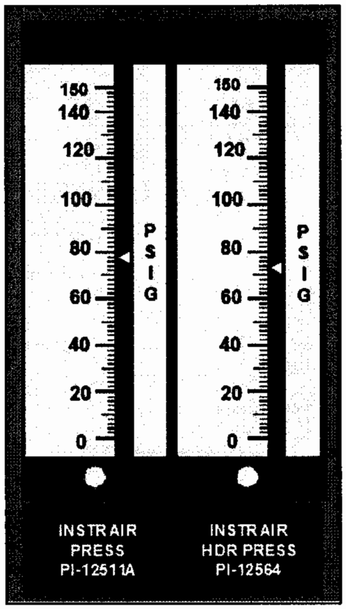
Previous NRC Exam **No**

Comments

| Exam | RO | Tier | 2 | Group | 1 | Cognitive Level | High | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 300000 A4.01 Instrument Air | | | | | Importance | 2.6 | | |
| Statement | Ability to manually operate and/or monitor in the control room: Pressure gauges | | | | | | | | |

QUESTION 20

Unit 1 is operating at 100% power with the following:




- Instrument Air Compressor (IAC) 1A is operating as the lead compressor.
- IAC 1B is the standby compressor.
- Then, an air leak develops in the plant.
- Control Room air pressure indications are as shown in the following picture, and have been slowly lowering since the leak developed.

Which one of the following describes the status of IAC 1B and PCV-12560, Service Air Crosstie to Instrument Air, and the need for a manual Reactor scram based on current plant parameters, in accordance with ON-118-001?

A. IAC 1B is running.
PCV-12560 is closed.
A manual Reactor scram is NOT required.

B. IAC 1B remains in standby.
PCV-12560 is open.
A manual Reactor scram is NOT required.

C. IAC 1B is running.
PCV-12560 is open.
A manual Reactor scram is NOT required.

D. IAC 1B is running.
PCV-12560 is closed.
A manual Reactor scram is required.

| | |
|---|---|
| Proposed Answer | **C** |
| Applicant References | **None** |
| Explanation | **Instrument Air pressure indications are below the point at which IAC 1B starts (87 psig on PI-12511A) and PCV-12560 opens (~95 psig on PI-12511A, ~85 psig on PI-12564). Instrument Air pressure is above the threshold requiring a manual Reactor scram (65 psig).**<br>**A Incorrect – PCV-12560 is open.**<br>**B Incorrect – IAC 1B is running.**<br>**C Correct.**<br>**D Incorrect – A manual Reactor scram is not yet required.** |
| 10CFR55 | **41.5** |
| Technical References | **ON-118-001** |
| Learning Objectives | **1769.a/c, 1769** |
| Question Source | **Modified Bank JAF 4/14 NRC #19** |
| Previous NRC Exam | **No** |
| Comments | |

| Exam | RO | Tier | 2 | Group | 1 | Cognitive Level | Low | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | | 212000 2.4.34 RPS | | | | Importance | | 4.2 | |
| Statement | | Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects. | | | | | | | |

QUESTION 21

Unit 1 is operating at 100% power with the following:

- Control Room evacuation becomes necessary due to a fire.
- The Control Room is evacuated ***before*** any of the immediate actions of ON-100-009, Control Room Evacuation, can be performed.

Which one of the following describes the required control of the Reactor in accordance with ON-100-009?

A. Scram the Reactor by manually venting the scram air header.

B. Scram the Reactor by opening RPS power distribution breakers.

C. Lower Reactor power by manually tripping Recirculation MG Set breakers.

D. Lower Reactor power by manually operating Recirculation MG Set scoop tubes.

Proposed Answer **B**

Applicant References **None**

Explanation **ON-100-009 contains an operator action to place the Mode Switch to SHUTDOWN prior to leaving the Control Room, as time permits. The procedure also contains subsequent action in case there is not enough time to scram the Reactor prior to leaving the Control Room. In this case, the Reactor is scrammed by opening RPS power distribution breakers.**

A **Incorrect – Manually venting the scram air header would achieve the desired control rod insertion and is a viable option during failure to scram conditions. However, ON-100-009 achieves the Reactor scram by opening RPS power distribution breakers.**

**B Correct.**

**C Incorrect - The Reactor is required to be scrammed and no power reduction with Recirculation flow is performed before the scram.**

**D Incorrect – The Reactor is required to be scrammed and no power reduction with Recirculation flow is performed before the scram.**

10CFR55 **41.10**

Technical References **ON-100-009**

Learning Objectives

Question Source **New**

Previous NRC Exam **No**

Comments

| Exam | RO | Tier | 2 | Group | 1 | Cognitive Level | High | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 211000 2.1.23 SLC | | | | | Importance | | 4.3 | |
| Statement | Ability to perform specific system and integrated plant procedures during all modes of plant operation. | | | | | | | | |

QUESTION 22

Unit 1 has experienced a failure to scram with the following:

- Initial ATWS power was >5%.
- Based on the rate of control rod insertion, it is expected that EO-100-113, Level/Power Control, will be exited in two hours.
- The Unit Supervisor has directed boron injection using Standby Liquid Control (SBLC) in accordance with OP-153-001, Standby Liquid Control System.
- Initial SBLC tank volume is 1800 gallons.

Which one of the following describes when EO-100-113 requires stopping boron injection and the approximate time that this amount of boron injection will take, in accordance with OP-153-001?

EO-100-113 requires stopping boron injection when SBLC tank volume reaches ___(1)___.
Per OP-153-001, this amount of boron injection will take approximately ___(2)___.

| | (1) | (2) |
|---|---|---|
| A. | 450 gallons | 17 minutes |
| B. | 450 gallons | 34 minutes |
| C. | 0 gallons | 23 minutes |
| D. | 0 gallons | 45 minutes |

Proposed Answer **D**

Applicant References **None**

Explanation **EO-100-113 requires stopping boron injection when SBLC tank volume reaches 0 gallons. OP-153-001 section 2.2 and Attachment A provide the direction for injecting boron with the SBLC system. These procedures direct injecting with one SBLC pump. This provides approximately 40 gpm of flow. Therefore, it will take approximately 45 minutes to inject the entire 1800 gallons of solution from the SBLC tank into the Reactor.**

**A Incorrect – 450 gallons is the tank volume at which the cold shutdown boron weight (1350 gallons) will have been injected, however EO-100-113 does not direct securing boron injection until tank volume drops to 0 gallons. 17 minutes is the approximate time it would take 2 SBLC pumps to inject 1350 gallons.**

**B Incorrect - 450 gallons is the tank volume at which the cold shutdown boron weight (1350 gallons) will have been injected, however EO-100-113 does not direct securing boron injection until tank volume drops to 0 gallons. 34 minutes is the approximate time it would take 1 SBLC pump to inject 1350 gallons.**

**C** **Incorrect – 0 gallons is the correct tank volume at which to secure boron injection. However, OP-153-001 only directs starting one SBLC pump, which provides an injection rate of approximately 40 gpm. This results in boron injection lasting approximately 45 minutes, not 23 minutes.**

**D** **Correct.**

| | |
|---|---|
| 10CFR55 | **41.10** |
| Technical References | **EO-100-113, EO-000-113, OP-153-001** |
| Learning Objectives | **10098.a** |
| Question Source | **New** |
| Previous NRC Exam | **No** |
| Comments | |

| Exam | RO | Tier | 2 | Group | 1 | Cognitive Level | High | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 261000 K4.04 SGTS | | | | | Importance | 2.7 | | |
| Statement | Knowledge of STANDBY GAS TREATMENT SYSTEM design feature(s) and/or interlocks which provide for the following: Radioactive particulate filtration | | | | | | | | |

QUESTION 23

Both Units are operating at 100% power with the following:

- The Standby Gas Treatment System (SGTS) is in a normal standby lineup.
- A steam leak develops inside the Unit 2 Drywell.
- Unit 2 is manually scrammed.
- Unit 2 Reactor water level reaches a low of -25" during the transient.
- Unit 2 Reactor water level is now +35", steady.
- Unit 2 Drywell pressure is 7.5 psig, up slow.

Which one of the following describes the response of SGTS and the Reactor Building Ventilation System Zones?

A. Both SGTS fans start. Zones 1, 2, and 3 isolate.

B. Both SGTS fans start. Zones 2 and 3 isolate, only.

C. Only one SGTS fan starts. Zones 1, 2, and 3 isolate.

D. Only one SGTS fan starts. Zones 2 and 3 isolate, only.

Proposed Answer **B**

Applicant References **None**

Explanation **High Drywell pressure (>1.72 psig) on either Unit causes both SGTS fans to automatically start (both are aligned in "Auto Lead" in a normal standby lineup). High Drywell pressure on a single Unit only causes two of the RB Ventilation Zones to isolate (1 and 3 isolate on Unit 1 high Drywell pressure, 2 and 3 isolate on Unit 2 high Drywell pressure). This response ensures SGTS is drawing suction from the Secondary Containment surrounding the affected Drywell to filter any radioactive leakage from the Primary Containment.**

**A Incorrect – No isolation signal is present for RB Ventilation Zone 1.**

**B Correct.**

**C Incorrect – Both SGTS fans start. No isolation signal is present for RB Ventilation Zone 1.**

**D Incorrect – Both SGTS fans start.**

10CFR55 **41.9**

Technical References **ON-259-002**

Learning Objectives **1991.a**

Question Source **Bank** **Vision SYSID 33385**

Previous NRC Exam **No**

Comments

| Exam | RO | Tier | 2 | Group | 1 | Cognitive Level | Low | Level of Difficulty | 3.5 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 400000 K6.07 Component Cooling Water | | | | | Importance | 2.7 | | |
| Statement | Knowledge of the effect that a loss or malfunction of the following will have on the CCWS: Breakers, relays, and disconnects | | | | | | | | |

QUESTION 24

Unit 1 is operating at 100% power with the following:

- A loss of Reactor Building Chilled Water occurs.
- Given the following possible independent conditions:

  (1) Trip of the supply breaker to 1B236
  (2) Trip of the supply breaker to 1B246
  (3) A high Drywell pressure Primary Containment Isolation signal

Which one of the following identifies which of these conditions would prevent re-aligning RBCCW to the Drywell to supply the Drywell Coolers and other Drywell loads?

A. (1) or (2), only

B. (1) or (3), only

C. (2) or (3), only

D. (1), (2), or (3)

Proposed Answer **D**

Applicant References **None**

Explanation **The RBCCW supply and return lines to Drywell loads each have two normally closed Containment Isolation valves. For each line, the outboard valve is powered by 1B236 and the inboard valve is powered by 1B246. Loss of either of these two power supplies prevents opening these valves. Additionally, a high Drywell pressure isolation signal would prevent opening these valves.**

**A Incorrect – A high Drywell pressure isolation signal would also prevent aligning RBCCW to the Drywell.**

**B Incorrect – Loss of 1B246 would also prevent aligning RBCCW to the Drywell.**

**C Incorrect – Loss of 1B236 would also prevent aligning RBCCW to the Drywell.**

**D Correct.**

10CFR55 **41.9**

Technical References **ON-104-203 and 204 Attachment E, TM-OP-014**

Learning Objectives **1676.b**

Question Source **Bank LOC25 Cert #3**

Previous NRC Exam **No**

Comments

| Exam | RO | Tier | 2 | Group | 1 | Cognitive Level | High | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 262001 A4.05 AC Electrical Distribution | | | | | Importance | 3.3 | | |
| Statement | Ability to manually operate and/or monitor in the control room: Voltage, current, power, and frequency on A.C. buses | | | | | | | | |

QUESTION 25

A plant startup and heatup is in progress on Unit 1 with the following:

- Unit 1 Aux Buses are being supplied by the Startup Bus 10.
- Grid voltage is degrading.
- Startup and Aux Bus voltages decrease from approximately 13.8kV to 10.35kV over several minutes and then stabilizes.

Which one of the following describes the plant response if these conditions continue?

A. Aux Buses will experience undervoltage load shedding.

B. Startup Bus 10 will experience undervoltage load shedding.

C. Tie Bus to Aux Bus Feeder Breakers will trip open on undervoltage.

D. Startup Bus Feeder Breaker 0A10301 will trip open on undervoltage.

Proposed Answer: **A**

Applicant References: **None**

Explanation: **The given bus voltage is approximately 75% of nominal voltage (10.35/13.8=0.75). The Aux Buses automatically load shed on ≤81% (11.4 kV) voltage (with inverse time delay between 5 and 50 seconds depending on how low voltage is).**

**A Correct.**

**B Incorrect – Startup bus load shedding does not occur until approximately 25% voltage (~3.25kV).**

**C Incorrect – Tie Bus to Aux Bus feeder breakers do not trip on undervoltage.**

**D Incorrect - Startup bus feeder breaker does not trip until voltage is 64% (~8.88kV) for 0.5 seconds.**

10CFR55: **41.7**

Technical References: **ON-103-001, ON-103-003, TM-OP-003**

Learning Objectives: **10054.j**

Question Source: **Bank** **LOC25 Cert #50**

Previous NRC Exam: **No**

Comments

| Exam | RO | Tier | 2 | Group | 1 | Cognitive Level | Low | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | | **263000 K5.01 DC Electrical Distribution** | | | | Importance | | **2.6** | |
| Statement | | **Knowledge of the operational implications of the following concepts as they apply to D.C. ELECTRICAL DISTRIBUTION: Hydrogen generation during battery charging** | | | | | | | |

QUESTION 26

Unit 1 is operating at 100% power with the following:

- An equalizing charge is in progress on 125 VDC Battery 1D610.
- Approximately 24 hours remain in the equalizing charge.
- All Battery Room Ventilation is lost.
- Maintenance reports that it will take approximately 6 hours to restore Battery Room Ventilation.
- Battery Room temperature is 80°F for Battery 1D610.

Which one of the following describes how to control the equalizing charge and the associated reason or limitation, in accordance with OP-102-001, 125V DC System, and ON-030-002, Loss of Control Structure HVAC?

A. Return Battery Charger 1D613 to FLOAT to prevent hydrogen buildup.

B. Return Battery Charger 1D613 to FLOAT to prevent excessive temperatures.

C. Continue the equalizing charge as long as room temperature remains below 102°F.

D. Continue the equalizing charge as long as air samples indicate less than 1% hydrogen.

Proposed Answer: **A**

Applicant References: **None**

Explanation: **OP-102-001 Precaution 2.4.2 requires returning the charger to float to prevent hydrogen buildup. ON-030-002 also requires ensuring all battery chargers are in float within 3 hours.**

**A Correct.**

**B Incorrect – The equalizing charge must be secured, but the reason is to prevent hydrogen buildup, not prevent excessive temperatures.**

**C Incorrect – The equalizing charge must be secured regardless of actual room temperature rise based on hydrogen generation concerns. 102°F is a high temperature limitation associated with Reactor Building Emergency Switchgear Rooms.**

**D Incorrect – The equalizing charge must be secured regardless of actual hydrogen concentration. 1% hydrogen concentration is the lowest value used in EOP combustible gas control strategies.**

10CFR55: **41.10**

Technical References: **OP-102-001, ON-030-002**

Learning Objectives: **10145**

Question Source: **Bank** **JAF4/14 NRC #9**

Previous NRC Exam: **Yes** **JAF 4/14 NRC #9**

Comments

| Exam | RO | Tier | 2 | Group | 2 | Cognitive Level | High | Level of Difficulty | 2.5 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | | 241000 K1.04 Reactor/Turbine Pressure Regulator | | | | Importance | | 3.7 | |
| Statement | | Knowledge of the physical connections and/or cause-effect relationships between REACTOR/TURBINE PRESSURE REGULATING SYSTEM and the following: Reactor steam flow | | | | | | | |

QUESTION 27

Unit 1 is operating at approximately 15% power during a startup with the following:

- Three Turbine Bypass Valves (TBVs) are fully open.
- Preparations are being made to roll the Main Turbine.
- Reactor pressure is 950 psig, steady.
- Then, an SRV inadvertently opens.

Which one of the following describes the response of the TBVs and Reactor pressure?

TBVs will...

A. close slightly. Reactor pressure will stabilize at 950 psig.

B. close slightly. Reactor pressure will lower and stabilize less than 950 psig.

C. remain at their initial position. Reactor pressure will lower until MSIVs close.

D. remain at their initial position. Reactor pressure will lower and stabilize less than 950 psig.

Proposed Answer **B**

Applicant References **None**

Explanation **When the SRV opens, Reactor steam flow rises as extra steam is passed to the Suppression Pool. Due to the rise in steam flow, Reactor pressure and Main Steam Line pressure lower. EHC senses the lower pressure and attempts to raise pressure by throttling TBVs further closed. EHC will return sensed pressure Turbine valve manifold to the pre-transient value. With less steam flow through the Main Steam lines, there will be less head loss from the Reactor to the Turbine valve manifold. Therefore, while EHC sensed pressure will return to the pre-transient value, Reactor pressure will be lower than the initial 950 psig.**

**A Incorrect – Due to lower head losses from less steam flow down the Main Steam lines, Reactor pressure will stabilize at less than the initial 950 psig.**

**B Correct.**

**C Incorrect – EHC does not directly sense the SRV opening through position indication or steam flow measurement. However, the extra steam flow through the SRV will cause Reactor pressure to lower. This in turn causes EHC sensed pressure to lower. Since EHC sensed pressure lowers, EHC will throttle the TBVs further closed. If EHC did not throttle TBVs, then Reactor pressure would continue to lower until MSIVs closed.**

**D Incorrect – EHC does not directly sense the SRV opening through position indication or steam flow measurement. However, the extra steam flow through the SRV will cause Reactor pressure to lower. This in turn causes EHC sensed pressure to lower. Since EHC sensed pressure lowers, EHC will throttle the TBVs further closed. Reactor pressure will stabilize less than 950 psig, but due to throttling of the TBVs.**

10CFR55 **41.5**

| | | |
|---|---|---|
| Technical References | **ON-183-001, TM-OP-093L** | |
| Learning Objectives | **1641.c** | |
| Question Source | **Bank** | **Vision SYSID 689** |
| Previous NRC Exam | **No** | |
| Comments | | |

| Exam | RO | Tier | 2 | Group | 2 | Cognitive Level | Low | Level of Difficulty | 2.5 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 239001 K2.01 Main and Reheat Steam | | | | | Importance | 3.2 | | |
| Statement | Knowledge of electrical power supplies to the following: Main steam isolation valve solenoids | | | | | | | | |

QUESTION 28

Unit 1 is operating at 100% power when 1Y201A, RPS Distribution Panel A, de-energizes due to a sustained electrical fault.

Which one of the following describes the response of HV-141-F022A, Mn Stm Line A IB Iso (MSIV 1A), and its associated pilot solenoids?

A. Both solenoids re-position and the MSIV closes.

B. One solenoid re-positions and the MSIV closes.

C. One solenoid re-positions, but the MSIV remains open.

D. Neither solenoid re-positions and the MSIV remains open.

Proposed Answer: **C**

Applicant References: **None**

Explanation: **MSIV 1A has two pilot solenoids that are energized to open the MSIV. One solenoid is powered from 1Y201A and the other solenoid is DC powered. Both solenoids must de-energize for the MSIV to close. Therefore, with only one solenoid de-energized by the loss of 1Y201A, MSIV 1A remains open.**

**A Incorrect – Only one of the pilot solenoids is powered from 1Y201A. The other pilot solenoid is DC powered.**

**B Incorrect – Only one of the pilot solenoids re-positions, however the MSIV remains open.**

**C Correct.**

**D Incorrect – The pilot solenoids are energized when the MSIV is open. Loss of power to one of the solenoids causes it to re-position.**

10CFR55: **41.3**

Technical References: **TM-OP-083**

Learning Objectives: **10233**

Question Source: **Bank** **Vision SYSID 33651**

Previous NRC Exam: **No**

Comments

| Exam | RO | Tier | 2 | Group | 2 | Cognitive Level | Low | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 230000 K3.02 RHR/LPCI: Torus/Pool Spray Mode | | | | | Importance | | 3.3 | |
| Statement | Knowledge of the effect that a loss or malfunction of the RHR/LPCI: TORUS/SUPPRESSION POOL SPRAY MODE will have on following: Suppression pool temperature | | | | | | | | |

QUESTION 29

Unit 1 has experienced a loss of coolant accident with the following:

- Suppression Chamber spray is being initiated on RHR loop 1A per OP-149-004, RHR Containment Cooling.
- HV-151-F028A, SUPP CHMBR SPR TEST SHUTOFF, fails to open.
- Operators in the field have been unable to manually open HV-151-F028A.

Which one of the following describes the ability to utilize RHR loop 1A to spray the Suppression Chamber with RHRSW or cool the Suppression Pool?

| | Suppression Chamber Spray With RHRSW Through RHR Loop 1A | Suppression Pool Cooling With RHR Loop 1A |
|---|---|---|
| A. | Available | Available |
| B. | Available | Unavailable |
| C. | Unavailable | Available |
| D. | Unavailable | Unavailable |

Proposed Answer **D**

Applicant References **None**

Explanation **Failure of HV-151-F028A prevents the use of RHR loop 1A for normal Suppression Chamber spray, alternate Suppression Chamber spray with RHRSW, and Suppression Pool cooling. RHR loop 1A remains available for LPCI and Drywell spray.**

**A Incorrect – Alternate Suppression Chamber spray using RHRSW and Suppression Pool cooling both require HV-151-F028A to open.**

**B Incorrect – Alternate Suppression Chamber spray using RHRSW requires HV-151-F028A to open.**

**C Incorrect – Suppression Pool cooling requires HV-151-F028A to open.**

**D Correct.**

10CFR55 **41.8**

Technical References **OP-149-004, OP-116-001**

Learning Objectives **176.m**

Question Source **New**

Previous NRC Exam **No**

Comments

| Exam | RO | Tier | 2 | Group | 2 | Cognitive Level | Low | Level of Difficulty | 2.5 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | **215002 K4.01 RBM** | | | | | Importance | 3.4 | | |
| Statement | **Knowledge of ROD BLOCK MONITOR SYSTEM design feature(s) and/or interlocks which provide for the following: Prevent control rod withdrawal: BWR-3,4,5** | | | | | | | | |

QUESTION 30

A startup is in progress on Unit 1 with the following:

- APRMs indicate 30%.
- A hardware problem on Rod Block Monitor (RBM) A results in an INOP trip.

Which one of the following describes the status of RBM A and control rod blocks?

RBM A is...

A. auto-bypassed. No control rod blocks are enforced.

B. in-service. No control rod blocks are enforced.

C. in-service. A control rod withdraw block is enforced, only.

D. in-service. Control rod withdraw and insert blocks are enforced.

Proposed Answer **C**

Applicant References **None**

Explanation **The RBM is in-service since APRMs indicate above the low power setpoint of 24.9%. The INOP trip of RBM A initiates a withdraw block, but not an insert block.**

**A Incorrect – With APRMs >24.9%, RBM A is in-service, not auto-bypassed.**

**B Incorrect – RBM A is in-service, but the INOP trip results in a withdraw block.**

**C Correct.**

**D Incorrect – RBM A is in-service and enforces a control rod withdraw block, but does not enforce an insert block, such as an INOP RWM would.**

10CFR55 **41.7**

Technical References **TM-OP-78K**

Learning Objectives **15804.c**

Question Source **Bank LOC25 NRC #29**

Previous NRC Exam **Yes LOC25 NRC #29**

Comments

| Exam | RO | Tier | 2 | Group | 2 | Cognitive Level | High | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 201001 K5.02 CRD Hydraulic | | | | | Importance | 2.6 | | |
| Statement | Knowledge of the operational implications of the following concepts as they apply to CONTROL ROD DRIVE HYDRAULIC SYSTEM: Flow indication | | | | | | | | |

QUESTION 31

Unit 1 is operating at 100% power with the following:

- FC-C12-1R600, CRD Flow Controller, is in AUTO with a setpoint of 63 gpm.
- Then, the signal from the CRD system flow element to FC-C12-1R600 fails high.

Given the following possible effects:

(1) CRDM temperatures will rise.
(2) CRD drive water pressure will lower.
(3) Control rod scram times will be slower.

Which one of the following identifies which of these effects occur as a result of this failure?

A. (1) only.

B. (2) only.

C. (1) and (2) only.

D. (1), (2), and (3).

Proposed Answer C

Applicant References None

Explanation With high indicated CRD system flow, FC-C12-1R600 will close in an attempt to lower flow. Based on the location of the CRD flow control valve, this lowers CRD cooling water flow, lowers CRD drive water pressure, and raises CRD charging header pressure. CRDM temperatures will rise due to the lower cooling water flow. Control rod scram times will be faster, not slower, due to higher charging header pressure.

A Incorrect – CRDM temperatures will rise, however CRD drive water pressure will also be lower.

B Incorrect – CRD drive water pressure will lower, however CRDM temperatures will also rise.

C Correct.

D Incorrect – Control rod scram times will be faster, not slower, because charging header pressure rises.

10CFR55 41.6

Technical References TM-OP-055

Learning Objectives 2419.i

Question Source Modified Bank Vision SYSID 34788

Previous NRC Exam No

Comments

| Exam | RO | Tier | 2 | Group | 2 | Cognitive Level | Low | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 290003 K6.02 Control Room HVAC | | | | | Importance | | 2.7 | |
| Statement | Knowledge of the effect that a loss or malfunction of the following will have on the CONTROL ROOM HVAC: Component cooling water systems | | | | | | | | |

QUESTION 32

Both Units are operating at 100% power with the following:

- Control Structure Chiller A (0K112A) in service.
- Then, 0K112A trips due to high bearing temperature.

Which one of the following describes the effect on Control Room temperature?

Control Room temperature will...

A. rise until the standby chiller can be manually placed in service.

B. remain stable because the standby chiller will automatically start.

C. rise until the CREOASS system can be manually placed in service.

D. remain stable because the CREOASS system will automatically start.

Proposed Answer: **B**

Applicant References: **None**

Explanation: **The standby chiller automatically starts, resulting in Control Room temperature remaining stable.**

**A Incorrect – The standby chiller automatically starts, resulting in Control Room temperature remaining stable.**

**B Correct.**

**C Incorrect - The standby chiller automatically starts, resulting in Control Room temperature remaining stable. CREOASS would only be manually started if both chillers were lost.**

**D Incorrect – The standby chiller will automatically start, not CREOASS. CREOASS does have multiple automatic start signals, such as a LOCA signal or various high radiation levels.**

10CFR55: **41.7**

Technical References: **ON-030-001**

Learning Objectives: **1969.h**

Question Source: **Bank** **LOC23 Cert #32**

Previous NRC Exam: **No**

Comments

| Exam | RO | Tier | 2 | Group | 2 | Cognitive Level | Low | Level of Difficulty | 2.5 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | | 233000 A1.02 Fuel Pool Cooling/Cleanup | | | | Importance | 2.9 | | |
| Statement | | Ability to predict and/or monitor changes in parameters associated with operating the FUEL POOL COOLING AND CLEAN-UP controls including: Pool level | | | | | | | |

QUESTION 33

Unit 1 is operating at 100% power with the following:

- Normal makeup is in progress to the Fuel Pool Cooling and Cleanup System per OP-135-001, Fuel Pool Cooling and Cleanup System.

Which one of the following describes the behavior of Spent Fuel Pool and Skimmer Surge Tank levels and the source of the makeup water?

| | Spent Fuel Pool Level | Skimmer Surge Tank Level | Source of Makeup Water |
|---|---|---|---|
| A. | Rises | Remains stable | Condensate Transfer System |
| B. | Rises | Remains stable | Demineralized Water System |
| C. | Remains stable | Rises | Condensate Transfer System |
| D. | Remains stable | Rises | Demineralized Water System |

Proposed Answer **C**

Applicant References **None**

Explanation **Normal makeup to the Fuel Pool Cooling and Cleanup System adds water from the Condensate Transfer System to the Skimmer Surge Tanks. Skimmer Surge Tank level is maintained in band through periodic manual makeup. Spent Fuel Pool level remains constant (due to the weirs) during this operation, with just Skimmer Surge Tank level rising.**

**A Incorrect – SFP level remains constant and Skimmer Surge Tank level rises.**

**B Incorrect – SFP level remains constant and Skimmer Surge Tank level rises. Normal makeup is from Condensate Transfer. The Demineralized Water System is the Unit 2 source of makeup.**

**C Correct.**

**D Incorrect – Normal makeup is from Condensate Transfer. The Demineralized Water System is the Unit 2 source of makeup.**

10CFR55 **41.4**

Technical References **OP-135-001**

Learning Objectives **2198.e**

Question Source **New**

Previous NRC Exam **No**

Comments

| Exam | RO | Tier | 2 | Group | 2 | Cognitive Level | High | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | **201003 A2.08 Control Rod and Drive Mechanism** | | | | | Importance | **3.8** | | |
| Statement | **Ability to (a) predict the impacts of the following on the CONTROL ROD AND DRIVE MECHANISM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Low HCU accumulator pressure/high level** | | | | | | | | |

QUESTION 34

Unit 1 is operating at 100% power with the following:

- Control Rod Drive (CRD) pump 1B is out of service for preventative maintenance.
- CRD pump 1A trips due to an overcurrent condition.
- Electrical Maintenance reports that the pump motor has failed.
- The following is a timeline of events:

| Time (minutes) | Event |
|---|---|
| $T_0$ | CRD pump 1A trips. |
| $T_0 + 10$ | FIRST accumulator trouble alarm is received on Control Rod 22-23.<br>Control Rod 22-23 is at position 10.<br>HCU 22-23 accumulator pressure is 935 psig, down slow. |
| $T_0 + 15$ | SECOND accumulator trouble alarm is received on Control Rod 42-15.<br>Control Rod 42-15 is at position 48.<br>HCU 42-15 accumulator pressure is 935 psig, down slow. |

Which one of the following describes the potential impact of these low pressures and the action REQUIRED by ON-155-007, Loss of CRD System Flow?

| | Potential Impact | Required Action |
|---|---|---|
| A. | Drifting control rods | Place the Reactor Mode Switch in SHUTDOWN immediately at time $T_0$ + 15. |
| B. | Loss of scram capability | Place the Reactor Mode Switch in SHUTDOWN immediately at time $T_0$ + 15. |
| C. | Drifting control rods | Restore charging water header pressure >940 psig or place the Reactor Mode Switch in SHUTDOWN by time $T_0$ + 35 minutes. |
| D. | Loss of scram capability | Restore charging water header pressure >940 psig or place the Reactor Mode Switch in SHUTDOWN by time $T_0$ + 35 minutes. |

| | |
|---|---|
| Proposed Answer | **D** |
| Applicant References | **None** |
| Explanation | **The concern with low accumulator pressure is loss of scram capability. With Reactor pressure greater than 900 psig, a Reactor scram is required within 20 minutes of receipt of a second accumulator alarm (for a control rod that is not fully inserted) unless charging water header pressure can be restored.** |

**A** **Incorrect – The concern is loss of scram capability, not drifting control rods. Drifting control rods is the concern if there was low scram air header pressure. A scram is not required immediately, just within 20 minutes if charging water header pressure cannot be restored.**

**B** **Incorrect – The concern is loss of scram capability, not drifting control rods. Drifting control rods is the concern if there was low scram air header pressure.**

**C** **Incorrect –A scram is not required immediately, just within 20 minutes if charging water header pressure cannot be restored. A scram would be required immediately if Reactor pressure was <900 psig.**

**D** **Correct.**

| | |
|---|---|
| 10CFR55 | **41.10** |
| Technical References | **ON-155-007, Technical Specification 3.1.5** |
| Learning Objectives | **2436.I** |
| Question Source | **Modified Bank LOC25 Cert #34** |
| Previous NRC Exam | **No** |
| Comments | |

| Exam | RO | Tier | 2 | Group | 2 | Cognitive Level | Low | Level of Difficulty | 2.5 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 201002 A3.03 RMCS | | | | | Importance | 3.2 | | |
| Statement | Ability to monitor automatic operations of the REACTOR MANUAL CONTROL SYSTEM including: Rod drift alarm | | | | | | | | |

QUESTION 35

Which one of the following describes:

(1) an operation that may cause annunciator AR-104-H05, ROD DRIFT, to alarm, and

(2) when this annunciator will clear after being received?

| | (1) Moving a control rod with... | (2) AR-104-H05 will clear... |
|---|---|---|
| A. | CONT ROD INSERT pushbutton. | only when manually reset. |
| B. | CONT ROD INSERT pushbutton. | automatically once the control rod settles. |
| C. | CONT W/DRAW ROD pushbutton. | only when manually reset. |
| D. | CONT W/DRAW ROD pushbutton. | automatically once the control rod settles. |

Proposed Answer **A**

Applicant References **None**

Explanation **OP-156-001 cautions that use of the CONT ROD INSERT pushbutton may cause a control rod drift alarm because the Rod Motion Timer is bypassed while the rod is moving through odd reed switch positions. The control rod drift alarm seals-in and will not clear until manually reset.**

**A** **Correct.**

**B** **Incorrect - The control rod drift alarm seals-in and will not clear until manually reset, even once the initiating condition clears.**

**C** **Incorrect – Use of the CONT W/DRAW ROD pushbutton does not bypass the Rod Motion Timer, therefore it will not result is a control rod drift alarm.**

**D** **Incorrect - Use of the CONT W/DRAW ROD pushbutton does not bypass the Rod Motion Timer, therefore it will not result is a control rod drift alarm. The control rod drift alarm seals-in and will not clear until manually reset, even once the initiating condition clears.**

10CFR55 **41.6**

Technical References **OP-156-001 (page 19 of 22), AR-104-H05, ON-155-001**

Learning Objectives **2469.c**

Question Source **Modified Bank** **LOC25 NRC #35**

Previous NRC Exam **No**

Comments

| Exam | RO | Tier | 2 | Group | 2 | Cognitive Level | High | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | | 214000 A4.02 RPIS | | | | Importance | | 3.8 | |
| Statement | | Ability to manually operate and/or monitor in the control room: Control rod position | | | | | | | |

QUESTION 36

Unit 1 is in Mode 2 with the following:

- A control rod has just been withdrawn to position 48 and a coupling check is being performed.
- Annunciator AR-104-H06, ROD OVERTRAVEL, alarms.
- The control rod FULL OUT indicating light is extinguished on the Full Core Display.
- The Four Rod Display shows blank-blank for the selected control rod.

Which one of the following describes the meaning of these indications?

A. The RMCS settle function has failed.

B. The reed switch for position 48 has failed.

C. The position of the control rod blade is unknown.

D. The position of the control rod drive mechanism is unknown.

Proposed Answer **C**

Applicant References **None**

Explanation **Receipt of the ROD OVERTRAVEL alarm along with the FULL OUT light extinguishing and blank control rod position indication indicates that the control rod is uncoupled from the control rod drive mechanism. Therefore, while it is known that the control rod drive mechanism is slightly beyond the full out position, it is unknown where the control rod blade is.**

**A Incorrect – With a coupling check being performed, the control rod is being given a continuous withdraw signal, which is preventing the RMCS settle function on purpose.**

**B Incorrect – With a ROD OVERTRAVEL alarm in, the control rod drive mechanism has moved beyond the position 48 reed switch. The blank-blank rod position indication is appropriate for this condition.**

**C Correct.**

**D Incorrect – The position of the control rod blade is unknown due to the uncoupled condition, however the position of the drive mechanism is known (slightly beyond full out).**

10CFR55 **41.6**

Technical References **AR-104-H06, ON-155-001, TM-OP-56A**

Learning Objectives **2472.j**

Question Source **Bank** **LOC25 Cert #36**

Previous NRC Exam **No**

Comments

| Exam | RO | Tier | 2 | Group | 2 | Cognitive Level | Low | Level of Difficulty | 3.5 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | | 204000 2.4.35 RWCU | | | | Importance | | 3.8 | |
| Statement | | Knowledge of local auxiliary operator tasks during emergency and the resultant operational effects. | | | | | | | |

QUESTION 37

Unit 1 is operating at 25% power with the following:

- Both Standby Liquid Control pumps are inoperable and unavailable.
- Then, all MSIVs spuriously close.
- A failure to scram occurs.
- Initial ATWS power is 4%.
- Reactor water level is controlled between +20" and +45" with HPCI.
- Suppression Pool temperature is 145°F, up slow.
- The Unit Supervisor directs injecting boron per ES-150-002, Boron Injection Via RCIC.
- Reactor Water Cleanup is currently in service.

Which one of the following describes how ES-150-002 is executed and the control of the Reactor Water Cleanup (RWCU) system per ES-150-002 and EO-100-113, Level/Power Control?

| | ES-150-002 | RWCU... |
|---|---|---|
| A. | All required actions are completed by field operators. | automatically isolates based on actions taken in ES-150-002. |
| B. | All required actions are completed by field operators. | must be manually isolated or filter/demins manually bypassed. |
| C. | Some required actions are completed by field operators and some required actions are completed by Control Room personnel. | automatically isolates based on actions taken in ES-150-002. |
| D. | Some required actions are completed by field operators and some required actions are completed by Control Room personnel. | must be manually isolated or filter/demins manually bypassed. |

Proposed Answer **D**

Applicant References **None**

Explanation

**ES-150-002 has many required actions in the field to align the SBLC tank to the suction of the RCIC pump. ES-150-002 also has required actions to be taken by Control Room personnel to control RCIC and RWCU.**

**Boron injection is required even with initial ATWS power <5% before Suppression Pool temperature reaches 150°F. RWCU is required to be controlled to prevent removing boron after it is injected. With normal SBLC injection, RWCU is automatically isolated. However, boron injection with RCIC does not automatically isolate RWCU. ES-150-002 does not contain any field actions to manually isolate RWCU. ES-150-002 requires the NPOs to contact the Control Room to ensure RWCU is either isolated or the Filter/Demins are bypassed and isolated.**

**A** **Incorrect – Although ES-150-002 will initially be handed off to field operators to align the SBLC tank to RCIC, they will then need Control Room personnel to take actions to initiate injection. Nothing done in ES-150-002 will cause RWCU to automatically isolate, such as happens when boron is injected with SBLC.**

**B** **Incorrect – Although ES-150-002 will initially be handed off to field operators to align the SBLC tank to RCIC, they will then need Control Room personnel to take actions to initiate injection.**

**C** **Incorrect – Nothing done in ES-150-002 will cause RWCU to automatically isolate, such as happens when boron is injected with SBLC.**

**D** **Correct.**

| | |
|---|---|
| 10CFR55 | **41.10** |
| Technical References | **ES-150-002, EO-100-113** |
| Learning Objectives | |
| Question Source | **New** |
| Previous NRC Exam | **No** |
| Comments | |

| Exam | RO | Tier | 2 | Group | 2 | Cognitive Level | Low | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 286000 K1.08 Fire Protection | | | | | Importance | 3.0 | | |
| Statement | Knowledge of the physical connections and/or cause-effect relationships between FIRE PROTECTION SYSTEM and the following: Intake canals: Plant-Specific | | | | | | | | |

QUESTION 38

Which one of the following describes the normal water supply to the Motor Driven Fire Pump (0P512) and the Diesel Engine Driven Fire Pump (0P511)?

These pumps are normally aligned to take suction from the...

A. Well Water Storage Tank, only.

B. Clarified Water Storage Tank, only.

C. Unit 1 and Unit 2 Cooling Tower basins, only.

D. Clarified Water Storage Tank and one of the Cooling Tower basins, only.

Proposed Answer **D**

Applicant References **None**

Explanation **0P512 and 0P511 are normally aligned to take suction from the Clarified Water Storage Tank and one of the Cooling Water basins, only.**

**A Incorrect – The Backup Fire Protection System is supplied by the Well Water Storage Tank only, but the main Fire Protection System is supplied by the Clarified Water Storage Tank and one of the Cooling Tower basins during normal alignment.**

**B Incorrect – 0P512 and 0P511 are normally aligned to take suction from the Clarified Water Storage Tank and also one of the Cooling Tower basins.**

**C Incorrect – 0P512 and 0P511 are normally aligned to take suction from one of the Cooling Tower basins, as well as the Clarified Water Storage Tank.**

**D Correct.**

10CFR55 **41.4**

Technical References **TM-OP-013**

Learning Objectives **2293.b**

Question Source **New**

Previous NRC Exam **No**

Comments

| Exam | RO | Tier | 1 | Group | 1 | Cognitive Level | Low | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 295026 EK2.01 Suppression Pool High Water Temperature | | | Importance | 3.9 | | | | |
| Statement | Knowledge of the interrelations between SUPPRESSION POOL HIGH WATER TEMPERATURE and the following: Suppression pool cooling | | | | | | | | |

QUESTION 39

Unit 1 has experienced a transient with the following:

- The Reactor is scrammed.
- Reactor water level is being controlled between +20" and +45" with HPCI.
- Suppression Pool water temperature is 85°F, up slow.
- The Unit Supervisor has directed placing one loop of RHR in the Suppression Pool Cooling mode per OP-149-005, RHR Suppression Pool Cooling.

Which one of the following identifies the preferred loop of RHR to be placed in Suppression Pool Cooling mode and the maximum allowed total RHR loop flow rate, in accordance with OP-149-001?

| | Preferred RHR Loop | Maximum RHR Loop Flow Rate |
|---|---|---|
| A. | RHR Loop A | 6,000 gpm |
| B. | RHR Loop A | 10,000 gpm |
| C. | RHR Loop B | 6,000 gpm |
| D. | RHR Loop B | 10,000 gpm |

| | |
|---|---|
| Proposed Answer | **D** |
| Applicant References | **None** |
| Explanation | **OP-149-005 identifies RHR Loop B as the preferred loop when HPCI is in operation due to location of HPCI exhaust near the RHR Loop B suction. OP-149-005 requires the total RHR loop flow rate to be ≤10,000 gpm to prevent damage to the RHR heat exchanger.** |

**A** **Incorrect – RHR Loop B is the preferred loop with HPCI in operation. The total RHR loop flow rate is required to be ≤10,000 gpm. 6,000 gpm is a limit used in OP-149-005 during slow fill of RHR for Suppression Pool Cooling from RSP.**

**B** **Incorrect – RHR Loop B is the preferred loop with HPCI in operation.**

**C** **Incorrect - The total RHR loop flow rate is required to be ≤10,000 gpm. 6,000 gpm is a limit used in OP-149-005 during slow fill of RHR for Suppression Pool Cooling from RSP.**

**D** **Correct.**

| | |
|---|---|
| 10CFR55 | **41.8** |
| Technical References | **OP-149-005** |
| Learning Objectives | **197.e** |
| Question Source | **New** |

| | |
|---|---|
| Previous NRC Exam | **No** |
| Comments | |

| Exam | RO | Tier | 1 | Group | 1 | Cognitive Level | High | Level of Difficulty | 3.5 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 295030 EK1.03 Low Suppression Pool Water Level | | | | | Importance | | 3.8 | |
| Statement | Knowledge of the operational implications of the following concepts as they apply to LOW SUPPRESSION POOL WATER LEVEL: Heat capacity | | | | | | | | |

QUESTION 40

Unit 1 has experienced an accident with the following:

- All control rods are at position 00.
- Reactor pressure is 700 psig, down slow.
- Suppression Pool level is 19', down slow.
- Average SPOTMOS indication is 184°F, up slow.
- Bottom SPOTMOS indication is 198°F, up slow.

**Note:** The Heat Capacity Temperature Limit (HCTL) is provided on the following page.

Which one of the following describes the status of the HCTL and the required control of Reactor pressure, in accordance with the Emergency Operating Procedures?

The HCTL is...

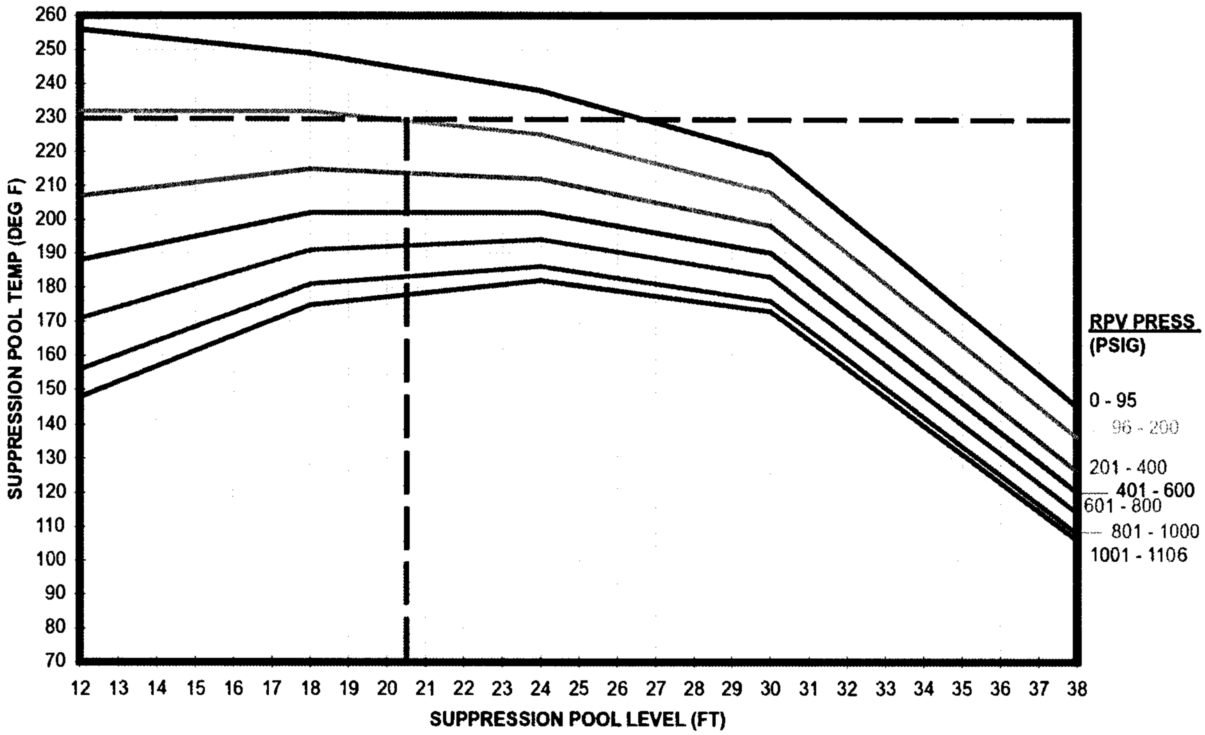
A. exceeded. A Rapid Depressurization is required.

B. exceeded. A Rapid Depressurization is NOT required.

C. NOT exceeded. The Reactor cooldown rate may exceed 100°F/hr.

D. NOT exceeded. The Reactor cooldown rate must be maintained less than 100°F/hr.

**FIG 2 HCTL**
**HEAT CAPACITY TEMPERATURE LIMIT**




| | |
|---|---|
| Proposed Answer | **A** |
| Applicant References | **None** |
| Explanation | **With Suppression Pool level <20.5 feet, the average SPOTMOS indication is no longer a reliable indicator of Suppression Pool water temperature. Bottom SPOTMOS indication must then be used to determine status of HCTL. Using bottom SPOTMOS indication, the HCTL is currently exceeded. Because HCTL is exceeded, EO-103-001 step SP/T-8 requires Rapid Depressurization.** |

**A** Correct.

**B** Incorrect – Prior to exceeding HCTL, action is allowed to lower Reactor pressure in an attempt to avoid exceeding HCTL. However, once the HCTL is exceeded, an attempt at restoration under the curve is not allowed. Rapid Depressurization is required.

**C** Incorrect – While average SPOTMOS indication is below the HCTL, this indication is the wrong one to use for Suppression Pool water temperature due to Suppression Pool level <20.5 feet. Using bottom SPOTMOS indication, the HCTL is exceeded.

**D** Incorrect - While average SPOTMOS indication is below the HCTL, this indication is the wrong one to use for Suppression Pool water temperature due to Suppression Pool level <20.5 feet. Using bottom SPOTMOS indication, the HCTL is exceeded.

| | |
|---|---|
| 10CFR55 | **41.10** |
| Technical References | **EO-100-103** |
| Learning Objectives | |
| Question Source | **New** |
| Previous NRC Exam | **No** |

Comments

| Exam | RO | Tier | 1 | Group | 1 | Cognitive Level | Low | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|

| | | | |
|---|---|---|---|
| K/A | 295018 AK1.01 Partial or Complete Loss of CCW | Importance | 3.5 |
| Statement | Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: Effects on component/system operations | | |

QUESTION 41

Unit 1 is operating at 50% power with the following:

- RBCCW temperature control is malfunctioning.
- RBCCW supply temperature is 92°F, up slow.
- Initial attempts to manually control RBCCW temperature from the Control Room have been unsuccessful.
- Operators in the field are attempting further actions to restore RBCCW temperature.

Which one of the following identifies the first system load that should be secured, if necessary, to lower heat load on RBCCW, in accordance with ON-114-001, Loss of RBCCW?

A. Fuel Pool Cooling and Cleanup Heat Exchangers

B. A Reactor Recirculation Pump

C. Reactor Water Cleanup

D. Offgas Chillers

Proposed Answer: C

Applicant References: None

Explanation: ON-114-001 gives guidance for first removing RWCU, since it is a major heat load and operation can continue for a significant period of time without RWCU in-service.

A Incorrect - ON-114-001 gives guidance for first removing RWCU, since it is a major heat load and operation can continue for a significant period of time without RWCU in-service. Additionally, the Fuel Pool Cooling and Cleanup Heat Exchangers are directly cooled by Service Water, not RBCCW.

B Incorrect - ON-114-001 gives guidance for first removing RWCU, since it is a major heat load and operation can continue for a significant period of time without RWCU in-service.

C Correct.

D Incorrect - ON-114-001 gives guidance for first removing RWCU, since it is a major heat load and operation can continue for a significant period of time without RWCU in-service.

10CFR55: 41.4

Technical References: ON-114-001

Learning Objectives: AD045 / 15304

Question Source: Bank LOC25 Cert #43

Previous NRC Exam **No**

Comments

| Exam | RO | Tier | 1 | Group | 1 | Cognitive Level | High | Level of Difficulty | 3.5 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | | 295005 AK2.02 Main Turbine Generator Trip | | | | Importance | | 2.9 | |
| Statement | | Knowledge of the interrelations between MAIN TURBINE GENERATOR TRIP and the following: Feedwater temperature | | | | | | | |

QUESTION 42

Unit 1 is operating at 18% power during a startup with the following:

- The Main Generator is on-line.
- Feedwater Heating is in service.
- Then, EHC fluid pressure lowers to 500 psig and then stabilizes.

Which one of the following describes (1) the effect on Main Turbine operation and (2) the status of Reactor power five minutes later?

| | (1) The Main Turbine... | (2) Five minutes later, Reactor power... |
|---|---|---|
| A. | trips. | is >18%. |
| B. | trips. | is <18%. |
| C. | remains on-line. | is > 18%. |
| D. | remains on-line. | is < 18%. |

Proposed Answer **A**

Applicant References **None**

Explanation **The Main Turbine trips due to EHC fluid pressure <1100 psig. With Reactor power <26%, the Reactor does not scram as a result of the Main Turbine trip. When the Main Turbine trips, steam flow is diverted through the Turbine Bypass Valves. This causes a loss of Feedwater heating. Feedwater temperature into the Reactor lowers which causes Reactor power to rise above the initial value of 18%. The expected rise is approximately 4% power, therefore the Reactor will remain <26% power and will not scram.**

**A Correct.**

**B Incorrect – The Feedwater heating loss causes Reactor power to rise. The Reactor will not scram because Reactor power is initially (and remains) <26%.**

**C Incorrect – The Main Turbine trips due to EHC fluid pressure <1100 psig.**

**D Incorrect – The given EHC fluid pressure is below the Main Turbine trip setpoint of 1100 psig. The Feedwater heating loss causes Reactor power to rise.**

10CFR55 **41.4**

Technical References **AR-105-C01, GO-100-004**

Learning Objectives **10341.a, 1641.l**

Question Source **Modified Bank JAF 2010 NRC #76**

Previous NRC Exam **No**

Comments

| Exam | RO | Tier | 1 | Group | 1 | Cognitive Level | High | Level of Difficulty | 2.5 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 295024 EK2.18 High Drywell Pressure | | | | | Importance | | 3.3 | |
| Statement | Knowledge of the interrelations between HIGH DRYWELL PRESSURE and the following: Ventilation | | | | | | | | |

QUESTION 43

Both Units are operating at 100% power with the following:

- Div 1 Control Structure HVAC fans are in service.
- Then, a steam leak develops inside the Unit 1 Drywell.
- Unit 1 Drywell pressure is 2.5 psig, up slow.

Which one of the following describes the response of Control Structure HVAC?

CREOASS automatically starts in the ___(1)___.

Control Structure HVAC Unit A Fan (0V103A) ___(2)___.

| | (1) | (2) |
|---|---|---|
| A. | Recirculation Mode | trips |
| B. | Recirculation Mode | remains in service |
| C. | Pressurization/Filtration Mode | trips |
| D. | Pressurization/Filtration Mode | remains in service |

Proposed Answer: D

Applicant References: None

Explanation: A high drywell pressure signal (>1.72 psig) causes CREOASS to auto start in the Pressurization/Filtration Mode. Fan 0V103A will not receive a trip signal and will continue to run, providing air flow from the discharge of the CREOASS system

A Incorrect – CREOASS automatically initiates in the Pressurization/Filtration Mode, not the Recirculation Mode. Certain Control Room exhaust fans trip, however 0V103A remains in service.

B Incorrect – CREOASS automatically initiates in the Pressurization/Filtration Mode, not the Recirculation Mode.

C Incorrect – Certain Control Room exhaust fans trip, however 0V103A remains in service.

D Correct.

10CFR55: 41.7

Technical References: ON-159-002, TM-OP-030

Learning Objectives: 10457.b

Question Source: Bank Vision SYSID 453

Previous NRC Exam: No

Comments

| Exam | RO | Tier | 1 | Group | 1 | Cognitive Level | High | Level of Difficulty | 2.5 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | | 295006 AK2.07 SCRAM | | | | Importance | | 4.0 | |
| Statement | | Knowledge of the interrelations between SCRAM and the following: Reactor pressure control | | | | | | | |

QUESTION 44

Unit 1 is operating at 100% power when power is lost to both Startup Transformer 10 (0X103) and Startup Transformer 20 (0X104).

Which one of the following describes the automatic Reactor pressure control response?

Reactor pressure will be controlled...

A. by automatic cycling of the Safety Relief Valves.

B. at approximately 934 psig by the Turbine Bypass Valves.

C. at approximately 1040 psig by the Turbine Bypass Valves.

D. at approximately 1040 psig by the Turbine Control Valves.

Proposed Answer: **A**

Applicant References: **None**

Explanation: **Loss of power to both Startup Transformers 10 and 20 results in an automatic Reactor scram. MSIVs also automatically close due to the loss of power. With MSIVs closed, Reactor pressure will rise until SRVs automatically cycle.**

**A Correct.**

**B Incorrect – The Reactor does scram on this loss of power. Turbine Bypass Valves normally control Reactor pressure ~934 psig following a scram. However, Turbine Bypass Valves are unavailable in this situation due to automatic closure of the MSIVs due to the power loss.**

**C Incorrect - The Reactor does scram on this loss of power. Turbine Bypass Valves normally control Reactor pressure following a scram. 1040 psig is normal Reactor pressure at 100% power. However, Turbine Bypass Valves are unavailable in this situation due to automatic closure of the MSIVs due to the power loss.**

**D Incorrect – This loss of power results in an automatic Reactor scram. If no Reactor scram occurred, Reactor pressure would continue to be controlled ~1040 psig by the Turbine Control Valves.**

10CFR55: **41.5**

Technical References: **ON-104-001**

Learning Objectives: **5305.a**

Question Source: **Modified Bank JAF 3/12 NRC #5**

Previous NRC Exam: **No**

Comments

| Exam | RO | Tier | 1 | Group | 1 | Cognitive Level | Low | Level of Difficulty | 2.5 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 295038 EK3.04 High Off-site Release Rate | | | | | Importance | 3.6 | | |
| Statement | Knowledge of the reasons for the following responses as they apply to HIGH OFF-SITE RELEASE RATE: Emergency depressurization | | | | | | | | |

QUESTION 45

EO-100-105, Radioactivity Release Control, contains the following step:

RR-6 **BEFORE** EPB DOSE RATE/PROJECTED DOSE REACHES ________________ DECLARATION CRITERIA

RAPID DEPRESS IS REQ'D

Which one of the following identifies (1) the Emergency Action Level that correctly fills in the blank and (2) the reason for the Rapid Depressurization, in accordance with EO-100-105?

A. (1) Alert
(2) Limit dose to the general public.

B. (1) Alert
(2) Maintain Control Room habitability.

C. (1) General Emergency
(2) Limit dose to the general public.

D. (1) General Emergency
(2) Maintain Control Room habitability.

Proposed Answer **C**

Applicant References **None**

Explanation **EO-000-105 requires rapid depressurization before EPB dose rates exceed the General Emergency EAL threshold. The specific basis is to limit dose to the general public.**

**A Incorrect - EO-000-105 requires rapid depressurization before EPB dose rates exceed the General Emergency EAL threshold. The Alert level is used for entry into EO-100-105. The specific basis is to limit dose to the general public. Control Room habitability is a concern during a release and forms the basis for Control Room HVAC and shielding design.**

**B Incorrect - EO-000-105 requires rapid depressurization before EPB dose rates exceed the General Emergency EAL threshold. The Alert level is used for entry into EO-100-105.**

**C Correct.**

**D Incorrect – The specific basis is to limit dose to the general public. Control Room habitability is a concern during a release and forms the basis for Control Room HVAC and shielding design.**

10CFR55 **41.5**

Technical References **EO-000-105**

| | | |
|---|---|---|
| Learning Objectives | **PP002 / 14613** | |
| Question Source | **Bank** | **LOC25 Cert #47** |
| Previous NRC Exam | **No** | |
| Comments | | |

| Exam | RO | Tier | 1 | Group | 1 | Cognitive Level | Low | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | **295001 AK3.03 Partial or Complete Loss of Forced Core Flow Circulation** | | | | | Importance | | **2.8** | |
| Statement | **Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Idle loop flow** | | | | | | | | |

## QUESTION 46

Unit 1 is operating at 50% power with the following:

- Recirculation pump 1A has spuriously tripped.
- All actions in ON-164-002, Loss of Reactor Recirculation Flow, have been completed.

Which one of the following describes the status of flow through Recirculation loop 1A and the reason why flow is controlled this way?

Flow through Recirculation loop 1A...

A. is maintained to keep the loop warm.

B. is terminated to prevent a cold water addition.

C. is terminated to ensure accurate core flow indications.

D. is maintained to ensure accurate core flow indications.

Proposed Answer **A**

Applicant References **None**

Explanation **ON-164-002 maintains flow through the idle Recirculation loop to ensure the loop remains warm.**

**A Correct.**

**B Incorrect – While the idle pump's discharge valve is cycled closed for a period of time during performance of ON-164-002, flow is maintained through the discharge bypass valve and the discharge valve is eventually re-opened.**

**C Incorrect – While the idle pump's discharge valve is cycled closed for a period of time during performance of ON-164-002, flow is maintained through the discharge bypass valve and the discharge valve is eventually re-opened.**

**D Incorrect – Core flow indications are of concern following a Recirc pump trip due to idle loop flow, but are not the basis for why flow is maintained through the idle loop. Flow is maintained to keep the loop warm.**

10CFR55 **41.10**

Technical References **ON-164-002**

Learning Objectives **2512.b**

Question Source **New**

Previous NRC Exam **No**

Comments

| Exam | RO | Tier | 1 | Group | 1 | Cognitive Level | Low | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 295003 AK3.03 Partial or Complete Loss of AC Power | | | | | Importance | | 3.5 | |
| Statement | Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: Load shedding | | | | | | | | |

QUESTION 47

During a Station Blackout, Unit 1 is required per EO-100-030 to open 250 VDC Load Center Breakers to 1D155 and 1D165.

On Unit 2, EO-200-030 does NOT direct the same actions for 2D155 and 2D156.

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

A. The Non-Vital Loads that are shed on Unit 1 have a separate Non-Vital Battery Bank on Unit 2.

B. The Unit 2 loads are powered by the Portable Diesel Generator (Blue Max) installation per ES-002-001.

C. The 250 VDC Batteries on Unit 2 have more storage capacity than the associated batteries on Unit 1.

D. The Unit 2 Vital Loads are NOT shed since they are needed to cope with the Station Blackout per EO-200-030.

| | |
|---|---|
| Proposed Answer | **A** |
| Applicant References | **None** |
| Explanation | **The equivalent loads on Unit 2 are supplied by non-1E battery 2D140, and therefore do not require load shedding to preserve battery capacity as on Unit 1. Unit 1 does not have this extra non-1E battery.** |
| | **A Correct.** |
| | **B Incorrect – These loads are not supplied by the Blue Max.** |
| | **C Incorrect - The batteries have equal capacities.** |
| | **D Incorrect – These loads are not required for coping strategy.** |
| 10CFR55 | **41.10** |
| Technical References | **EO-100-030, EO-200-030** |
| Learning Objectives | **PP002 / 2679** |
| Question Source | **Bank LOC25 Cert #39** |
| Previous NRC Exam | **No** |
| Comments | |

| Exam | RO | Tier | 1 | Group | 1 | Cognitive Level | Low | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | | 295028 EA1.04 High Drywell Temperature | | | | Importance | | 3.9 | |
| Statement | | Ability to operate and/or monitor the following as they apply to HIGH DRYWELL TEMPERATURE: Drywell pressure | | | | | | | |

QUESTION 48

Unit 1 has experienced a steam leak inside the Drywell with the following:

- Drywell temperature is 320°F, up fast.
- Drywell pressure is 12 psig, up slow.
- Suppression Chamber pressure is 10 psig, up slow.
- The crew has determined that Drywell sprays are required.

Which one of the following describes a limitation that the crew must observe based on the above stated conditions, and the basis for this limitation, in accordance with EO-100-103, Primary Containment Control?

A. Limit initial Drywell spray flow to between 1000 and 2800 gpm to prevent cyclic condensation of steam at the downcomer openings of the Drywell vents.

B. Do NOT start Drywell sprays until Suppression Chamber pressure exceeds 13 psig to prevent cyclic condensation of steam at the downcomer openings of the Drywell vents.

C. Limit initial Drywell spray flow to between 1000 and 2800 gpm to prevent an excessive pressure drop that could damage primary containment internal components and structures.

D. Do NOT start Drywell sprays until Suppression Chamber pressure exceeds 13 psig to prevent an excessive pressure drop that could damage primary containment internal components and structures.

Proposed Answer: **C**

Applicant References: **None**

Explanation: **EO-100-103 step DW/T-6 requires Drywell sprays before Drywell temperature reaches 340°F and the crew has determined Drywell sprays are necessary. This step requires limiting Drywell spray flow to between 1000 gpm and 2800 gpm for the first 30 seconds. The basis of this limitation is to prevent a rapid, excessive evaporative cooling pressure drop in the Drywell that could result in damage to the Primary Containment.**

**A Incorrect – Cyclic condensation of steam at the downcomer openings is the reason for requiring initiation of Drywell sprays when Suppression Chamber pressure reaches 13 psig, not the reason for limiting initial Drywell spray flow rate.**

**B Incorrect – While Drywell sprays are not required in the DW/P leg of EO-100-103 until Suppression Chamber pressure reaches 13 psig, nothing prevents initiating Drywell sprays before this pressure if needed for Drywell temperature control. Cyclic condensation of steam at the downcomer openings is the reason for requiring initiation of Drywell sprays when Suppression Chamber pressure reaches 13 psig, not the reason for limiting initial Drywell spray flow rate.**

**C Correct.**

**D Incorrect – While Drywell sprays are not required in the DW/P leg of EO-100-103 until Suppression Chamber pressure reaches 13 psig, nothing prevents initiating Drywell sprays before this pressure if needed for Drywell temperature control.**

| | | |
|---|---|---|
| 10CFR55 | **41.10** | |
| Technical References | **EO-100-103** | |
| Learning Objectives | **14613** | |
| Question Source | **Bank** | **LOC23 NRC #44** |
| Previous NRC Exam | **Yes** | |
| Comments | | |

| Exam | RO | Tier | 1 | Group | 1 | Cognitive Level | Low | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | | 295037 EA1.01 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown | | | | Importance | | 4.6 | |
| Statement | | Ability to operate and/or monitor the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: Reactor Protection System | | | | | | | |

QUESTION 49

A hydraulic failure to scram has occurred with the following:

- Reactor power is 10%, down slow.
- Reactor water level is being controlled between -60" and -110".
- Control rods are to be inserted using repeated manual scrams.
- ARI must be disabled and RPS logic trips must be bypassed in accordance with ES-158-002, RPS and ARI Trip Bypass.

Which one of the following describes the methods to disable ARI and bypass RPS logic trips in accordance with ES-158-002?

| | Disable ARI by... | Bypass RPS logic trips by... |
|---|---|---|
| A. | opening breakers. | installing jumpers. |
| B. | opening breakers. | re-positioning keylock switches. |
| C. | re-positioning keylock switches. | installing jumpers. |
| D. | re-positioning keylock switches. | re-positioning keylock switches. |

Proposed Answer **A**

Applicant References **None**

Explanation **ES-158-002 disables ARI by opening two breakers to prevent ARI vent valves from opening and ARI block valves from closing. ES-158-002 bypasses RPS logic trips by installing jumpers around specific RPS logic contacts.**

**A Correct.**

**B Incorrect – Keylock switches are provided for bypassing some RPS logic trips (SDV high level), however ES-158-002 bypasses all RPS logic trips other than the manual scram by installing jumpers.**

**C Incorrect – Keylock switches are provided in the Reactor Building that allow bypassing ARI logic for testing, however ES-158-002 does not use these switches but opens breakers to disable ARI.**

**D Incorrect - Keylock switches are provided in the Reactor Building that allow bypassing ARI logic for testing, however ES-158-002 does not use these switches but opens breakers to disable ARI. Keylock switches are provided for bypassing some RPS logic trips (SDV high level), however ES-158-002 bypasses all RPS logic trips other than the manual scram by installing jumpers.**

10CFR55 **41.7**

| | |
|---|---|
| Technical References | **ES-158-002** |
| Learning Objectives | **2486.o** |
| Question Source | **Modified Bank** **NMP1 2010 Audit #43** |
| Previous NRC Exam | **No** |
| Comments | |

| Exam | RO | Tier | 1 | Group | 1 | Cognitive Level | High | Level of Difficulty | 3.5 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | | **295023 AA1.08 Refueling Accidents** | | | | Importance | | **3.4** | |
| Statement | | **Ability to operate and/or monitor the following as they apply to REFUELING ACCIDENTS: Radiation monitoring equipment** | | | | | | | |

QUESTION 50

Unit 1 is in a refueling outage with the following:

- Annunciator AR-101-E05, SPENT FUEL POOL AREA HI RADIATION, alarms.
- Area Radiation Monitors (ARMs) 14 and 47, SPENT FUEL CRIT MON, are both in alarm high at 150 mR/hr, but below their Max Safe radiation level.

Which one of the following describes the need to enter EO-100-104, Secondary Containment Control, and the behavior of annunciator AR-101-E05 once these ARMs lower below their high setpoints?

EO-100-104 entry is ___(1)___. When these ARMs lower below their high setpoints, AR-101-E05 will ___(2)___.

| | (1) | (2) |
|---|---|---|
| A. | required | automatically reset. |
| B. | NOT required | automatically reset. |
| C. | required | remain in alarm until the ARMs are manually reset. |
| D. | NOT required | remain in alarm until the ARMs are manually reset. |

Proposed Answer **C**

Applicant References **None**

Explanation **EO-100-104 entry is required based on unexplained Reactor Building area radiation level above the high alarm (100 mR/hr for these ARMs). Annunciator AR-101-E05 will remain in alarm even after the ARM returns to below the high setpoint. An operator must manually reset the ARMs to clear the annunciator.**

**A Incorrect - Annunciator AR-101-E05 will remain in alarm even after the ARM returns to below the high setpoint. An operator must manually reset the ARMs to clear the annunciator.**

**B Incorrect – EO-100-104 entry is required with the ARMs above the high alarm, even when still below the Max Safe level. The Max Safe level is used to determine the need for subsequent actions in EO-100-104.**

**C Correct.**

D Incorrect - EO-100-104 entry is required with the ARMs above the high alarm, even when still below the Max Safe level. The Max Safe level is used to determine the need for subsequent actions in EO-100-104. Annunciator AR-101-E05 will remain in alarm even after the ARM returns to below the high setpoint. An operator must manually reset the ARMs to clear the annunciator.

| | |
|---|---|
| 10CFR55 | **41.11** |
| Technical References | **AR-101-E05, EO-100-104, TM-OP-79, TM-OP-079B (pages 9 & 10)** |
| Learning Objectives | **1181.c** |
| Question Source | **Modified Bank** **Vision SYSID 116** |
| Previous NRC Exam | **No** |
| Comments | |

| Exam | RO | Tier | 1 | Group | 1 | Cognitive Level | Low | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 600000 AA2.16 Plant Fire On-site | | | | | Importance | 3.0 | | |
| Statement | Ability to determine and interpret the following as they apply to PLANT FIRE ON SITE: Vital equipment and control systems to be maintained and operated during a fire | | | | | | | | |

QUESTION 51

Unit 1 is operating at 100% power with the following:

- A fire occurs outside the Control Room.
- ON-013-001, Response to Fire, is being performed.
- The Reactor is scrammed in accordance with ON-100-101, Scram, Scram Imminent.
- The RCIC turbine initiates and CANNOT be overridden or isolated.
- Reactor water level is 56", up fast.

Which one of the following describes the required action and the reason for this action in accordance with ON-013-001?

A. Close the MSIVs to preserve Reactor coolant inventory.

B. Close the MSIVs to prevent damage to the Main Steam Lines.

C. Depressurize the RPV to prevent damage to the RCIC turbine.

D. Depressurize the RPV to prevent damage to the SRV tailpipes.

Proposed Answer **C**

Applicant References **None**

Explanation **ON-013-001 provides specific guidance to depressurize the Reactor before Reactor water level reaches +118" (Main Steam Lines) if RCIC injects and cannot be either overridden or isolated. The reason for this action is to prevent damage to the RCIC turbine.**

**A Incorrect – The MSIVs are NOT closed, but are relied upon as part of the Appendix R safe shutdown paths. Depressurization is required before Reactor water level reaches the level of the Main Steam Lines (+118").**

**B Incorrect – The MSIVs are NOT closed, but are relied upon as part of the Appendix R safe shutdown paths. Depressurization is required before Reactor water level reaches the level of the Main Steam Lines (+118").**

**C Correct.**

**D Incorrect – While high Reactor water level could cause water flow through the SRV tailpipes, a calculation has been performed that proves the tailpipes will not be damaged by this water flow.**

10CFR55 **41.10**

Technical References **ON-013-001**

Learning Objectives **2015.e**

Question Source **Bank** **LOC24 NRC #19**

Previous NRC Exam **Yes**

Comments

| Exam | RO | Tier | 1 | Group | 1 | Cognitive Level | High | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 295016 AA2.06 Control Room Abandonment | | | | | Importance | 3.3 | | |
| Statement | Ability to determine and/or interpret the following as they apply to CONTROL ROOM ABANDONMENT: Cooldown rate | | | | | | | | |

QUESTION 52

Unit 1 is operating at 80% power with the following:

- A fire occurs requiring Control Room evacuation per ON-100-009, Control Room Evacuation.
- The Reactor is manually scrammed.
- A cooldown is initiated from the Remote Shutdown Panel at time $T_0$.
- Reactor pressure responds as follows:

| Time | Reactor Pressure |
|---|---|
| $T_0$ | 1025 psig |
| $T_0$ + 60 minutes | 475 psig |

**Note:** ON-100-009 Attachment A, Pressure Vs. Temperature for Saturated Steam, is provided on the following page.

Which one of the following describes the status of the cooldown rate in accordance with ON-100-109?

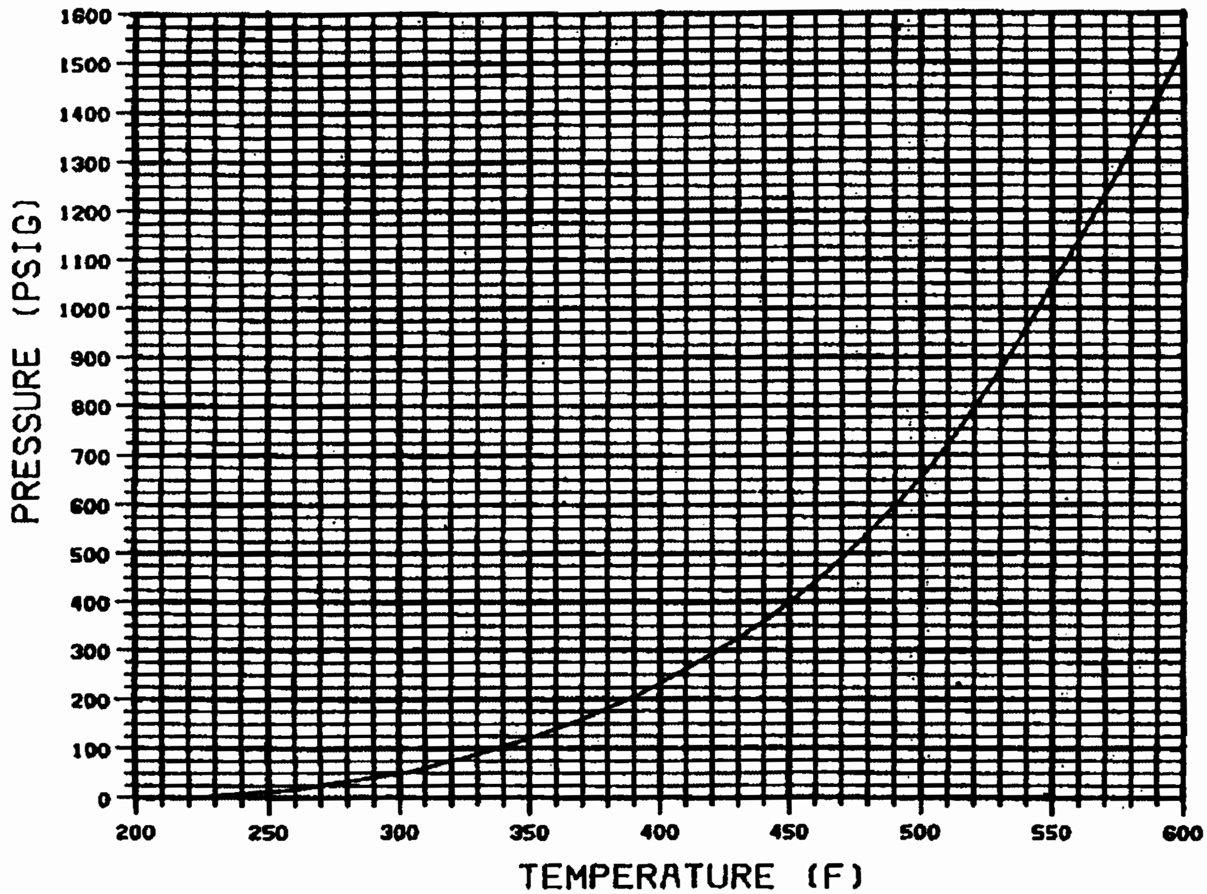
The cooldown rate is...

A. less than 100°F/hr and acceptable.

B. greater than 100°F/hr and acceptable.

C. less than 100°F/hr and NOT acceptable.

D. greater than 100°F/hr and NOT acceptable.




| | |
|---|---|
| Proposed Answer | A |
| Applicant References | None |
| Explanation | The given Reactor pressures indicate that temperature has lowered from approximately 550°F to approximately 465°F in one hour. This is a cooldown rate of approximately 85°F/hr and is less than 100°F/hr. ON-100-109 assumes that EO-100-102 is entered concurrently, and therefore the cooldown rate limit is 100°F/hr. Therefore, the current cooldown rate is acceptable. |

A Correct.

B Incorrect – The cooldown rate for the first hour is less than 100°F/hr.

C Incorrect – Even under the given Control Room evacuation conditions, the cooldown rate is controlled to under 100°F/hr. Therefore, the current cooldown rate is acceptable. The cooldown rate would be unacceptable if ON-100-109 required a rapid depressurization by opening SRVs and leaving them open.

D Incorrect – The cooldown rate for the first hour is less than 100°F/hr.

| | |
|---|---|
| 10CFR55 | 41.5 |
| Technical References | ON-100-109, EO-100-102 |
| Learning Objectives | AD045 / 15307 |
| Question Source | New |
| Previous NRC Exam | No |
| Comments | |

| Exam | RO | Tier | 1 | Group | 1 | Cognitive Level | High | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|

| K/A | 700000 AA2.04 Generator Voltage and Electric Grid Disturbances | Importance | 3.6 |
|---|---|---|---|
| Statement | Ability to determine and/or interpret the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: VARs outside capability curve | | |

QUESTION 53

Unit 1 is operating at 100% power with the following:

- Grid instabilities are causing Main Generator reactive load to rise.
- ON-103-001, Grid Instabilities, has been entered.
- The Main Generator voltage regulator is in AUTO.
- Both Reactor Recirculation pumps are being controlled in the MANUAL mode.
- Main Generator output is 1150 MWe, stable.
- Main Generator reactive load is 500 MVAR lagging, up slow.
- Main Generator hydrogen pressure is 70 psig, stable.

**Note:** The Generator Capability Curve is provided on the following pages.

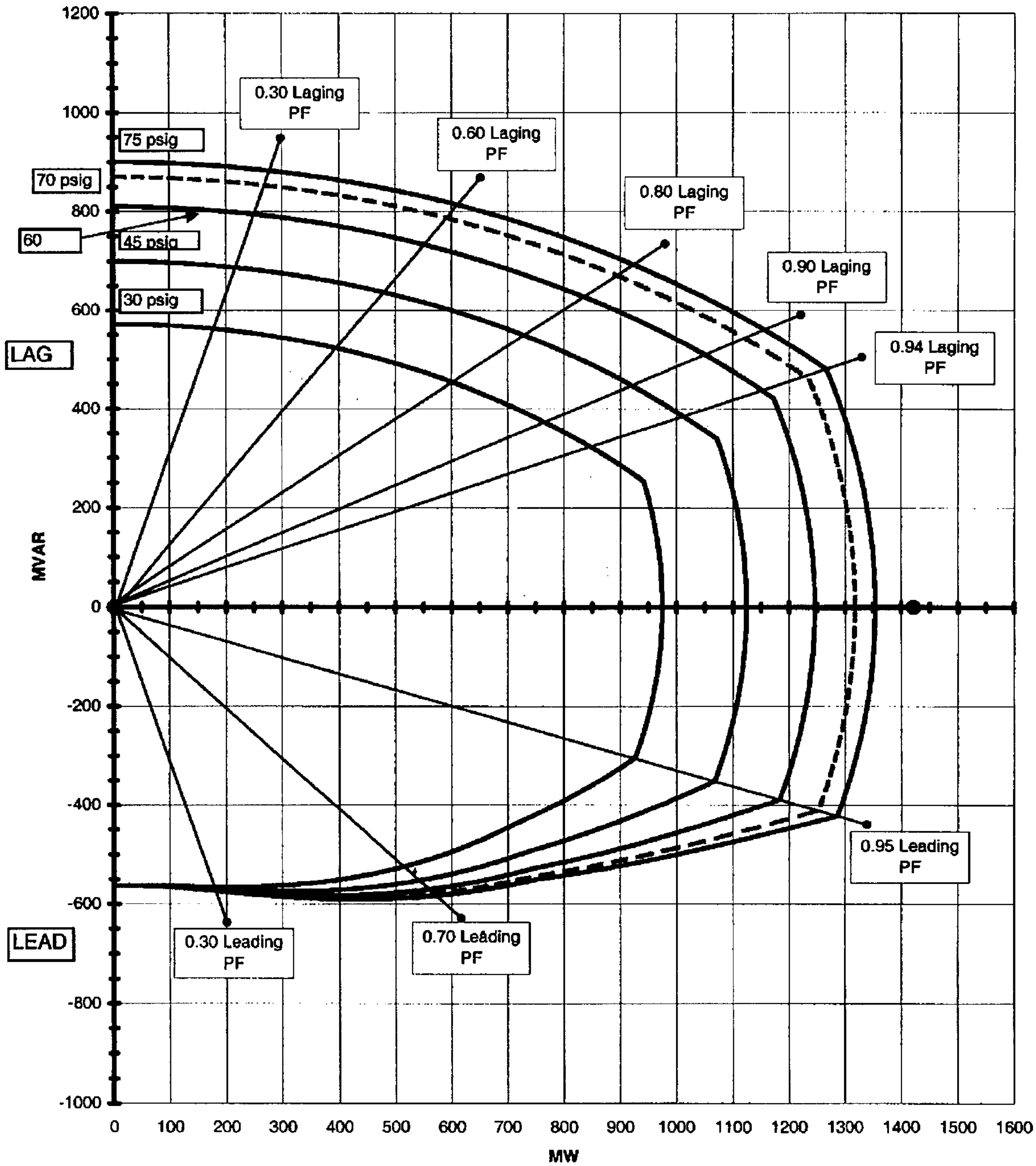
Which one of the following identifies (1) the approximate limit on reactive load for these conditions and (2) the action required **if this limit is approached or exceeded**, in accordance with ON-103-001?

A. (1) 520 MVAR lagging
(2) Immediately lower MVARs using the Main Generator voltage regulator.

B. (1) 520 MVAR lagging
(2) Notify TCC and wait up to two minutes for the AUTO voltage regulator to adjust before taking manual action to lower MVARs.

C. (1) 565 MVAR lagging
(2) Immediately lower MVARs using the Main Generator voltage regulator.

D. (1) 565 MVAR lagging
(2) Notify TCC and wait up to two minutes for the AUTO voltage regulator to adjust before taking manual action to lower MVARs.


0.30 Laging PF
0.60 Laging PF
0.80 Laging PF
0.90 Laging PF
0.94 Laging PF
0.95 Leading PF
0.30 Leading PF
0.70 Leading PF
75 psig
70 psig
60
45 psig
30 psig
LAG
LEAD
MVAR
MW
1200
1000
800
600
400
200
0
-200
-400
-600
-800
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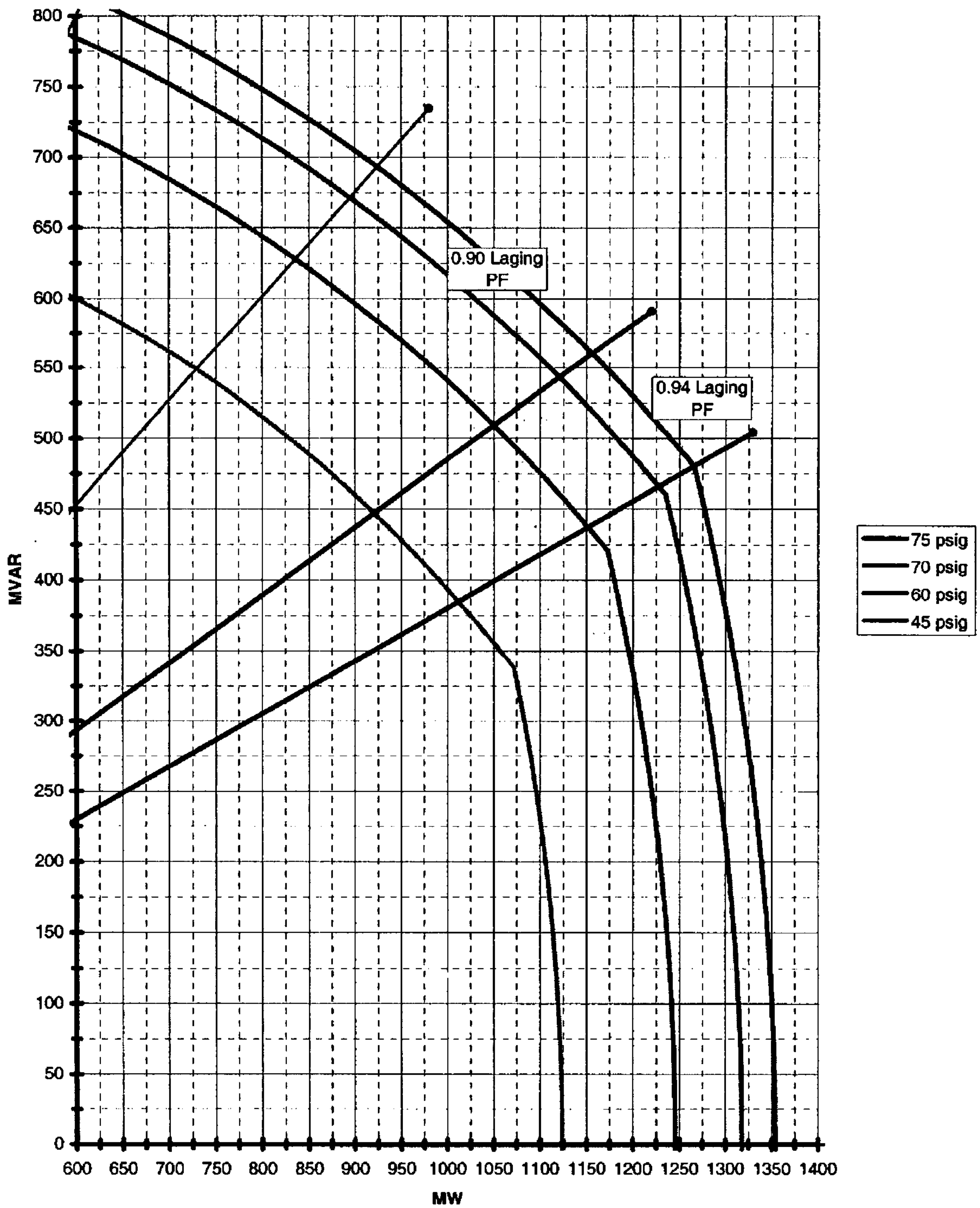
1600




| | |
|---|---|
| Proposed Answer | **B** |
| Applicant References | **None** |
| Explanation | **With Main Generator hydrogen pressure at 70 psig and output at 1150 MWe, reactive load is limited to approximately 520 MVAR by the Generator Capability Curve. If this limit is approached or exceeded, ON-103-001 requires notifying TCC and waiting two minutes before taking manual action to lower MVARs.** |

**A Incorrect – ON-103-001 requires waiting two minutes to allow the auto voltage regulator time to attempt to adjust MVARs before taking manual action.**

B Correct.

C Incorrect – The limit is approximately 520 MVAR for these conditions. 565 MVAR would be the approximate limit if hydrogen pressure were 75 psig. ON-103-001 requires waiting two minutes to allow the auto voltage regulator time to attempt to adjust MVARs before taking manual action.

D Incorrect – The limit is approximately 520 MVAR for these conditions. 565 MVAR would be the approximate limit if hydrogen pressure were 75 psig.

| | |
|---|---|
| 10CFR55 | 41.5 |
| Technical References | ON-103-001 |
| Learning Objectives | 1176 |
| Question Source | New |
| Previous NRC Exam | No |
| Comments | |

| Exam | RO | Tier | 1 | Group | 1 | Cognitive Level | High | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 295025 2.2.40 High Reactor Pressure | | | | | Importance | 3.4 | | |
| Statement | Ability to apply technical specifications for a system. | | | | | | | | |

QUESTION 54

Unit 1 is operating at 100% power with the following:

- A spurious closure of all MSIVs occurs.
- The Reactor scrams.
- Multiple SRVs fail to open as required.
- Peak Reactor pressure during the transient is 1360 psig.
- Level control malfunctions result in Reactor water level reaching a low of -135".
- Reactor pressure and water level have been recovered to normal bands.

Which one of the following describes the status of the Safety Limits for Reactor pressure and Reactor water level during this transient, in accordance with Technical Specifications?

| | Reactor Pressure Safety Limit | Reactor Water Level Safety Limit |
|---|---|---|
| A. | Violated | Violated |
| B. | Violated | NOT violated |
| C. | NOT violated | Violated |
| D. | NOT violated | NOT violated |

Proposed Answer **B**

Applicant References **None**

Explanation **The peak Reactor pressure of 1360 psig violated the Reactor pressure Safety Limit of 1325 psig. The lowest Reactor water level of -135" was below Level 1, but did not violate the Reactor water level Safety Limit of -161" (TAF).**

**A Incorrect - The lowest Reactor water level of -135" was below Level 1, but did not violate the Reactor water level Safety Limit of -161" (TAF).**

**B Correct.**

**C Incorrect - The peak Reactor pressure of 1360 psig violated the Reactor pressure Safety Limit of 1325 psig. The lowest Reactor water level of -135" was below Level 1, but did not violate the Reactor water level Safety Limit of -161" (TAF).**

**D Incorrect - The peak Reactor pressure of 1360 psig violated the Reactor pressure Safety Limit of 1325 psig.**

10CFR55 **41.5**

Technical References **Technical Specifications Safety Limits**

Learning Objectives **13427**

Question Source **New**

Previous NRC Exam **No**

Comments

| Exam | RO | Tier | 1 | Group | 1 | Cognitive Level | High | Level of Difficulty | 3.5 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 295004 2.4.9 Partial or Complete Loss of DC Power | | | | | Importance | 3.8 | | |
| Statement | Knowledge of low power / shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies. | | | | | | | | |

QUESTION 55

A Unit 1 shutdown is in progress with the following:

- Reactor pressure is 0 psig.
- 125 VDC Bus 1D620 de-energizes due to a sustained electrical fault.
- NO actions have yet been taken for the loss of Bus 1D620.
- A coolant leak results in Reactor water level -129", down fast.

Which one of the following describes the availability of injection with Core Spray with only actions in the Control Room?

A. Only two Core Spray pumps will automatically start and only one additional Core Spray pump can be manually started.

B. Only two Core Spray pumps will automatically start, but two additional Core Spray pumps can be manually started.

C. Only three Core Spray pumps will automatically start and NO additional Core Spray pumps can be manually started.

D. NO Core Spray pumps will automatically start and only three Core Spray pumps can be manually started.

Proposed Answer: **A**

Applicant References: **None**

Explanation: **At a Reactor pressure of 0 psig and a Reactor water level of -129", all four Core Spray pumps should normally automatically start and inject. The loss of Bus 1D620 affects Core Spray in two ways. Power is lost for automatic initiation of Div. 2 Core Spray, therefore only Core Spray pumps 1A and 1C will automatically start. Control power is lost to Core Spray pump 1B breaker, preventing manual start of the pump with only actions in the Control Room. Control power can be transferred to an alternate source or the breaker could be manually closed, but these actions are accomplished outside of the Control Room. Only Core Spray pump 1D is available for manual start in the Control Room.**

**A Correct.**

**B Incorrect – Only one additional Core Spray pump can be manually started with only actions in the Control Room due to loss of control power for Core Spray pump 1B breaker. Control power can be transferred to an alternate source or the breaker could be manually closed, but these actions are accomplished outside of the Control Room**

**C Incorrect – Three Core Spray pumps have breaker control power, but one of these pumps will still not automatically start due to loss of power to Div. 2 auto initiation circuitry.**

**D Incorrect – Power is lost to Div. 2 Core Spray automatic initiation circuitry, however the Div. 1 circuitry will still cause two Core Spray pumps (1A and 1C) to automatically start.**

| | |
|---|---|
| 10CFR55 | **41.8** |
| Technical References | **ON-102-620** |
| Learning Objectives | **2093.d** |
| Question Source | **New** |
| Previous NRC Exam | **No** |
| Comments | |

| Exam | RO | Tier | 1 | Group | 1 | Cognitive Level | High | Level of Difficulty | 3.5 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 295019 2.4.50 Partial or Complete Loss of Instrument Air | | | | | Importance | | 4.2 | |
| Statement | Ability to verify system alarm setpoints and operate controls identified in the alarm response manual. | | | | | | | | |

QUESTION 56

Unit 1 is operating at 100% power with the following:

- A significant air leak is reported on the Scram Pilot Valve Air Header.
- Annunciator AR-107-G01, SCRAM PILOT VALVE AIR HEADER LO PRESS, alarms.
- Scram Pilot Valve Air Header pressure is 65 psig, down slow.

Which one of the following describes the validity of annunciator AR-107-G01 and the need to perform a Reactor scram?

Annunciator AR-107-G01 is...

A. NOT valid. A Reactor scram is NOT required.

B. valid. A Reactor scram is required.

C. valid. A Reactor scram is NOT required unless pressure lowers further.

D. Valid. A Reactor scram is NOT required unless control rods begin to drift.

Proposed Answer: **B**

Applicant References: **None**

Explanation: **Annunciator AR-107-G01 has a setpoint of 65 psig in the Scram Pilot Valve Air Header, therefore the alarm is valid. The response procedure requires a Reactor scram based on valid receipt of the alarm.**

**A Incorrect – This alarm is valid due to Scram Pilot Valve Air Header pressure of 65 psig.**

**B Correct.**

**C Incorrect – A Reactor scram is required with Scram Pilot Valve Air Header pressure at 65 psig.**

**D Incorrect – A Reactor scram is required with Scram Pilot Valve Air Header pressure at 65 psig. This is required to insert control rods before they begin to drift.**

10CFR55: **41.10**

Technical References: **AR-107-G01**

Learning Objectives: **10012**

Question Source: **New**

Previous NRC Exam: **No**

Comments

| Exam | RO | Tier | 1 | Group | 1 | Cognitive Level | High | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 295021 AA1.02 Loss of Shutdown Cooling | | | | | Importance | 3.5 | | |
| Statement | Ability to operate and/or monitor the following as they apply to LOSS OF SHUTDOWN COOLING: RHR/shutdown cooling | | | | | | | | |

QUESTION 57

Unit 1 is shutdown with the following:

- Reactor coolant temperature is 250°F, down slow.
- RHR loop B is operating in the Shutdown Cooling mode.
- RPS Bus A de-energizes due to a sustained electrical fault.

Which one of the following describes the status of Shutdown Cooling / RHR?

Shutdown Cooling...

A. remains in service on RHR loop B.

B. is lost. RHR loop A is available to be placed in Shutdown Cooling mode.

C. is lost. Neither loop of RHR is available to be placed in Shutdown Cooling mode. Only RHR loop B is available to be placed in Fuel Pool Cooling Assist mode.

D. is lost. Neither loop of RHR is available to be placed in Shutdown Cooling mode. Both loops of RHR are available to be placed in Fuel Pool Cooling Assist mode.

Proposed Answer: **D**

Applicant References: **None**

Explanation: **Both loops of RHR are unavailable for Shutdown Cooling mode because Shutdown Cooling suction IV F009 closes due to loss of RPS Bus A. This failure does not affect the RHR Fuel Pool Cooling Assist mode, therefore it remains available with either RHR loop.**

**A Incorrect – Although loss of RPS Bus A only causes one of the Shutdown Cooling suction IVs to close, this still causes loss of Shutdown Cooling on RHR loop B.**

**B Incorrect – Although loss of RPS Bus A only causes one of the Shutdown Cooling suction IVs to close, this still causes loss of Shutdown Cooling on both RHR loops. The sustained electrical fault on RPS Bus A prevents transferring power to the alternate supply to restore SDC using RHR.**

**C Incorrect – Loss of RPS A does not prevent placing RHR loop A in the Fuel Pool Cooling Assist mode.**

**D Correct.**

10CFR55: **41.7**

Technical References: **ON-158-001, ON-149-001, OP-149-003**

Learning Objectives: **192.a**

Question Source: **Bank** **Vision SYSID 1056**

Previous NRC Exam: **No**

Comments

| Exam | RO | Tier | 1 | Group | 1 | Cognitive Level | High | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | | 295031 EK3.02 Reactor Low Water Level | | | | Importance | | 4.4 | |
| Statement | | Knowledge of the reasons for the following responses as they apply to REACTOR LOW WATER LEVEL: Core coverage | | | | | | | |

QUESTION 58

Unit 1 has experienced a loss of coolant accident with the following:

- All control rods are in.
- Reactor water level is -150", down slow.
- Reactor pressure is 800 psig, down slow.
- CRD is injecting to the Reactor and has been maximized.
- Condensate pumps are running and available for injection.
- No other injection sources are available.

Which one of the following describes the lowest Reactor water level that assures adequate core cooling under these conditions and the associated basis, in accordance with the Emergency Operating Procedures?

The lowest Reactor water level that assures adequate core cooling is __(1)__ under these conditions. Maintaining Reactor water level above this level ensures peak fuel clad temperature is limited to a maximum of __(2)__.

| | (1) | (2) |
|---|---|---|
| A. | -161" | 1500°F |
| B. | -161" | 1800°F |
| C. | -210" | 1500°F |
| D. | -210" | 1800°F |

Proposed Answer **A**

Applicant References **None**

Explanation **Under non-ATWS conditions, with CRD injection but without Core Spray available, the only allowable method to assure adequate core cooling is core submergence. Core submergence is lost if Reactor water level goes below -161". Adequate core cooling by core submergence ensures fuel clad temperature remains below a maximum of 1500°F.**

**A Correct.**

**B Incorrect – 1500°F is the peak fuel clad temperature associated with adequate core cooling by core submergence. 1800°F is associated with steam cooling without injection, which is not applicable in this situation due to injection from CRD.**

**C Incorrect – -210" is the lowest level that assures adequate core cooling if Core Spray flow is >6350 gpm. With no available Core Spray, -161" is the lowest Reactor water level that assures adequate core cooling.**

**D Incorrect – -210" is the lowest level that assures adequate core cooling if Core Spray flow is >6350 gpm. With no available Core Spray, -161" is the lowest Reactor water level that assures adequate core cooling. 1500°F is the peak fuel clad temperature associated with adequate core cooling by core submergence. 1800°F is associated with steam cooling without injection, which is not applicable in this situation due to injection from CRD.**

| | |
|---|---|
| 10CFR55 | **41.10** |
| Technical References | **EO-100-102, EPG/SAG Appendix B Volume 1 Section 3** |
| Learning Objectives | |
| Question Source | **New** |
| Previous NRC Exam | **No** |
| Comments | |

| Exam | RO | Tier | 1 | Group | 2 | Cognitive Level | High | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 295008 AK1.03 High Reactor Water Level | | | | | Importance | 3.2 | | |
| Statement | Knowledge of the operational implications of the following concepts as they apply to HIGH REACTOR WATER LEVEL: Feed flow/steam flow mismatch | | | | | | | | |

QUESTION 59

Unit 1 is operating at 100% power with the following:

- One "A" RFP Discharge Flow transmitter is automatically removed from the ICS/DCS total feed flow calculation due to deviation >0.50 Mlbm/hr.
- LIC-C32-1R600, FW Level Ctl/Demand Signal Controller, is in AUTOMATIC.
- Reactor level control is selected to 3 ELEMENT.
- Reactor water level is +35", stable.

Then, a second "A" RFP Discharge Flow transmitter fails upscale.

Which one of the following describes the response of Reactor water level with NO operator action?

Reactor water level...

A. remains stable at approximately +35".

B. lowers to approximately +32" and then stabilizes at this level.

C. lowers to approximately +32" and then rises and stabilizes at approximately +35".

D. lowers to +13" and causes a Reactor scram.

Proposed Answer: **C**

Applicant References: **None**

Explanation: **If this upscale failure were the first failure, ICS would automatically remove the instrument from service and Reactor water level would remain stable at approximately +35". However, since this is the second instrument to fail, it is NOT removed from service automatically, and will cause a Reactor water level transient. Reactor water level will lower to approximately 32", but will return to approximately +35" automatically due to the Feedwater Level Control system.**

**A** **Incorrect – If this upscale failure were the first failure, ICS would automatically remove the instrument from service and Reactor water level would remain stable at approximately +35". However, since this is the second instrument to fail, it is NOT removed from service automatically, and will cause a Reactor water level transient.**

**B** **Incorrect – Reactor water level will lower to approximately 32", but will return to approximately +35" automatically due to the Feedwater Level Control system.**

**C** **Correct.**

**D** **Incorrect – Reactor water level will begin to lower, but the deviation caused by upscale failure of this one flow transmitter does not cause enough of an error signal to lower Reactor water level to +13".**

| | | |
|---|---|---|
| 10CFR55 | **41.5** | |
| Technical References | **ON-145-001** | |
| Learning Objectives | **16014.i** | |
| Question Source | **Modified Bank** | **LOC24 Cert #86** |
| Previous NRC Exam | **No** | |
| Comments | | |

| Exam | RO | Tier | 1 | Group | 2 | Cognitive Level | Low | Level of Difficulty | 2.5 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 295032 EK2.01 High Secondary Containment Area Temperature | | | | | Importance | 3.5 | | |
| Statement | Knowledge of the interrelations between HIGH SECONDARY CONTAINMENT AREA TEMPERATURE and the following: Area/room coolers | | | | | | | | |

QUESTION 60

Unit 1 is operating at 100% power with the following:

- The Reactor automatically scrams due to a Feedwater Level Control malfunction.
- HPCI automatically starts on low Reactor water level.
- A small steam leak develops in the HPCI Equipment Room.

Which one of the following describes the resulting operation of the HPCI Equipment Room Unit Coolers?

A. Both Unit Coolers are running due to the start of HPCI.

B. Both Unit Coolers remain in standby, but will automatically start if air temperature exceeds a limit.

C. One Unit Cooler is running due to the start of HPCI. The other cooler remains in standby unless manually started.

D. One Unit Cooler is running due to the start of HPCI. The other cooler remains in standby, but will automatically start if air temperature exceeds a limit.

| | |
|---|---|
| Proposed Answer | **D** |
| Applicant References | **None** |
| Explanation | **One Unit Cooler automatically starts upon start of HPCI. The second Unit Cooler remains in standby, but will automatically start if air temperature at the discharge of the running Unit Cooler exceeds 120°F.** |
| | **A Incorrect – Only one Unit Cooler automatically starts due to start of HPCI.** |
| | **B Incorrect – One Unit Cooler automatically starts upon start of HPCI.** |
| | **C Incorrect – The second Unit Cooler will automatically start if air temperature at the discharge of the running Unit Cooler exceeds 120°F.** |
| | **D Correct.** |
| 10CFR55 | **41.8** |
| Technical References | **TM-OP-052** |
| Learning Objectives | **1268.f** |
| Question Source | **Modified Bank Vision SYSID 79** |
| Previous NRC Exam | **No** |
| Comments | |

| Exam | RO | Tier | 1 | Group | 2 | Cognitive Level | High | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | | 295020 AK3.07 Inadvertent Containment Isolation | | | | Importance | | 3.4 | |
| Statement | | Knowledge of the reasons for the following responses as they apply to INADVERTENT CONTAINMENT ISOLATION: Suppression pool temperature response | | | | | | | |

QUESTION 61

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

- An SRV is leaking.
- Suppression Pool Cooling is in service.
- Suppression Pool temperature is 80°F, stable.

Then, a spurious -38" Reactor water level isolation occurs.

Which one of the following describes the response of Suppression Pool temperature two (2) minutes later?

Suppression Pool temperature...

A. remains stable due to continued Suppression Pool Cooling.

B. begins to rise due to trip of the running RHRSW pump.

C. begins to rise due to actuation of SRVs for Reactor pressure control.

D. begins to rise due to re-alignment of RHR from Suppression Pool Cooling to LPCI.

Proposed Answer **B**

Applicant References **None**

Explanation **A -38" Reactor water level isolation causes actuations in multiple systems. RHR remains in the Suppression Pool Cooling lineup. Additionally, nothing in the isolation will cause MSIVs to close (as they would on a -129" isolation), therefore the Main Condenser remains in service as the Reactor's heat sink and SRVs do not actuate for Reactor pressure control.**

**The running RHRSW pump does trip on the -38" signal. With loss of cooling to the RHR heat exchanger, Suppression Pool Cooling heat removal lowers. Since Suppression Pool temperature was initially stable, the lower heat removal rate will result in Suppression Pool temperature beginning to rise.**

**A Incorrect – RHR remains aligned in Suppression Pool Cooling, loss of RHRSW flow to the heat exchanger will reduces the heat removal rate and cause Suppression Pool temperature to rise due to the leaking SRV.**

**B Correct.**

**C Incorrect – The Main Condenser remains in service as the Reactor's heat sink, therefore SRVs do not actuate for Reactor pressure control. MSIVs would close on a -129" isolation signal and result in SRV actuation.**

**D Incorrect – RHR remains in Suppression Pool Cooling mode. Only the SDC mode would isolate on a -38" isolation, if initially in service. RHR would only re-align to LPCI if another signal were received (-129" or Drywell pressure 1.72 psig).**

| | |
|---|---|
| 10CFR55 | **41.7** |
| Technical References | **AR-111-G03, ON-159-002, OP-116-001** |
| Learning Objectives | **2123.a/d** |
| Question Source | **New** |
| Previous NRC Exam | **No** |
| Comments | |

| Exam | RO | Tier | 1 | Group | 2 | Cognitive Level | High | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 295002 AA1.01 Loss of Main Condenser Vacuum | | | | | Importance | | 2.6 | |
| Statement | Ability to operate and/or monitor the following as they apply to LOSS OF MAIN CONDENSER VACUUM: Condensate system | | | | | | | | |

QUESTION 62

Unit 1 is operating at 40% power with the following:

- Main Condenser vacuum begins to degrade due to air in-leakage.
- The Reactor mode switch is placed in SHUTDOWN.
- No other operator actions are performed.
- Reactor water level reaches a low value of +5" following the scram.
- Reactor pressure is 930 psig, stable.
- Main Condenser pressure is 14.2" Hga, stable.

Which one of the following describes the status of injection with Condensate and Feedwater?

A. Feedwater pumps are available and will control Reactor water level at +22".

B. Feedwater pumps are available and will control Reactor water level at +35".

C. Feedwater pumps are unavailable. Condensate pumps will automatically inject if Reactor water level is low and Reactor pressure lowers to within their capacity.

D. Feedwater pumps are unavailable. Condensate pumps will NOT automatically inject if Reactor water level is low and pressure lowers to within their capacity.

Proposed Answer **C**

Applicant References **None**

Explanation **Reactor water level <+13" causes Setpoint Setdown to initiate and Feedwater Level Control system to automatically transfer from the Flow Control Mode to the Startup Level Control Mode. Setpoint Setdown will control Reactor water level at +22". Feedwater pumps are unavailable for injection because they trip with Main Condenser pressure >11.8" Hga. With Feedwater Level Control in the Startup Level Control Mode, Low Load Valve LV-10641 will re-position to control Reactor water level. This will allow Condensate pumps to automatically inject if Reactor water level is low and Reactor pressure lowers to within their capacity.**

**A** **Incorrect – Feedwater pumps are unavailable because they have tripped on Main Condenser pressure >11.8" Hga.**

**B** **Incorrect – Feedwater pumps are unavailable because they have tripped on Main Condenser pressure >11.8" Hga. Feedwater Level Control will also control Reactor water level at +22" since Setpoint Setdown initiated on <+13".**

**C** **Correct.**

**D** **Incorrect – With Feedwater Level Control automatically swapping to the Startup Level Control Mode, LV-10641 will automatically open and allow Condensate pump injection if Reactor water level is low and Reactor pressure lowers to within their capacity.**

10CFR55 **41.4**

Technical References **ON-143-000, ON-101-100, OP-145-001 Attachment A**

| | |
|---|---|
| Learning Objectives | **16006.e** |
| Question Source | **New** |
| Previous NRC Exam | **No** |
| Comments | |

| Exam | RO | Tier | 1 | Group | 2 | Cognitive Level | High | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|

| K/A | 295010 AA2.01 High Drywell Pressure | Importance | 3.4 |
|---|---|---|---|
| Statement | Ability to determine and/or interpret the following as they apply to HIGH DRYWELL PRESSURE: Leak rates | | |

QUESTION 63

Unit 1 is operating at 100% power with the following:

- Drywell pressure is 1.2 psig, up slow.
- Drywell Floor Drain Sump and Equipment Drain Tank levels are as given below.
- No Drywell Floor Drain Sump pump downs or Equipment Drain Tank drainings have occurred during this time.

| Time | Combined Drywell Floor Drain Sump A & B Level | Drywell Equipment Drain Tank Level |
|---|---|---|
| 15 minutes ago | 160 gallons | 340 gallons |
| Now | 181 gallons | 425 gallons |

Which one of the following describes:

(1) the approximate value of Unidentified Leak Rate during this time period, and

(2) whether this value is above or below the Technical Specification (TS) limit?

A. (1) 1.4 gpm
(2) Below the TS limit

B. (1) 1.4 gpm
(2) Above the TS limit

C. (1) 5.7 gpm
(2) Below the TS limit

D. (1) 5.7 gpm
(2) Above the TS limit

Proposed Answer: **A**

Applicant References: **None**

Explanation: **Unidentified Leak Rate is calculated using Drywell Floor Drain Sump levels. For the given 15 minute period, the leak rate into the Drywell Floor Drain Sumps was approximately 1.4 gpm ((181 gallons – 160 gallons) / 15 minutes). This is below the Technical Specification limit of 5 gpm.**

**A Correct.**

**B Incorrect – 1.4 gpm is below the TS limit of 5 gpm.**

C Incorrect - This leak rate is calculated erroneously from the Drywell Equipment Drain Tank levels, which is identified leakage, not unidentified.

D Incorrect - This leak rate is calculated erroneously from the Drywell Equipment Drain Tank levels, which is identified leakage, not unidentified.

| | | |
|---|---|---|
| 10CFR55 | **41.5** | |
| Technical References | **Technical Specification 3.4.4** | |
| Learning Objectives | **10715, 12657** | |
| Question Source | **Bank** | **NMP2 2014 NRC #37** |
| Previous NRC Exam | **Yes** | **NMP2 2014 NRC #37** |
| Comments | | |

| Exam | RO | Tier | 1 | Group | 2 | Cognitive Level | High | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 295014 2.1.32 Inadvertent Reactivity Addition | | | | | Importance | 3.8 | | |
| Statement | Ability to explain and apply all system limits and precautions. | | | | | | | | |

QUESTION 64

Unit 1 is initially operating at approximately 97% power with the following:

- A unanticipated reactivity addition occurs.
- ON-156-001, Unanticipated Reactivity Change, has been entered.
- Investigation regarding the cause of the reactivity addition is in progress.
- Computer point NBA01, CTP Instantaneous, indicates 3950 MWth, up slow.
- Computer point NBA103, CTP 60 Minute Average, indicates 3930 MWth, up slow.

Which one of the following describes the need for a Reactor power reduction based on only these indications, in accordance with ON-156-001?

A Reactor power reduction...

A. is currently required because Reactor power is rising.

B. is NOT currently required, but will first be required if NBA01 exceeds 3952 MWth.

C. is NOT currently required, but will first be required if NBA01 exceeds 3991 MWth.

D. is NOT currently required, but will first be required if NBA103 exceeds 3952 MWth.

Proposed Answer **A**

Applicant References **None**

Explanation **The power rise has not yet exceeded the licensed power level of 3952 MWth, however ON-156-001 requires a power reduction in the event of any unanticipated reactivity change that causes a rise in power. Since both of the given CTP indications are rising, Reactor power is rising (significantly above the initial 97%) and ON-156-001 currently requires a power reduction to the pre-transient power level.**

**A Correct.**

**B Incorrect – A Reactor power reduction is currently required. NBA01 exceeding 3952 MWth would be the first indication that instantaneous CTP is greater than the licensed limit.**

**C Incorrect – A Reactor power reduction is currently required. NBA01 exceeding 3991 MWth would be a direct entry condition into ON-100-004.**

**D Incorrect – A Reactor power reduction is currently required. NBA103 exceeding 3952 MWth would be an indication that Reactor power has exceeded the licensed limit for a sustained period of time. NBA103 exceeding 3952 MWth would also be a direct entry condition into ON-100-004.**

10CFR55 **41.10**

Technical References **ON-156-001, ON-100-004**

Learning Objectives

Question Source **New**

Previous NRC Exam **No**

Comments

| Exam | RO | Tier | 1 | Group | 2 | Cognitive Level | High | Level of Difficulty | 3.5 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | | 295029 2.4.46 High Suppression Pool Water Level | | | | Importance | | 4.2 | |
| Statement | | Ability to verify that the alarms are consistent with the plant conditions. | | | | | | | |

QUESTION 65

Unit 1 has experienced a loss of coolant accident with the following:

- Annunciator AR-114-F01, SUPPRESSION POOL HI LEVEL, is in alarm.
- Suppression Pool water level is 34', up slow.
- Reactor water level is +20", up slow.
- HPCI is injecting to the Reactor with suction aligned to the CSTs.
- RCIC is available in the standby alignment.
- Suppression Pool Cooling is in service.
- Suppression Pool water temperature is 95°F, down slow.

Which one of the following describes the proper operation of HPCI and RCIC in accordance with EO-100-103, Primary Containment Control?

A. HPCI and RCIC are being operated properly.

B. RCIC is being operated properly, but HPCI is NOT.

C. HPCI is being operated properly, but RCIC is NOT.

D. Both HPCI and RCIC are NOT being operated properly.

| | |
|---|---|
| Proposed Answer | **D** |
| Applicant References | **None** |
| Explanation | **With Suppression Pool water level in alarm high and greater than 25', both HPCI and RCIC are required to be running per EO-100-103 step SP/L-11. Since RCIC is not running, it is not being operated properly. With HPCI injecting to the Reactor and Suppression Pool temperature being maintained <140°F with Suppression Pool Coolilng, EO-100-103 step SP/L-12 requires transferring the HPCI suction from the CST to the Suppression Pool. Since HPCI is injection with the suction still aligned to the CSTs, it is not being operated properly either.** |
| | **A Incorrect – Neither HPCI nor RCIC are being operated properly since RCIC should be running and HPCI suction should be aligned to the Suppression Pool, not the CSTs.** |
| | **B Incorrect – RCIC is not being operated proper since it should be running.** |
| | **C Incorrect - HPCI is not being operated properly since the suction should be aligned to the Suppression Pool, not the CSTs.** |
| | **D Correct.** |
| 10CFR55 | **41.10** |
| Technical References | **EO-100-103, AR-114-F01** |
| Learning Objectives | |
| Question Source | **New** |
| Previous NRC Exam | **No** |

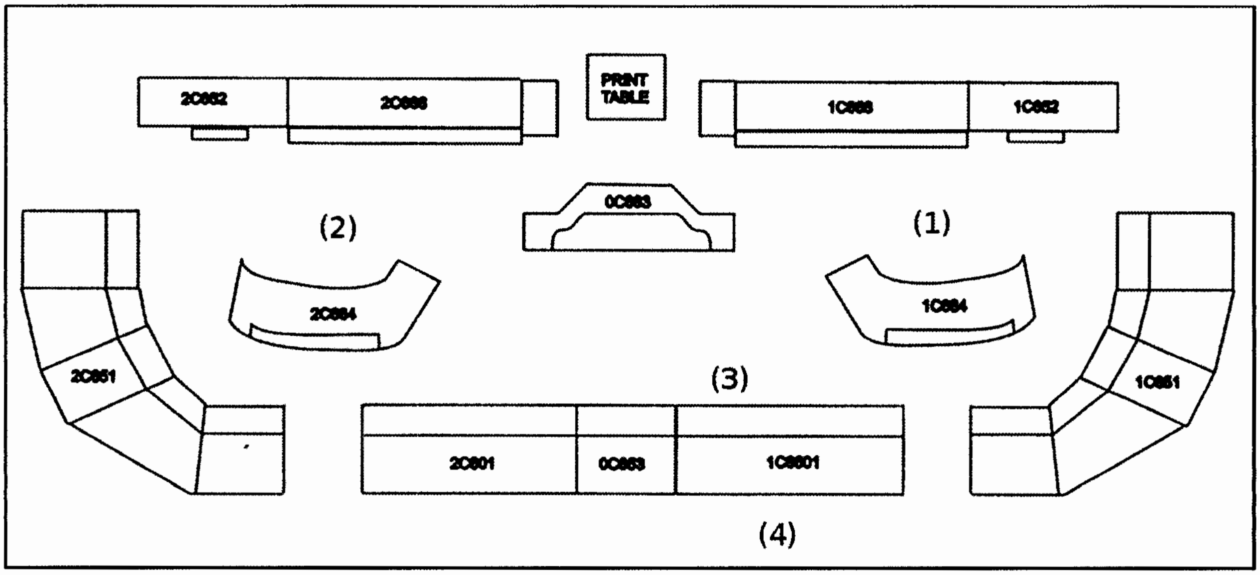
Comments

| | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| Exam | **RO** | Tier | **3** | Group | **N/A** | Cognitive Level | **Low** | Level of Difficulty | **2.5** |
| K/A | **2.1.2** | | | | | Importance | | **4.1** | |
| Statement | **Knowledge of operator responsibilities during all modes of plant operation.** | | | | | | | | |

QUESTION 66

Both Units are operating at 100% power.

Given the following map of the Main Control Room, with four locations marked (1), (2), (3), and (4):




Which one of the following identifies which of these areas the Unit 1 PCOM may access, without relief or special authorization, in accordance with OP-AD-002, Standards for Shift Operations?

A. (1) only

B. (1) and (3) only

C. (1), (2), and (3) only

D. (1), (3), and (4) only

Proposed Answer **B**

Applicant References **None**

| | |
|---|---|
| Explanation | **The individual designated as the Unit 1 PCOM must remain in the Unit 1 At The Controls (ATC) area. (1) is within the Unit 1 ATC area, and not the Unit 2 ATC area. (3) is also within the Unit 1 ATC area. (2) is within the Unit 2 ATC area, but not the Unit 1 ATC area. (4) is outside both the Unit 1 and Unit 2 ATC areas. Therefore, the Unit 1 PCOM may only access (1) and (3).** |
| | **A Incorrect – The Unit 1 PCOM may also access (3).** |
| | **B Correct.** |
| | **C Incorrect – The Unit 1 PCOM may not access (2) (Unit 2 ATC area only).** |
| | **D Incorrect – The Unit 1 PCOM may not access (4) (back panel area).** |
| 10CFR55 | **41.10** |
| Technical References | **OP-AD-002** |
| Learning Objectives | |
| Question Source | **New** |
| Previous NRC Exam | **No** |
| Comments | |

| Exam | RO | Tier | 3 | Group | N/A | Cognitive Level | Low | Level of Difficulty | 2.5 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 2.1.21 | | | | | Importance | 3.5 | | |
| Statement | Ability to verify the controlled procedure copy. | | | | | | | | |

QUESTION 67

Unit 1 is operating at 100% power with the following:

- A working copy of a continuous use Operating Procedure was prepared for an evolution one week ago.
- The evolution was placed on hold, but will now be performed on the current shift.

Which one of the following identifies the source(s) that may be used to verify the working copy of the procedure is the currently approved revision, in accordance with NDAP-QA-0029?

A. Nuclear Information Management System (NIMS) is the only source that is allowed to be used.

B. Electronic Shift Operations Management System (eSOMS) is the only source that is allowed to be used.

C. Either Nuclear Information Management System (NIMS) or a controlled copy of the procedure is allowed to be used.

D. Either Electronic Shift Operations Management System (eSOMS) or a controlled copy of the procedure is allowed to be used.

Proposed Answer: **C**

Applicant References: **None**

Explanation: **NDAP-QA-0029 section 5.2.1 requires verifying the working copy is still the current approved revision due to the delay in task performance. NIMS is an option to verify both procedures and work instructions are current. Use of a controlled copy is also an option for procedures (but not work instructions).**

**A Incorrect - A controlled copy of the procedure may also be used to verify the procedure is the currently approved revision. This option is only unavailable for work instructions that do not have controlled copies, but an Operating Procedure does have controlled copies.**

**B Incorrect – eSOMS is a database used for multiple functions related to plant operation, but not for tracking procedure revisions. A controlled copy of the procedure may also be used to verify the procedure is the currently approved revision. This option is only unavailable for work instructions that do not have controlled copies, but an Operating Procedure does have controlled copies.**

**C Correct.**

**D Incorrect – eSOMS is a database used for multiple functions related to plant operation, but not for tracking procedure revisions.**

10CFR55: **41.10**

Technical References: **NDAP-QA-0029**

Learning Objectives

| | |
|---|---|
| Question Source | **New** |
| Previous NRC Exam | **No** |
| Comments | |

| Exam | RO | Tier | 3 | Group | N/A | Cognitive Level | Low | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 2.2.43 | | | | | Importance | 3.0 | | |
| Statement | Knowledge of the process used to track inoperable alarms | | | | | | | | |

QUESTION 68

A Temporary Engineering Change is to be initiated on the Instrument Air system 'A' that will render the following Control Room Annunciator inoperable:

- AR-124-D01, INSTRUMENT AIR PANEL 1C140A, B SYSTEM TROUBLE

Which one of the following lists the location where the TMOD sticker is required to be placed, in accordance with NDAP-QA-1218, Temporary Changes?

A. On Panel 1C140A, only

B. On the AR-124-D01 annunciator window, only

C. On the alarm response procedure for AR-124-D01, only

D. On the AR-124-D01 annunciator window and on the alarm response procedure for AR-124-D01

Proposed Answer **C**

Applicant References **None**

Explanation **NDAP-QA-1218 requires a TMOD sticker be placed on the affected alarm response procedure only.**

**A Incorrect – The TMOD sticker is required to be placed on the affected alarm response procedure, not the local panel.**

**B Incorrect – The TMOD sticker is required to be placed on the affected alarm response procedure, not the annunciator window.**

**C Correct.**

**D Incorrect – The TMOD sticker is required to be placed on the affected alarm response procedure, but not the annunciator window.**

10CFR55 **41.10**

Technical References **NDAP-QA-1218**

Learning Objectives

Question Source **Bank** **Vision SYSID 292**

Previous NRC Exam **No**

Comments

| Exam | RO | Tier | 3 | Group | N/A | Cognitive Level | High | Level of Difficulty | 2.5 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 2.2.35 | | | | | Importance | 3.6 | | |
| Statement | Ability to determine Technical Specification Mode of Operation. | | | | | | | | |

QUESTION 69

A Unit 1 shutdown is in progress with the following:

- Control rod insertion is in progress.
- Reactor power is:
  - 5% on APRMs.
  - Mid-scale on IRM Range 9.
- Reactor pressure is 975 psig.
- The Reactor Mode Switch has been transferred from RUN to STARTUP/STBY.

Which one of the following is the Technical Specification Mode of Operation?

A. Mode 1 (POWER OPERATION)

B. Mode 2 (STARTUP)

C. Mode 3 (HOT SHUTDOWN)

D. Mode 4 (COLD SHUTDOWN)

Proposed Answer **B**

Applicant References **None**

Explanation **Technical Specification Table 1.1-1 contains the criteria for Reactor Mode of Operation. With the Mode Switch in STARTUP/STBY, the Reactor is in Mode 2 (STARTUP).**

**A Incorrect – While the Reactor is still generating power, as evidenced by APRMs, the Mode Switch is NOT in RUN.**

**B Correct.**

**C Incorrect – While a plant shutdown is in progress, the Mode Switch is NOT in SHUTDOWN.**

**D Incorrect - While a plant shutdown is in progress, the Mode Switch is NOT in SHUTDOWN. Additionally, even though Reactor pressure is lower than the normal operating value, it is still higher than a pressure corresponding to Cold Shutdown conditions.**

10CFR55 **41.10**

Technical References **Technical Specification Table 1.1-1**

Learning Objectives **TM–OP–401 / 13424**

Question Source **Bank JAF September 2012 NRC #70**

Previous NRC Exam **Yes JAF September 2012 NRC #70**

Comments

| Exam | RO | Tier | 3 | Group | N/A | Cognitive Level | Low | Level of Difficulty | 2.5 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 2.3.5 | | | | | Importance | | 2.9 | |
| Statement | Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc. | | | | | | | | |

QUESTION 70

Unit 1 is operating at 80% power with the following:

- Fuel damage has occurred.
- ON-179-001, Increasing Offgas / MSL Rad Levels, has been entered.

Which one of the following identifies the radiation monitor and associated threshold that requires an immediate Reactor scram in accordance with ON-179-001?

An immediate Reactor scram is ***first*** required when __(1)__ radiation monitor readings exceed the __(2)__ alarm setpoint.

| | (1) | (2) |
|---|---|---|
| A. | Offgas | Hi |
| B. | Offgas | Hi-Hi |
| C. | Main Steam Line | Hi |
| D. | Main Steam Line | Hi-Hi |

Proposed Answer **D**

Applicant References **None**

Explanation **The immediate operator actions of ON-179-001 require an immediate Reactor scram if the Main Steam Line radiation monitors exceed the Hi-Hi alarm setpoint.**

**A Incorrect – The Offgas radiation monitors are used for entry into ON-179-001, but are not used in the requirement for a Reactor scram.**

**B Incorrect – The Offgas radiation monitors are used for entry into ON-179-001, but are not used in the requirement for a Reactor scram.**

**C Incorrect – Main Steam Line radiation monitors above the Hi setpoint would require entry into ON-179-001, but a Reactor scram is not required until they exceed the Hi-Hi setpoint.**

**D Correct.**

10CFR55 **41.11**

Technical References **ON-179-001**

Learning Objectives

Question Source **New**

Previous NRC Exam **No**

Comments

| Exam | RO | Tier | 3 | Group | N/A | Cognitive Level | Low | Level of Difficulty | 2.5 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 2.3.4 | | | | | Importance | 3.2 | | |
| Statement | Knowledge of radiation exposure limits under normal or emergency conditions. | | | | | | | | |

QUESTION 71

Which one of the following identifies highest annual administrative dose control limit allowed with a valid dose extension in accordance with NDAP-QA-0625, Personnel Radiation Exposure Monitoring Program?

A. 2000 mRem

B. 3000 mRem

C. 4000 mRem

D. 5000 mRem

Proposed Answer **C**

Applicant References **None**

Explanation **NDAP-QA-0625 sets the highest annual administrative dose control limit to 4000 mRem. This dose control limit requires a dose extension approved by the Functional Unit Manager/Supervisor, Radiation Protection Manager, ALARA Supervision, and V.P.-Nuclear Operations.**

**A Incorrect – 2000 mRem is the highest annual administrative dose control limit without an extension.**

**B Incorrect – 3000 mRem is the highest annual administrative dose control limit that does not require V.P.-Nuclear Operations approval.**

**C Correct.**

**D Incorrect – 5000 mRem is the annual Federal dose limit, but above the administrative limit allowed by NDAP-QA-0625.**

10CFR55 **41.12**

Technical References **NDAP-QA-0625**

Learning Objectives

Question Source **New**

Previous NRC Exam **No**

Comments

| Exam | RO | Tier | 3 | Group | N/A | Cognitive Level | High | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 2.4.1 | | | | | Importance | 4.6 | | |
| Statement | Knowledge of EOP entry conditions and immediate action steps. | | | | | | | | |

QUESTION 72

A transient on Unit 1 has resulted in the following:

- Reactor water level is +18", up slow.
- Reactor pressure is 1090 psig, up slow.
- Reactor power is downscale on APRMs.
- Drywell pressure is 1.4 psig, up slow.
- Drywell average temperature is 145°F, up slow.
- Suppression Pool water level is 23.7 feet, up slow.
- Suppression Pool temperature is 82°F, up slow.
- Annunciator AR-113-H01, CORE SPRAY LOOP B PUMP ROOM FLOODED, is in alarm.
- An Operator in the field has confirmed this alarm is valid.

Which one of the following identifies the Emergency Operating Procedures that are required to be entered based on current conditions?

A. EO-100-102, RPV Control, and EO-100-103, Primary Containment Control, only

B. EO-100-102, RPV Control, and EO-100-104, Secondary Containment Control, only

C. EO-100-103, Primary Containment Control, and EO-100-104, Secondary Containment Control, only

D. EO-100-102, RPV Control, EO-100-103, Primary Containment Control, and EO-100-104, Secondary Containment Control

Proposed Answer **B**

Applicant References **None**

Explanation **EO-100-102 entry is required due to Reactor pressure >1087 psig. EO-100-104 entry is required due to a Reactor Building water level above the high alarm (AR-113-H01). No other EOP entry conditions are met or exceeded.**

**A Incorrect – EO-100-103 entry is not required. While multiple entry parameters are indicating abnormal values, none have met or exceeded an entry condition (Drywell pressure >1.72 psig, Drywell temp >150°F, Supp Pool Level >24', Supp Pool temp >90°F). EO-100-104 entry is required due to a Reactor Building water level above the high alarm (AR-113-H01).**

**B Correct.**

**C Incorrect - EO-100-103 entry is not required. While multiple entry parameters are indicating abnormal values, none have met or exceeded an entry condition (Drywell pressure >1.72 psig, Drywell temp >150°F, Supp Pool Level >24', Supp Pool temp >90°F). EO-100-102 entry is required due to Reactor pressure >1087 psig.**

D Incorrect - EO-100-103 entry is not required. While multiple entry parameters are indicating abnormal values, none have met or exceeded an entry condition (Drywell pressure >1.72 psig, Drywell temp >150°F, Supp Pool Level >24', Supp Pool temp >90°F).

| | | |
|---|---|---|
| 10CFR55 | **41.10** | |
| Technical References | **EO-100-102, EO-100-103, EO-100-104** | |
| Learning Objectives | | |
| Question Source | **Modified Bank** | **JAF 4/14 NRC #73** |
| Previous NRC Exam | **No** | |
| Comments | | |

| Exam | RO | Tier | 3 | Group | N/A | Cognitive Level | Low | Level of Difficulty | 2.5 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 2.4.13 | | | | | Importance | 4.0 | | |
| Statement | Knowledge of crew roles and responsibilities during EOP usage. | | | | | | | | |

QUESTION 73

Unit 1 is operating at 100% power when the following events occur:

- The Unit Supervisor directs an RO to manually scram the Reactor per ON-100-101, Scram, Scram Imminent.
- The RO places the Reactor Mode Switch to SHUTDOWN.
- No control rods insert.
- All RPS Rod Group Scram Indicator lights remain lit (in Upper and Lower Relay Rooms).

Given the following possible actions:

(1) Arm and depress manual scram pushbuttons.
(2) Initiate Alternate Rod Insertion (ARI).
(3) Inject Standby Liquid Control (SBLC).
(4) Trip both Recirculation pumps.

**Note:** Assume control rods continue to fail to insert.

Which one of the following identifies which of these actions the RO is allowed to take without further direction, in accordance with ON-100-101 and the Emergency Operating Procedures?

A. (1) only

B. (1) and (2) only

C. (1), (2), and (3) only

D. (1), (2), (3), and (4)

Proposed Answer **B**

Applicant References **None**

Explanation **ON-100-101 and the Emergency Operating Procedures contain guidance for performing each of these actions in an attempt to lower Reactor power during a failure to scram. ON-100-101 contains specific guidance allowing the RO to depress the manual scram pushbuttons and initiate ARI without further direction from the Unit Supervisor (immediate operator actions). Further actions to inject SBLC and trip both Recirculation pumps will be taken if the failure to scram persists. However, these actions are controlled in EO-100-113, Level/Power Control, and are NOT allowed to be performed by the RO until further direction is received from the Unit Supervisor.**

**A Incorrect – The RO is also allowed to initiate ARI without further direction from the Unit Supervisor.**

**B Correct.**

**C Incorrect – The RO is not allowed to inject SBLC without further direction from the Unit Supervisor.**

D Incorrect – The RO is not allowed to inject SBLC or trip Recirculation pumps without further direction from the Unit Supervisor.

| | | |
|---|---|---|
| 10CFR55 | 41.10 | |
| Technical References | ON-100-101, EO-100-102, EO-100-113 | |
| Learning Objectives | | |
| Question Source | Bank | NMP1 2009 Audit #88 |
| Previous NRC Exam | No | |
| Comments | | |

| Exam | RO | Tier | 3 | Group | N/A | Cognitive Level | Low | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 2.3.11 | | | | | Importance | 3.8 | | |
| Statement | Ability to control radiation releases. | | | | | | | | |

QUESTION 74

Unit 1 is operating at 100% with the following:

- A Main Steam leak develops in the Turbine Building.
- The Reactor is manually scrammed.
- MSIVs have failed to close.
- EO-100-105, Radioactivity Release Control, has been entered.
- The running Turbine Building HVAC fans have tripped.

Which one of the following describes how Turbine Building HVAC must be controlled and the associated reason, in accordance with EO-100-105?

Turbine Building HVAC must be…

A. restarted to minimize an unmonitored ground level release.

B. restarted to prevent equipment damage in the Turbine Building.

C. maintained out of service to minimize the total release to the environment.

D. maintained out of service to prevent damage to the redundant train of Turbine Building HVAC.

Proposed Answer **A**

Applicant References **None**

Explanation **EO-100-105 step RR-1 directs restarting Turbine Building HVAC. This is done to minimize the unmonitored ground level release by directing steam to the stack, an elevated, monitored release. This is done to assist in dispersing any radioactivity released and delaying the radioactivity from affecting populations before protective actions can be carried out.**

**A Correct.**

**B Incorrect – TB HVAC must be restarted, but the reason is not related to preventing equipment damage in the TB.**

**C Incorrect – TB HVAC is to be restarted.**

**D Incorrect – TB HVAC is to be restarted.**

10CFR55 **41.10**

Technical References **EO-100-105**

Learning Objectives **PP002 / 14613**

Question Source **Bank** **LOC25 Cert #59**

Previous NRC Exam **No**

Comments

| Exam | RO | Tier | 3 | Group | N/A | Cognitive Level | High | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 2.1.5 | | | | | Importance | 2.9 | | |
| Statement | Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc. | | | | | | | | |

QUESTION 75

Given the following:

- You are a licensed Reactor Operator.
- You have been on vacation for the previous 14 days.
- Today was your first shift back on watch.
- You worked the normal day shift from 0700 to 1900.
- You then were required to stay a total of four (4) hours past end of shift due to an on-coming Operator calling in sick.
- You were originally scheduled to cover day shift tomorrow from 0700-1900.

Which one of the following identifies the earliest time you can return to work tomorrow without receiving a waiver, in accordance with NDAP-QA-0025, Working Hour Limits for Station Staff?

A. 0700

B. 0800

C. 0900

D. 1100

Proposed Answer: **C**

Applicant References: **None**

Explanation: **NDAP-QA-0025 requires a minimum of a 10 hour break between successive work periods. The individual worked 4 hours past 1900, so they left work at 2300. 10 hours later is 0900.**

**A Incorrect – 0700 is only 8 hours after the individual's last work period and would not provide the required 10 hour break.**

**B Incorrect – 0800 is only 9 hours after the individual's last work period and would not provide the required 10 hour break.**

**C Correct.**

**D Incorrect – 1100 would satisfy the work hour requirements (12 hour break), but is not the earliest allowed time.**

10CFR55: **41.10**

Technical References: **NDAP-QA-0025**

Learning Objectives:

Question Source: **Bank** **NMP1 2013 Audit #66**

Previous NRC Exam: **No**

Comments:

| Exam | SRO | Tier | 1 | Group | 1 | Cognitive Level | High | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | | 295005 AA2.07 Main Turbine Generator Trip | | | | Importance | | 3.6 | |
| Statement | | Ability to determine and/or interpret the following as they apply to MAIN TURBINE GENERATOR TRIP: Reactor water level | | | | | | | |

QUESTION 76

Unit 1 is operating at rated power with the following:

- I&C reports that a document review has revealed the high Reactor water level Main Turbine trip setpoints associated with the following instruments are set according to the table below:

| Instrument | Setpoint |
|---|---|
| NRLA (PDT-C32-1N004A) | 53.8” |
| NRLB (PDT-C32-1N004B) | 56.0” |
| NRLC (PDT-C32-1N004C) | 52.9” |

Which one of the following describes the impact of these setpoints on compliance with Technical Specification (TS) 3.3.2.2, Feedwater - Main Turbine High Water Level Trip Instrumentation?

TS 3.3.2.2...

A. condition entry is NOT required.

B. requires placing a channel in trip within 7 days, only.

C. requires reducing Reactor power to less than 23% within 4 hours.

D. requires restoring trip capability within 2 hours, or then reducing Reactor power to less than 23% within 4 hours.

Proposed Answer **B**

Applicant References **Technical Specification 3.3.2.2 (with Allowable Value removed)**

Explanation **Technical Specification LCO 3.3.2.2 requires all three channels of Main Turbine high water level trip instrumentation to be operable. Surveillance Requirement 3.3.2.2.3 requires the associated setpoints to be ≤55.5”. Two of the given setpoints meets this requirement, however one setpoint is >55.5”. This makes one channel inoperable and requires entry into Technical Specification 3.3.2.2 Condition A. This requires placing the channel in trip within 7 days.**

**A Incorrect – All three of the given instruments must have a setpoint of ≤55.5”. Since one instrument setpoint is >55.5”, Technical Specification 3.3.2.2 Condition A entry is required.**

**B Correct.**

C Incorrect – With only one channel inoperable, Reactor power reduction is NOT required. The given Reactor power reduction would be required if an additional channel was inoperable and trip capability was not restored within 2 hours.

D Incorrect – With only one channel inoperable, trip capability is maintained. This would be the correct answer if an additional channel was inoperable.

| | |
|---|---|
| 10CFR55 | 43.2 |
| Technical References | Technical Specification 3.3.2.2 |
| Learning Objectives | 15882 |
| Question Source | New |
| Previous NRC Exam | No |
| Comments | |

| Exam | SRO | Tier | 1 | Group | 1 | Cognitive Level | High | Level of Difficulty | 3.5 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 700000 AA2.10 Generator Voltage and Electric Grid Disturbances | | | | | Importance | 3.8 | | |
| Statement | Ability to determine and/or interpret the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: Generator overheating and the required actions | | | | | | | | |

QUESTION 77

Unit 1 is operating at rated power with the following:

- Grid voltage fluctuations are occurring.
- Multiple annunciators are in alarm, including:
    - Annunciator AR-106-B04, STATOR COOLING WATER OUTLET HEADER HI TEMP
    - Annunciator AR-106-B07, EXCITER FLD OVERCURRENT
    - Annunciator AR-106-E09, GEN CORE MONITOR STATOR/FIELD OVERHEATING
- Each of these alarms has been confirmed to be valid.
- Stator Cooling Water outlet header temperature is 168°C, up fast.
- Generator field current is 6200 amps, up fast.

Which one of the following describes the action(s) required to be directed, in accordance with plant procedures?

A. Scram the Reactor per ON-100-101, Scram, Scram Imminent.

B. Lower Reactor power per the CRC book. NEITHER lowering Main Generator reactive load NOR a Reactor scram is required.

C. Lower Main Generator reactive load per ON-198-002, Unit 1 Main Generator MVAR Control for Manual Voltage Regulator Operation When Synched to Grid. NEITHER a Reactor power reduction NOR a Reactor scram is required.

D. Lower Main Generator reactive load per ON-198-002, Unit 1 Main Generator MVAR Control for Manual Voltage Regulator Operation When Synched to Grid. Lower Reactor power per the CRC book. A Reactor scram is NOT required.

Proposed Answer **A**

Applicant References **None**

Explanation **A valid annunciator AR-106-E09 indicates that the voltage and temperature issues have resulted in actual damage to Main Generator windings. The associated annunciator response procedure requires scramming the Reactor per ON-100-101, Scram, Scram Imminent.**

**A Correct.**

**B Incorrect – The high Stator Cooling Water outlet temperature condition requires lowering Reactor power per the CRC book. However, the Generator Core Monitor alarm requires the more significant action of scramming the Reactor.**

**C** **Incorrect – Main Generator current over 6150 amps requires lowering Main Generator MVARs. However, the Generator Core Monitor alarm requires the more significant action of scramming the Reactor.**

**D** **Incorrect – The high Stator Cooling Water temperature, combined with the high Main Generator current, requires lowering both Main Generator current and Reactor power. However, the Generator Core Monitor alarm requires the more significant action of scramming the Reactor.**

| | |
|---|---|
| 10CFR55 | **43.5** |
| Technical References | **AR-106-B04**<br>**AR-106-B07**<br>**AR-106-E09** |
| Learning Objectives | **10285** |
| Question Source | **Modified Bank** **Vision SYSID 727** |
| Previous NRC Exam | **No** |
| Comments | |

| Exam | SRO | Tier | 1 | Group | 1 | Cognitive Level | High | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | | 295028 EA2.06 High Drywell Temperature | | | | Importance | | 3.7 | |
| Statement | | Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE: Torus/suppression chamber air space temperature: Plant-Specific | | | | | | | |

QUESTION 78

Unit 1 is operating at 100% power with the following:

- An extended loss of Drywell cooling occurs.
- Then, a small steam leak in the Drywell develops.
- The Reactor Mode Switch is placed in SHUTDOWN.
- Reactor water level is +35", stable.
- Drywell average temperature is 210°F, stable.
- Drywell pressure is 2.1 psig, stable.
- Suppression Pool water temperature is 80°F, stable.
- Suppression Chamber air temperature is 210°F, stable.
- Suppression Chamber pressure 2.1 psig, stable.

Which one of the following describes an additional failure that would result in these indications and the required direction to be given regarding Containment sprays, in accordance with EO-100-103, Primary Containment Control?

| | Failure | Required Direction |
|---|---|---|
| A. | Stuck open SRV tail pipe vacuum relief valve | Initiate Drywell spray |
| B. | Stuck open SRV tail pipe vacuum relief valve | Initiate Suppression Chamber spray |
| C. | Stuck open Suppression Chamber to Drywell vacuum breaker | Initiate Drywell spray |
| D. | Stuck open Suppression Chamber to Drywell vacuum breaker | Initiate Suppression Chamber spray |

Proposed Answer **D**

Applicant References **None**

Explanation **The given indications show the results of both the Drywell cooling loss and small steam leak in the Drywell. Additionally, Suppression Chamber air temperature and pressure indicate excessive bypass leakage between the Drywell and Suppression Chamber. This would be the indication due to a stuck open Suppression Chamber to Drywell vacuum breaker. EO-100-103 entry is required due to Drywell average temperature >150°F and Drywell pressure >1.72 psig. Suppression Chamber spray is required to be directed due to Drywell pressure >1.72 psig. Drywell spray is NOT required to be directed (Suppression Chamber pressure is <13 psig and Drywell temperature is well below 340°F and stable).**

**A** **Incorrect – A stuck open SRV tail pipe vacuum relief valve would result in steam being released directly to the Drywell airspace upon SRV actuation, but would not cause Drywell and Suppression Chamber parameters to equalize as given. Since Suppression Chamber pressure is <13 psig and Drywell average temperature is well below 340°F and stable, Drywell spray is not currently required to be directed.**

**B** **Incorrect – A stuck open SRV tail pipe vacuum relief valve would result in steam being released directly to the Drywell airspace upon SRV actuation, but would not cause Drywell and Suppression Chamber parameters to equalize as given.**

**C** **Incorrect - Since Suppression Chamber pressure is <13 psig and Drywell average temperature is well below 340°F and stable, Drywell spray is not currently required to be directed.**

**D** **Correct.**

| | |
|---|---|
| 10CFR55 | **43.5** |
| Technical References | **EO-100-103, AR-111-E04** |
| Learning Objectives | |
| Question Source | **New** |
| Previous NRC Exam | **No** |
| Comments | |

| Exam | SRO | Tier | 1 | Group | 1 | Cognitive Level | High | Level of Difficulty | 3.5 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 295021 2.4.11 Loss of Shutdown Cooling | | | | | Importance | | 4.2 | |
| Statement | Knowledge of abnormal condition procedures. | | | | | | | | |

QUESTION 79

Unit 1 is cooling down for an outage with the following:

- RHR loop A is operating in the Shutdown Cooling mode.
- A leak causes Reactor water level to begin lowering.
- An Operator isolates the Shutdown Cooling lineup to stop the leak.
- All attempts to restore a loop of RHR to Shutdown Cooling mode have failed.
- Both Recirculation pumps are unavailable.
- Reactor water level is +35 inches, steady.
- Reactor coolant temperature is 180°F, up slow.

Which one of the following describes the requirements for control of Reactor water level and verifying alternate decay heat removal, in accordance with ON-149-001, Loss of RHR Shutdown Cooling Mode?

Reactor water level __(1)__. Verify functionality of a minimum of __(2)__ alternate method(s) capable of decay heat removal.

| | (1) | (2) |
|---|---|---|
| A. | may be maintained at the current value | one |
| B. | may be maintained at the current value | two |
| C. | must be raised to a higher value | one |
| D. | must be raised to a higher value | two |

Proposed Answer **D**

Applicant References **None**

Explanation **ON-149-001 requires prompt action to establish adequate Reactor coolant circulation given a loss of Shutdown Cooling. Two of the allowed actions are unavailable in this situation (restoring an RHR loop to Shutdown Cooling, starting a Recirculation pump). The only remaining action is to raise Reactor water level to ≥45". ON-149-001 and Technical Specification 3.4.8 also require verifying alternate decay heat removal methods are available. Since both of the two required RHR shutdown cooling loops of unavailable, a minimum of two alternate decay heat removal methods must be verified.**

**A Incorrect – With no available RHR or Recirculation loops, Reactor water level must be raised to ≥45" to establish adequate Reactor coolant circulation. Since both of the two required RHR shutdown cooling loops are unavailable, ON-149-001 and Technical Specification 3.4.8 require verifying a minimum of two alternate decay heat removal methods.**

B Incorrect – With no available RHR or Recirculation loops, Reactor water level must be raised to ≥45" to establish adequate Reactor coolant circulation.

C Incorrect –Since both of the two required RHR shutdown cooling loops are unavailable, ON-149-001 and Technical Specification 3.4.8 require verifying a minimum of two alternate decay heat removal methods.

D Correct.

| | |
|---|---|
| 10CFR55 | 43.5 |
| Technical References | ON-149-001<br>Technical Specification 3.4.8 |
| Learning Objectives | 192.bb |
| Question Source | New |
| Previous NRC Exam | No |
| Comments | |

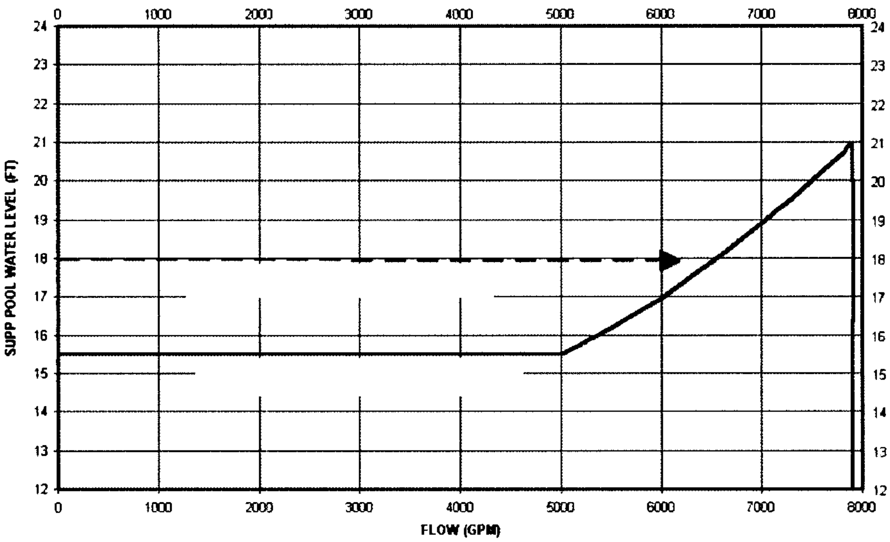
| Exam | SRO | Tier | 1 | Group | 1 | Cognitive Level | High | Level of Difficulty | 2.5 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 295030 2.1.25 Low Suppression Pool Water Level | | | | | Importance | | 4.2 | |
| Statement | Ability to interpret reference materials, such as graphs, curves, tables, etc. | | | | | | | | |

QUESTION 80

Unit 1 has experienced a loss of coolant accident with the following:

- Reactor water level is -170", steady.
- Core Spray Loop A is injecting 7500 gpm with flow maximized.
- No other injection sources are available.
- Suppression Pool water level is 16.5 feet, down slow.

FIG 7 VL
RHR & CS VORTEX LIMIT




Which one of the following describes the required control of Core Spray Loop A flow, in accordance with the Emergency Operating Procedures?

Core Spray Loop A flow...

A. must be maintained at maximum flow.

B. must be lowered as much as possible, but not below 6350 gpm.

C. must be lowered below approximately 5800 gpm irrespective of Reactor water level.

D. must be lowered below approximately 5800 gpm as long as Reactor water level remains above -210".

| | |
|---|---|
| Proposed Answer | **A** |
| Applicant References | **None** |
| Explanation | **With Reactor water level below -129", EO-000-102 step RC/L-10 requires raising Core Spray injection to maintain Reactor water level >-161" irrespective of vortex limits. Since Reactor water level has not yet been restored above -161", Core Spray flow must continue to be maximized, even though it is above the Core Spray Vortex Limit for the given Suppression Pool water level.** |

**A** **Correct.**

**B** **Incorrect – While Core Spray Loop A is operating above its Vortex Limit, EO-000-102 requires continuing to inject with maximum flow due to the low Reactor water level. 6350 gpm is the minimum Core Spray flow rate used in determination of adequate core cooling.**

**C** **Incorrect – While Core Spray Loop A is operating above its Vortex Limit, EO-000-102 requires continuing to inject with maximum flow due to the low Reactor water level. 5800 gpm is the approximate Vortex Limit for Suppression Pool water level of 16.5 feet.**

**D** **Incorrect – While Core Spray Loop A is operating above its Vortex Limit, EO-000-102 requires continuing to inject with maximum flow due to the low Reactor water level. 5800 gpm is the approximate Vortex Limit for Suppression Pool water level of 16.5 feet. -210" is the Reactor water level used in determination of adequate core cooling via spray cooling.**

| | |
|---|---|
| 10CFR55 | **43.5** |
| Technical References | **EO-100-103** |
| Learning Objectives | **15320** |
| Question Source | **Modified Bank** **LOC24 Cert #41** |
| Previous NRC Exam | **No** |
| Comments | |

| Exam | SRO | Tier | 1 | Group | 1 | Cognitive Level | High | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | | **295024 2.4.6 High Drywell Pressure** | | | | Importance | | **4.7** | |
| Statement | | **Knowledge of EOP mitigation strategies.** | | | | | | | |

QUESTION 81

Unit 1 has experienced a loss of coolant accident with the following:

- Reactor water level is +5", up slow, with HPCI injecting.
- Reactor pressure is 500 psig, down slow.
- Suppression Pool water temperature is 130°F, up slow.
- Suppression Pool water level is 23.5', up slow.
- Drywell pressure is 22 psig, down slow.
- Suppression Chamber pressure is 20 psig, down slow.
- Drywell temperature is 250°F, down slow.
- Suppression Chamber and Drywell sprays are in service with RHR pump A running.
- NO other RHR pumps are available.
- Then, RHR pump A trips.
- Drywell and Suppression Chamber pressures begin going up slow.

Which one of the following describes the action required to be directed, in accordance with EO-100-103, Primary Containment Control?

Direct...

A. performing a Rapid Depressurization per EO-100-112.

B. the TSC to make a Containment venting determination per EP-DS-004.

C. venting the Containment using Standby Gas Treatment per OP-173-001.

D. initiating alternate Containment Spray using RHR Service Water per OP-116-001.

Proposed Answer **D**

Applicant References **None**

Explanation **With RHR pump A tripped and no other RHR pumps available, normal Containment Sprays are unavailable. Containment pressures are elevated and threaten the Pressure Suppression Limit, however EO-100-103 requires initiation of alternate Containment Sprays from RHRSW. Use of this alternate source is only precluded if Drywell pressure cannot be maintained less than 65 psig.**

A Incorrect – Containment parameters are near the Pressure Suppression Limit and loss of normal Containment Sprays threaten the ability to maintain operation within this limit. However, EO-100-103 requires use of alternate Containment Spray via RHRSW since Drywell pressure is below 65 psig. Until alternate Containment Sprays are attempted and their effectiveness assessed, a determination cannot be made about the ability to maintain within the Pressure Suppression Limit.

B Incorrect – EO-100-103 requires directing the TSC to make a venting determination only after Suppression Chamber pressure cannot be maintained with the Pressure Suppression Limit and before Containment Pressure reaches 65 psig. Since all Containment Spray sources have not yet been attempted, a determination regarding Pressure Suppression Limit cannot yet be reached.

C Incorrect – The first step in the EO-100-103 Drywell pressure leg directs use of SGTS to lower pressure. However, SGTS can only be used per this step while Drywell pressure is less than 1.72 psig.

D Correct.

| | | |
|---|---|---|
| 10CFR55 | 43.5 | |
| Technical References | EO-100-103 | |
| Learning Objectives | | |
| Question Source | Bank | NMP1 2010 Audit #82 |
| Previous NRC Exam | No | |
| Comments | | |

| Exam | SRO | Tier | 1 | Group | 1 | Cognitive Level | High | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | | 295003 2.1.20 Partial or Complete Loss of AC Power | | | | Importance | | 4.6 | |
| Statement | | Ability to interpret and execute procedure steps. | | | | | | | |

QUESTION 82

A loss of all offsite power has resulted in the following:

- Diesel Generator (DG) A is NOT running.
- DG A has less than 20 psig in the starting air receivers.
- DG B tripped on a generator differential.
- Electrical is investigating to determine the cause of the DG B generator differential.
- DG C is running loaded to the 1C/2C ESS Buses.
- ESW pump C failed to start.
- DG D tripped on low lube oil pressure due to a large leak.

Which one of the following sets of actions is required to be directed?

Direct...

A. shut down of DG C. Then, direct substitution of DG E for DG C.

B. shut down of DG C. Then, direct substitution of DG E for DG A.

C. substitution of DG E for DG C. Direct continued operation of DG C until the substitution is completed.

D. substitution of DG E for DG A. Direct continued operation of DG C until the substitution is completed.

Proposed Answer: **B**

Applicant References: **None**

Explanation: **DG C must be shut down due to operation with no cooling water. OP-054-001 states that DG failure is imminent if operated without cooling water (4.5 minutes loaded, 8 minutes unloaded). EO-100-030 requires prioritizing substitution for DG A over DG C.**

**A Incorrect – EO-100-030 requires prioritizing substitution for DG A over DG C.**

**B Correct.**

**C Incorrect – DG C must be shut down due to operation with no cooling water.**

**D Incorrect – DG C must be shut down due to operation with no cooling water.**

10CFR55: **43.5**

Technical References: **OP-054-001, EO-100-030**

Learning Objectives: **2260.e**

| | | |
|---|---|---|
| Question Source | **Bank** | **LOC24 Cert #76** |
| Previous NRC Exam | **Yes** | **2007 SSES NRC Exam** |
| Comments | | |

| Exam | SRO | Tier | 1 | Group | 2 | Cognitive Level | Low | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | | 295029 EA2.01 High Suppression Pool Water Level | | | | Importance | | 3.9 | |
| Statement | | Ability to determine and/or interpret the following as they apply to HIGH SUPPRESSION POOL WATER LEVEL: Suppression pool water level | | | | | | | |

QUESTION 83

Unit 1 has experienced a loss of coolant accident with the following:

- Reactor water level is -140", up slow, with only Condensate injecting.
- No other injection systems are available.
- Reactor pressure is 230 psig and stable.
- Suppression Pool water level is 27 feet, up slow.
- Attempts to lower Suppression Pool water level have been unsuccessful.

Which one of the following describes:

(1) the next highest Suppression Pool water level limit in EO-100-103, Primary Containment Control, and

(2) the required action if Suppression Pool water level cannot be maintained below this limit?

| | (1) | (2) |
|---|---|---|
| A. | 38 feet | Direct Rapid Depressurization per EO-100-112. |
| B. | 38 feet | Direct Primary Containment water level determination per ON-159-003. |
| C. | 49 feet | Direct Rapid Depressurization per EO-100-112. |
| D. | 49 feet | Direct Primary Containment water level determination per ON-159-003. |

Proposed Answer **A**

Applicant References **None**

Explanation **With Suppression Pool water level at 27 feet, multiple actions have already been required by EO-100-103. The next highest Suppression Pool water level limit in EO-100-103 is 38 feet. If Suppression Pool water level cannot be maintained below this level, then Rapid Depressurization is required per EO-100-112.**

**A Correct.**

**B Incorrect – 38 feet is the next highest level limit, however the associated required action is to perform a Rapid Depressurization, not perform a level determination. Suppression Pool level indication remains an accurate indication for Primary Containment water level until 49 feet.**

C Incorrect – The next highest level limit is 38 feet, not 49 feet. 49 feet is significant because Suppression Pool level indication goes upscale at this level, requiring additional actions to determine actual Primary Containment water level.

D Incorrect – The next highest level limit is 38 feet, not 49 feet. 49 feet is significant because Suppression Pool level indication goes upscale at this level, requiring additional actions to determine actual Primary Containment water level.

| | |
|---|---|
| 10CFR55 | 43.5 |
| Technical References | EO-100-103 |
| Learning Objectives | |
| Question Source | New |
| Previous NRC Exam | No |
| Comments | |

| Exam | SRO | Tier | 1 | Group | 2 | Cognitive Level | High | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | | 295010 2.4.30 High Drywell Pressure | | | | Importance | | 4.1 | |
| Statement | | 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 84

Unit 1 is operating at 100% with the following:

- A spurious MSIV isolation occurs.
- The Reactor scrams.
- Peak Reactor pressure is 1337 psig.
- A small Drywell steam leak occurs.
- Peak Drywell pressure is 2.48 psig.

Which one of the following describes the most restrictive time requirement for notifying the NRC of this event?

A. 15 minutes

B. 1 hour

C. 4 hours

D. 8 hours

Proposed Answer **B**

Applicant References **NDAP-QA-0720**

Explanation **The declaration of any of the emergency classes is reportable within 1 hour as described in NDAP-QA-0720. While it is not important in this case to determine which specific classification is to be made, applicant should recognize that drywell pressure >1.72 psig is cause for entry into EAL and declaration of an emergency (in this case, an alert, FA1). This is the most limiting report required to be made for this event. The injection of ECCS, Tech Spec required shutdown, and RPS actuation with the reactor critical are all 4 hour notifications IAW NDAP-QA-0720.**

**A Incorrect - There are no 15 minute reporting requirements within NDAP-QA-0720. This is a plausible distracter for those candidates that recognize the Emergency classification condition and then select the reporting requirement to the State and County per the Emergency Plan and not the Regulatory Reporting requirement asked for in the stem.**

**B Correct.**

**C Incorrect - The injection of ECCS into the reactor, RPS actuation with the reactor critical, and violation of a safety limit (>1325 psig reactor steam dome pressure) require 4 hour notifications IAW NDAP-QA-0720. This is not the most limiting report required to be made. This is a plausible distractor for those candidates that do not recognize the Emergency Classification that must be made with this event.**

D Incorrect - The automatic isolations that occurred at +13" (following the scram from 100% power) require 8 hour notifications IAW NDAP-QA-0720, however the 1 hour notification of emergency declaration is more limiting.

| | | |
|---|---|---|
| 10CFR55 | **43.5** | |
| Technical References | **NDAP-QA-0720** | |
| Learning Objectives | | |
| Question Source | **Bank** | **LOC24 NRC #83** |
| Previous NRC Exam | **Yes** | **LOC24 NRC #83** |
| Comments | | |

| Exam | SRO | Tier | 1 | Group | 2 | Cognitive Level | High | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | | **295015 AA2.02 Incomplete SCRAM** | | | | Importance | | 4.2 | |
| Statement | | **Ability to determine and/or interpret the following as they apply to INCOMPLETE SCRAM: Control rod position** | | | | | | | |

QUESTION 85

Unit 1 is operating at 100% power with the following:

- Then, a Main Turbine trip occurs.
- APRMs indicate downscale.
- Reactor water level lowers to -25" and then rises with Feedwater injecting.
- Refer to attached OD7 for post-scram Control Rod position.

Which one of the following describes the proper Emergency Operating Procedure execution for this transient?

Enter EO-100-102, RPV Control, and...

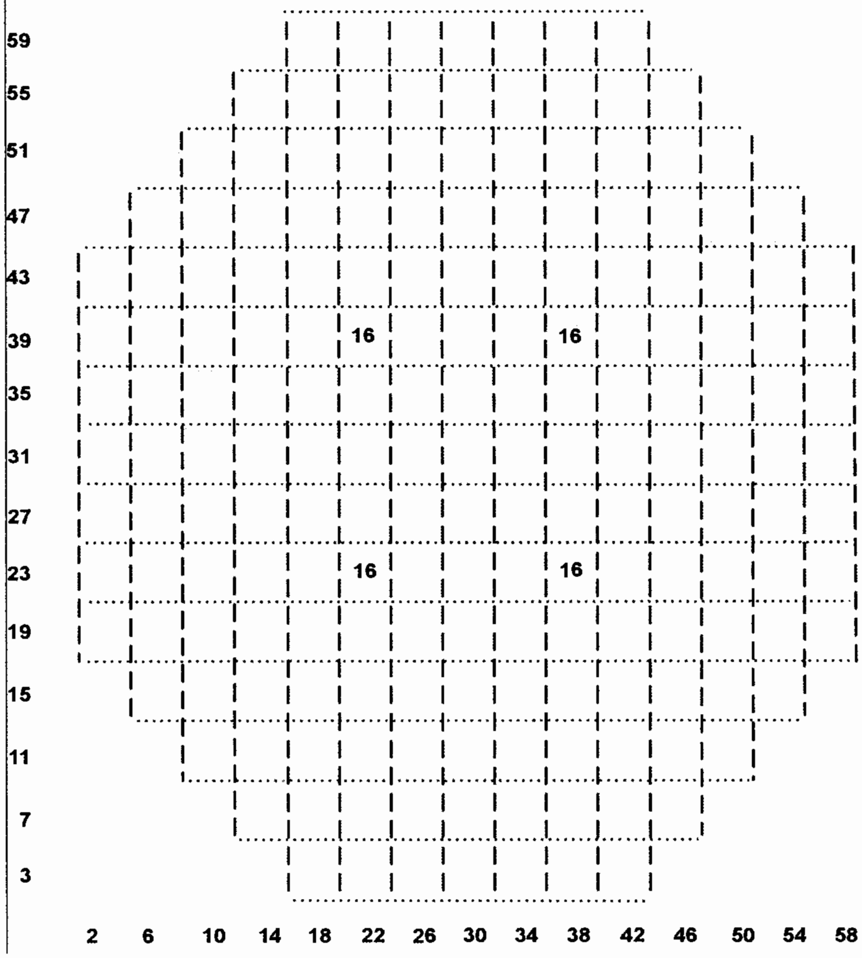
A. remain in EO-100-102 until all EO-100-102 entry conditions clear.

B. then exit to EO-100-113, Level/Power Control. EO-100-113 may NOT be exited until control rod positions are known.

C. then exit to EO-100-113, Level/Power Control. EO-100-113 may be exited once all IRMs are fully inserted and indicate less than range 7.

D. then exit to EO-100-113, Level/Power Control. EO-100-113 may be exited once Standby Liquid Control has injected the Cold Shutdown Boron Weight.

OD - 7. CONTROL ROD NOTCH POSITIONS
SSES UNIT 1



| | |
|---|---|
| Proposed Answer | **B** |
| Applicant References | **None** |
| Explanation | **EO-100-102 entry is required since Reactor water level is below +13". Without control rod position indication, control rods cannot be assumed to be inserted, therefore step RC-2 requires exiting EO-100-102 and entering EO-100-113. Step LQ-2 does not allow exiting EO-100-113 until it can be determined that more than one control rod is not >00. Therefore EO-100-113 may not be exited until control rod positions are known.** |

**A** **Incorrect - With control rod position unknown, EO-100-102 must be exited and EO-100-113 must be entered. APRMs being downscale is not enough to avoid EO-100-113 entry.**

**B** **Correct.**

**C** **Incorrect – EO-100-113 cannot be exited until control rod positions are known. IRMs less than range 7 only gives indication that the Reactor is currently below the POAH.**

**D** **Incorrect – EO-100-113 cannot be exited until control rod positions are known. Injection of Cold Shutdown Boron Weight only allows RPV cooldown and restoration of normal water level while still in EO-100-113.**

| | | |
|---|---|---|
| 10CFR55 | **43.5** | |
| Technical References | **EO-100-102, EO-100-113** | |
| Learning Objectives | | |
| Question Source | **Bank** | **JAF 4/14 NRC #85** |
| Previous NRC Exam | **No** | |

| Exam | SRO | Tier | 2 | Group | 1 | Cognitive Level | High | Level of Difficulty | 3.5 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 209001 A2.05 LPCS | | | | | Importance | | 3.6 | |
| Statement | Ability to (a) predict the impacts of the following on the LOW PRESSURE CORE SPRAY SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Core spray line break | | | | | | | | |

QUESTION 86

Unit 1 is operating at 100% power with the following:

- Annunciator AR-109-E02, CORE SPRAY LOOP A HDR BREAK DETECT HI DIFF PRESS, alarms.
- The associated differential pressure detector (PDIS-E21-1N004A) indicates +4.0 psid.

Which one of the following describes the location of this piping break and the most limiting Technical Specification (TS) condition entry required for this failure?

The piping break is __(1)__. The most limiting TS condition entry requires __(2)__.

| | (1) | (2) |
|---|---|---|
| A. | inside the core shroud. | restoration to operable status within 7 days. |
| B. | inside the core shroud. | immediately entering an LCO 3.0.3 shutdown. |
| C. | between the reactor vessel and core shroud. | restoration to operable status within 7 days. |
| D. | between the reactor vessel and core shroud. | immediately entering an LCO 3.0.3 shutdown. |

Proposed Answer **C**

Applicant References **TS 3.5.1**

Explanation **The given indications show a valid Core Spray line break between the Reactor vessel and core shroud for Core Spray system A. This means that the design spray flow from Core Spray system A cannot be assured. This makes Core Spray system A inoperable. This requires entry into Technical Specification 3.5.1 Condition A, which requires restoring to operable status within 7 days.**

**A Incorrect – The break is between the Reactor vessel and core shroud.**

**B Incorrect – The break is between the Reactor vessel and core shroud and only affects one Core Spray system. If both Core Spray systems were inoperable, then the most limiting condition entry would require an LCO 3.0.3 shutdown.**

**C Correct.**

**D** **Incorrect – The break is between the Reactor vessel and core shroud, but only affects one Core Spray system. If both Core Spray systems were inoperable, then the most limiting condition entry would require an LCO 3.0.3 shutdown.**

| | | |
|---|---|---|
| 10CFR55 | **43.2** | |
| Technical References | **AR-109-E02, TS 3.5.1** | |
| Learning Objectives | **2080.c, 12591** | |
| Question Source | **Bank** | **JAF 2010 NRC #86** |
| Previous NRC Exam | **No** | |
| Comments | | |

| Exam | SRO | Tier | 2 | Group | 1 | Cognitive Level | High | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | | 262001 A2.06 AC Electrical Distribution | | | | Importance | | 2.9 | |
| Statement | | Ability to (a) predict the impacts of the following on the A.C. ELECTRICAL DISTRIBUTION; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: De-energizing a plant bus | | | | | | | |

QUESTION 87

Unit 1 is operating at 100% power with the following:

- A fire alarm is received.
- Investigation reveals that smoke is coming from 1B217, Reactor Building ESS Division 1 480 V Motor Control Center (MCC).
- Operators de-energize 1B217.

Given the following Technical Specifications:

(1) 3.5.1, ECCS - Operating
(2) 3.8.1, AC Sources - Operating
(3) 3.8.7, Distribution Systems - Operating

Which one of the following identifies which of these Technical Specifications require condition entry due to de-energizing 1B217?

A. (1) only

B. (2) only

C. (3) only

D. (1) and (3) only

Proposed Answer: **C**

Applicant References: **TS 3.0.6, TS 3.5.1, TS 3.8.1, TS 3.8.7**

Explanation: **With MCC 1B217 de-energized, multiple safety related loads are unavailable due to loss of power. These loads include Core Spray A injection valves. Technical Specification 3.8.7 condition entry is required due to loss of the MCC. Technical Specification 3.0.6 precludes the need to then cascade into individual Technical Specifications for affected equipment. Therefore Technical Specification 3.5.1 condition entry is not required even though Core Spray is inoperable. Technical Specification 3.8.1 condition entry is not required since loss of the MCC does not make either an offsite line or DG inoperable.**

**A Incorrect – Even though Core Spray A is inoperable due to de-energizing MCC 1B217, Technical Specification 3.5.1 condition entry is not required due to Technical Specification 3.0.6.**

**B Incorrect – Technical Specification 3.8.1 condition entry is not required since loss of the MCC does not make either an offsite line or DG inoperable. Technical Specification 3.8.7 is applicable for loss of an individual MCC.**

**C** **Correct.**

**D** **Incorrect - Even though Core Spray A is inoperable due to de-energizing MCC 1B217, Technical Specification 3.5.1 condition entry is not required due to Technical Specification 3.0.6.**

| | |
|---|---|
| 10CFR55 | **43.2** |
| Technical References | **TS 3.5.1, TS 3.8.1, TS 3.8.7 and associated bases, TS 3.0.6, ON-104-201 Attachment E** |
| Learning Objectives | **13029** |
| Question Source | **Modified Bank     Vision SYSID 32604** |
| Previous NRC Exam | **No** |
| Comments | |

| Exam | SRO | Tier | 2 | Group | 1 | Cognitive Level | High | Level of Difficulty | 3.5 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | | 400000 2.1.28 Component Cooling Water | | | | Importance | | 4.1 | |
| Statement | | Knowledge of the purpose and function of major system components and controls. | | | | | | | |

QUESTION 88

Both plants are operating at 100% power with the following:

- Restoration is in progress from a surveillance test of RHRSW pump 1A.
- HV-01222A, ESW Pond Spr Bpv A, has failed in the closed position.

Which one of the following describes the impact of this failure on system operability, in accordance with Technical Specifications?

Declare RHRSW loop A inoperable for...

A. Unit 1, only. ESW loop A remains operable for both Units.

B. Unit 1, only. Declare ESW loop A inoperable for Unit 1, only.

C. Unit 1 and Unit 2. ESW loop A remains operable for both Units.

D. Unit 1 and Unit 2. Declare ESW loop A inoperable for Unit 1 and Unit 2.

Proposed Answer: **C**

Applicant References: **None**

Explanation: **Technical Specification 3.7.1 requires HV-01222A to be operable for RHRSW to be operable for both Units. Therefore RHRSW loop A is inoperable for both Units. Technical Specification 3.7.2 does not require HV-01222A to be operable for ESW to be operable. Therefore ESW loop A remains operable for both Units.**

**A Incorrect - RHRSW loop A is also inoperable for Unit 2.**

**B Incorrect - RHRSW loop A is also inoperable for Unit 2. ESW loop A remains operable for both Units.**

**C Correct.**

**D Incorrect - ESW loop A remains operable for both Units.**

10CFR55: **43.2**

Technical References: **Technical Specifications 3.7.1 and 3.7.2 and associated bases**

Learning Objectives: **2049.d**

Question Source: **Bank** **Vision SYSID 332**

Previous NRC Exam: **No**

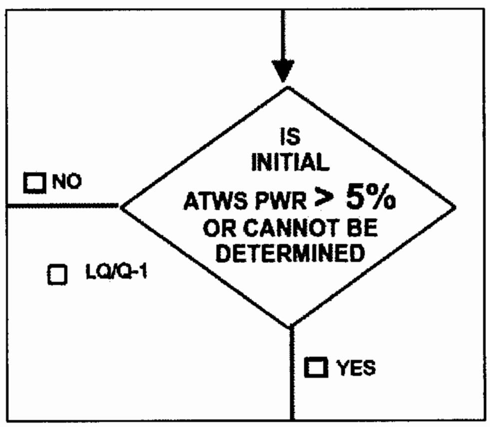
Comments

| Exam | SRO | Tier | 2 | Group | 1 | Cognitive Level | High | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 215005 2.4.18 APRM / LPRM | | | | | Importance | 4.0 | | |
| Statement | Knowledge of the specific bases for EOPs. | | | | | | | | |

QUESTION 89

Unit 1 is operating at 100% power with the following:

- The Main Turbine trips.
- Ten control rods fail to fully insert.
- All APRM indications are lost.
- Reactor pressure is 935 psig, steady.
- One Turbine Bypass Valve is approximately 50% open.
- IRMs are being inserted, but are NOT yet in the core.
- IRMs indicate downscale on range 3.
- EO-100-113, Level/Power Control, is being executed.
- No actions have yet been taken to reduce Reactor power.
- The next step in EO-100-113 is:




Which one of the following describes how this step is required to be answered, in accordance with EO-100-113 and the associated bases?

Answer...

A. "Yes" because initial ATWS power CANNOT be determined with APRMs unavailable.

B. "Yes" based on Reactor pressure and Turbine Bypass Valve response.

C. "No" based on Reactor pressure and Turbine Bypass Valve response.

D. "No" based on IRM indication.

| | |
|---|---|
| Proposed Answer | **C** |
| Applicant References | **None** |
| Explanation | **Initial ATWS power is normally determined based on APRM indications. However, the bases for EO-100-113 step LQ-6 (recording of initial ATWS power) state that "loss of APRM indication, by itself, does not mean that reactor power cannot be determined". The bases state that Reactor pressure and Turbine Bypass Valve response are valid indications to use in determining initial ATWS power. Each Turbine Bypass Valve passes a steam flow equivalent to 4.5% Reactor power when full open. With one Turbine Bypass Valve 50% open, initial ATWS power is <5%.** |

**A** **Incorrect – Other indications, such as Reactor pressure and Turbine Bypass Valve response, may be used to determine initial ATWS power even with a loss of all APRM indications.**

**B** **Incorrect – Reactor pressure and Turbine Bypass Valve response indicate initial ATWS power is <5%.**

**C** **Correct.**

**D** **Incorrect – If IRMs were fully inserted, an indication of downscale on range 3 would support determining initial ATWS power is <5%. However, with the IRMs still driving in, this indication is not yet valid to support such a determination.**

| | |
|---|---|
| 10CFR55 | **43.5** |
| Technical References | **EO-000-113, EO-100-113, TM-OP-083** |
| Learning Objectives | |
| Question Source | **New** |
| Previous NRC Exam | **No** |
| Comments | |

| Exam | SRO | Tier | 2 | Group | 1 | Cognitive Level | High | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 239002 A2.04 SRVs | | | | | Importance | 4.2 | | |
| Statement | Ability to (a) predict the impacts of the following on the RELIEF/SAFETY VALVES; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: ADS actuation | | | | | | | | |

QUESTION 90

Unit 1 has experienced a loss of coolant accident with the following:

- Reactor water level is -165", down fast.
- Reactor pressure is 450 psig, down slow.
- The Automatic Depressurization System (ADS) automatically initiated, however, only two (2) SRVs opened.
- A Reactor Operator has armed and depressed the ADS Manual Initiation pushbuttons, resulting in NO additional SRVs open.

Which one of the following describes the required direction to be given to control Reactor pressure, in accordance with the Emergency Operating Procedures?

Direct...

A. opening additional SRVs until a total of four (4) SRVs are open.

B. opening additional SRVs until a total of six (6) SRVs are open.

C. lowering Reactor pressure using alternate depressurization systems.

D. placing Shutdown Cooling in service once the interlock clears. No additionally SRVs or alternate depressurization systems are required to be utilized.

Proposed Answer **B**

Applicant References **None**

Explanation **With Reactor water level at -165", down fast, both ADS actuation and entry to EO-100-112, Rapid Depressurization, are required. The ADS actuation has partially failed, since 6 SRVs should have opened. EO-100-112 requires opening additional SRVs until a total of 6 SRVs are open.**

**A Incorrect – Four SRVs is the minimum number required for rapid depressurization, however it is required to direct opening six SRVs.**

**B Correct.**

**C Incorrect – Using alternate depressurization systems is only required if less than four SRVs can be opened and Reactor pressure remains ≥95 psig above Supression Chamber pressure.**

**D Incorrect – Additional SRVs are required to be opened. Shutdown Cooling is to be eventually placed in service, but opening additional SRVs must be attempted first.**

| | | |
|---|---|---|
| 10CFR55 | **43.5** | |
| Technical References | **EO-100-102, EO-100-112** | |
| Learning Objectives | **14593, 14594** | |
| Question Source | **Bank** | **LOC24 Cert #87** |
| Previous NRC Exam | **No** | |
| Comments | | |

| Exam | SRO | Tier | 2 | Group | 2 | Cognitive Level | High | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | | 239001 A2.07 Main and Reheat Steam | | | | Importance | | 3.9 | |
| Statement | | Ability to (a) predict the impacts of the following on the MAIN AND REHEAT STEAM SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Main steam area high temperature or differential temperature high | | | | | | | |

QUESTION 91

Unit 1 is operating at 3% power during a startup with the following:

- Main Steam Line Rad Monitor readings are slightly elevated.
- A steam leak occurs in the Turbine Building Steam Tunnel.
- Turbine Building Steam Tunnel temperature is 201°F, up slow.
- The MSIVs are open.

Which one of the following identifies the required action(s) to direct and the required procedure entry?

A. Close the MSIVs. A Reactor scram is NOT required.
Enter ON-184-001, Main Steam Line Isolation and Quick Recovery.

B. Scram the Reactor and close the MSIVs.
Enter ON-100-101, Scram, Scram Imminent.

C. Trip the Mechanical Vacuum Pump. A Reactor scram is NOT required.
Enter ON-143-001, Main Condenser Vacuum and Offgas System Off-Normal Operation.

D. Scram the Reactor and trip the Mechanical Vacuum Pump.
Enter ON-143-001, Main Condenser Vacuum and Offgas System Off-Normal Operation, and ON-100-101, Scram, Scram Imminent.

Proposed Answer **B**

Applicant References **None**

Explanation **Steam tunnel temperature above 191°F should have already caused MSIV closure. Since this automatic action has failed, it is required to direct MSIVs closed manually. With Reactor power 3% during a startup, the Reactor mode switch is in Startup and not yet taken to Run. Therefore the MSIV closure scram is still bypassed. However, after closing MSIVs, Reactor pressure will rise and cause a scram. Therefore, a Reactor scram is required and ON-100-101 must be entered.**

**A** **Incorrect – MSIVs are required to be closed, however if they are closed without a manual Reactor scram, then an automatic Reactor scram will occur on high Reactor pressure.**

**B** **Correct.**

**C** **Incorrect – If Main Steam Line rad monitor readings were to continue to rise above the hi-hi setpoint, then MVP isolation would be required. Nothing requires tripping the MVP in this situation.**

**D** **Incorrect – If Main Steam Line rad monitor readings were to continue to rise above the hi-hi setpoint, then MVP isolation would be required. Nothing requires tripping the MVP in this situation.**

| | | |
|---|---|---|
| 10CFR55 | **43.5** | |
| Technical References | **AR-111-B03, ON-184-001, ON-100-101** | |
| Learning Objectives | | |
| Question Source | **Bank** | **LOC25 Cert #91** |
| Previous NRC Exam | **No** | |
| Comments | | |

| Exam | SRO | Tier | 2 | Group | 2 | Cognitive Level | High | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 230000 2.4.41 RHR/LPCI: Torus/Pool Spray Mode | | | | | Importance | 4.6 | | |
| Statement | Knowledge of the emergency action level thresholds and classifications. | | | | | | | | |

QUESTION 92

Unit 1 is operating at 100% power with the following:

- Indications are received of a small coolant leak in the Drywell.
- A manual Reactor scram is inserted.
- Unidentified Drywell leakage is 40 gpm, up slow.
- Drywell pressure has just reached the threshold requiring Suppression Chamber sprays per EO-100-103, Primary Containment Control, and is up slow.

Which one of the following describes the required Emergency Action Level (EAL) declaration, if any, in accordance with EP-TP-001, EAL Classification Levels?

A. None

B. Notification of Unusual Event

C. Alert

D. Site Area Emergency

Proposed Answer: **C**

Applicant References: **EP-TP-001, EAL Classification Levels**

Explanation: **Since Suppression Chamber sprays have been ordered due to Drywell pressure, Drywell pressure must be >1.72 psig. Drywell pressure ≥ 1.72 psig and indication of a RCS leak inside Drywell defines a loss of the RCS barrier in EP-TP-001 Table F. Loss of the RCS barrier requires declaration of an Alert.**

**A Incorrect – Although unidentified Drywell leakage has not exceeded an EAL threshold, Drywell pressure has. Therefore, an Alert declaration is required.**

**B Incorrect – Loss of the RCS barrier requires declaration of an Alert, not an NOUE. Loss of the Primary Containment barrier only requires declaration of an NOUE.**

**C Correct.**

**D Incorrect – An SAE is only required if a second fission product barrier is lost/potentially lost. There are no indications of loss/potential loss of either the fuel clad or the Primary Containment.**

10CFR55: **43.5**

Technical References: **EP-TP-001, EAL Classification Levels**

Learning Objectives:

Question Source: **New**

Previous NRC Exam: **No**

Comments

| Exam | SRO | Tier | 2 | Group | 2 | Cognitive Level | High | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | | 271000 2.2.12 Off-gas | | | | Importance | | 4.1 | |
| Statement | | Knowledge of surveillance procedures. | | | | | | | |

QUESTION 93

Both Units are operating at 100% power with the following:

- It is discovered that Technical Specification 3.7.5, Main Condenser Offgas, Surveillance Requirement 3.7.5.1 has NOT been completed on time.
- The surveillance CANNOT be immediately completed due to instrumentation issues.
- The required surveillance frequency is 31 days.
- The surveillance was last performed 48 days ago.
- A risk evaluation has been performed and the associated impact is being managed.

Which one of the following describes the required administrative control of the associated LCO, 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 14 days from the time of discovery or then the associated LCO must be declared NOT met.

D. Complete the surveillance within a maximum of 31 days from the time of discovery or then the associated LCO must be declared NOT met.

Proposed Answer: **D**

Applicant References: **None**

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, 31 days), whichever is greater. Therefore up to 31 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.**

**B Incorrect - SR 3.0.3 allows the longer of 24 hours or the specified frequency (31 days), as long as a risk assessment is performed, which is stated in the question stem.**

**C Incorrect - SR 3.0.3 allows the longer of 24 hours or the specified frequency (31 days), as long as a risk assessment is performed, which is stated in the question stem. 14 days is based on the remaining number of days until the surveillance reaches twice the normal frequency.**

**D Correct.**

| | | |
|---|---|---|
| 10CFR55 | **43.2** | |
| Technical References | **Technical Specification Surveillance Requirement 3.0.3** | |
| Learning Objectives | **15983** | |
| Question Source | **Bank** | **JAF 9/14 NRC #99** |
| Previous NRC Exam | **Yes** | **JAF 9/14/ NRC #99** |
| Comments | | |

Q 94 - contains SSI

Not for public disclosure

| Exam | SRO | Tier | 3 | Group | N/A | Cognitive Level | High | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 2.2.5 | | | | | Importance | 3.2 | | |
| Statement | Knowledge of the process for making design or operating changes to the facility. | | | | | | | | |

QUESTION 95

Which one of the following Switching Orders requires a 10CFR50.59 review per OI-AD-085, Operations Switching Orders?

A. HPCI is out-of-service with a clearance applied to the steam supply and injection valves. The Switching Order will direct the NPO to open the vacuum pump supply breaker.

B. System Engineering wants to adjust the air flow on the Refuel Floor. The Switching Order will direct the NPO to throttle Reactor Building Zone III Exhaust Dampers HD-17538A and HD-17538B by 10 degrees.

C. Engineering is installing a Temporary Modification to the MSL Radiation Monitors. The Switching Order will implement the Temporary Modification by directing the NPO to remove fuses to de-energize the radiation monitors.

D. Operations is investigating in-leakage into the Suppression Pool. The NPO will open the supply breaker for the Core Spray minimum flow valve, and then manually close the valve. The associated Core Spray Loop has been declared inoperable and a clearance applied to the loop suction and injection valves.

Proposed Answer **B**

Applicant References **None**

Explanation **Switching Orders are used to control plant evolutions or manipulate plant components when there is no other pre-existing document to control the event. OI-AD-085 requires a 50.59 review unless one of the following exceptions are met:**

- **The Switching Order is coordinating activities that are controlled by other existing documents.**
- **The Switching Order is directing component manipulations on equipment that is out of service and within the boundary of a clearance order.**
- **The Switching Order is directing component manipulations on equipment that supports the station in performance of maintenance activities (such as hoists or cranes).**

**Adjusting air flow on the Refueling Floor does not meet any of these exceptions, and therefore requires a 50.59 review.**

**A** **Incorrect – This Switching Order is controlling a component manipulation within the boundary of a clearance order, and is therefore exempt from the requirement for a 50.59 review.**

**B** **Correct.**

**C** **Incorrect – Since the manipulation is part of a Temporary Modification, the Switching Order is exempt from the requirement for a 50.59 review.**

**D** **Incorrect – This Switching Order is controlling a component manipulation within the boundary of a clearance order, and is therefore exempt from the requirement for a 50.59 review.**

| | | |
|---|---|---|
| 10CFR55 | **43.3** | |
| Technical References | **OI-AD-085** | |
| Learning Objectives | **AD044: 14891** | |
| Question Source | **Bank** | **LOC24 NRC #97** |
| Previous NRC Exam | **Yes** | **LOC24 NRC #97** |
| Comments | | |

| Exam | SRO | Tier | 3 | Group | N/A | Cognitive Level | Low | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 2.3.13 | | | | | Importance | 3.8 | | |
| Statement | Knowledge of Radiological Safety Procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high radiation areas, aligning filters, etc. | | | | | | | | |

QUESTION 96

Unit 1 is operating at 2% power with the following:

- Maintenance personnel have entered the Drywell to perform emergent repairs on elevation 738'.
- Due to expected Xenon burnout, Reactor power begins to slowly rise.

Which one of the following describes the required control of Reactor power and control rod insertion in accordance with NDAP-QA-0309, Primary Containment Access and Control?

Reactor power must remain less than or equal to a maximum value of __(1)__ to continue allowing access to Drywell elevation 738'.

If reactor power approaches the limit while personnel are on Drywell elevation 738', inserting control rods with RMCS is __(2)__.

| | (1) | (2) |
|---|---|---|
| A. | 3% | allowed |
| B. | 3% | NOT allowed |
| C. | 10% | allowed |
| D. | 10% | NOT allowed |

Proposed Answer: **A**

Applicant References: **None**

Explanation: **NDAP-QA-0309 requires Reactor power to be ≤3% for access to Drywell elevation 738'. NDAP-QA-0309 allows inserting control rods with RMCS while personnel are in the Drywell as long as Reactor power is ≤3%.**

**A** **Correct.**

**B** **Incorrect – Reactor power changes are restricted while personnel are in the Drywell, however control rod insertion is allowed with personnel in the Drywell as long as Reactor power is ≤3%.**

**C** **Incorrect – 10% is the Reactor power threshold for lower Drywell elevations, but 3% is the threshold for Drywell elevation 738'.**

**D** **Incorrect - 10% is the Reactor power threshold for lower Drywell elevations, but 3% is the threshold for Drywell elevation 738'. Reactor power changes are restricted while personnel are in the Drywell, however control rod insertion is allowed with personnel in the Drywell as long as Reactor power is ≤3%.**

| | | |
|---|---|---|
| 10CFR55 | **43.4** | |
| Technical References | **NDAP-QA-0309** | |
| Learning Objectives | **15314** | |
| Question Source | **Modified Bank** | **LOC23 NRC #100** |
| Previous NRC Exam | **No** | |
| Comments | | |

| Exam | SRO | Tier | 3 | Group | N/A | Cognitive Level | High | Level of Difficulty | 2.5 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 2.4.6 | | | | | Importance | 4.7 | | |
| Statement | Knowledge of EOP mitigation strategies. | | | | | | | | |

QUESTION 97

Unit 1 has experienced a failure to scram with the following:

- The Reactor Mode Switch is in SHUTDOWN.
- ARI has been initiated.
- Reactor power is 15%, steady.
- Reactor water level is 30", steady, with Feedwater injecting.
- Reactor pressure is 1025 psig, steady, with the Main Generator still online.

Which one of the following describes the required strategy for control of reactor injection in accordance with EO-100-113, Level/Power Control?

Reactor injection ___(1)___ required to be throttled.

Injection using systems that inject inside the core shroud is ___(2)___.

| | (1) | (2) |
|---|---|---|
| A. | is | restricted |
| B. | is | NOT restricted |
| C. | is NOT | restricted |
| D. | is NOT | NOT restricted |

Proposed Answer **A**

Applicant References **None**

Explanation **EO-100-113 contains different strategies for control of Reactor injection depending on plant conditions. With Reactor power above 5% and Reactor water level above -60", Reactor injection must be throttled to lower level <-60". Injection systems are limited to those listed on Table 15. Table 15 does not include Core Spray, because Core Spray injects inside the core shroud. Injection with systems inside the core shroud is restricted to prevent power excursions.**

**A Correct.**

**B Incorrect – Injection with systems that inject inside the core shroud is restricted, as evidenced by Table 15 not including Core Spray.**

**C Incorrect – Reactor injection is required to be throttled because Reactor power is >5% and Reactor water level is >-60 inches.**

**D** **Incorrect – Reactor injection is required to be throttled because Reactor power is >5% and Reactor water level is >-60 inches. Injection with systems that inject inside the core shroud is restricted, as evidenced by Table 15 not including Core Spray.**

| | | |
|---|---|---|
| 10CFR55 | **43.5** | |
| Technical References | **EO-100-113** | |
| Learning Objectives | | |
| Question Source | **Modified Bank** | **JAF 9/12 NRC #75** |
| Previous NRC Exam | **No** | |
| Comments | | |

| Exam | SRO | Tier | 3 | Group | N/A | Cognitive Level | Low | Level of Difficulty | 3.5 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 2.3.14 | | | | | Importance | 3.8 | | |
| Statement | **Knowledge of radiation or containment hazards that may arise during normal, abnormal, or emergency conditions or activities.** | | | | | | | | |

QUESTION 98

Which one of the following identifies a parameter and associated threshold reading used to define <u>Loss of the Fuel Clad Barrier</u>, in accordance with EP-TP-001, EAL Classification Levels, Table F, Fission Product Barrier Degradation?

A. Drywell radiation; 7 R/hr

B. Drywell radiation; 3000 R/hr

C. Offgas radiation; hi alarm setpoint

D. Offgas radiation; hi-hi alarm setpoint

Proposed Answer **B**

Applicant References **None**

**<u>Note:</u> The provided reference for Question 92 must have EP-TP-001 Table F Row e deleted to avoid making this a direct lookup.**

Explanation **EP-TP-001 Table F identifies Drywell radiation >3000 R/hr as the threshold for Loss of the Fuel Clad Barrier.**

**A Incorrect - 3000 R/hr is the threshold value, NOT 7 R/hr. 7 R/hr is used in Table F as part of the definition for loss of the RCS barrier.**

**B Correct.**

**C Incorrect - Offgas radiation monitoring is used to detect fuel clad degradation, both in EP-TP-001 and in ON-179-001. However, Offgas radiation monitoring is not used to define loss of the fuel clad barrier in Table F.**

**D Incorrect - Offgas radiation monitoring is used to detect fuel clad degradation, both in EP-TP-001 and in ON-179-001. However, Offgas radiation monitoring is not used to define loss of the fuel clad barrier in Table F.**

10CFR55 **43.4**

Technical References **EP-TP-001**

Learning Objectives

Question Source **Bank JAF 4/14 NRC #99**

Previous NRC Exam **Yes JAF 4/14 NRC #99**

Comments **TRH 1/6/15 – Replaced question based on NRC comment.**

| Exam | SRO | Tier | 3 | Group | N/A | Cognitive Level | Low | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 2.2.7 | | | | | Importance | 3.6 | | |
| Statement | **Knowledge of the process for conducting special or infrequent tests.** | | | | | | | | |

QUESTION 99

A Special, Infrequent Or Complex Test/Evolution (SICT/E) is to be performed.

Which one of the following identifies the positions required to provide the initial briefings for the SICT/E in accordance with NDAP-QA-0320, Special, Infrequent Or Complex Test/Evolutions?

A. Unit Supervisor and Shift Manager

B. Shift Manager and Senior Manager

C. Test/Evolution Coordinator and Unit Supervisor

D. Senior Manager and Test/Evolution Coordinator

Proposed Answer **D**

Applicant References **None**

Explanation **NDAP-QA-0320 specifically requires the initial briefing to be provided by the Test/Evolution Coordinator and the Senior Manager.**

**A Incorrect – The Unit Supervisor and Shift Manager would be expected to attend the brief, however NDAP-QA-0320 specifically requires the initial briefing to be provided by the Test/Evolution Coordinator and the Senior Manager.**

**B Incorrect - The Shift Manager would be expected to attend the brief, however NDAP-QA-0320 specifically requires the initial briefing to be provided by the Test/Evolution Coordinator and the Senior Manager.**

**C Incorrect - The Unit Supervisor would be expected to attend the brief, however NDAP-QA-0320 specifically requires the initial briefing to be provided by the Test/Evolution Coordinator and the Senior Manager.**

**D Correct.**

10CFR55 **43.3**

Technical References **NDAP-QA-0320**

Learning Objectives **14657**

Question Source **Bank** **Vision SYSID 494**

Previous NRC Exam **No**

Comments

| Exam | SRO | Tier | 3 | Group | N/A | Cognitive Level | High | Level of Difficulty | 3 |
|---|---|---|---|---|---|---|---|---|---|
| K/A | 2.1.7 | | | | | Importance | 4.7 | | |
| Statement | Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation. | | | | | | | | |

QUESTION 100

Unit 1 is operating at 100% power with the following:

- OPRMs are inoperable.
- All required LPRM upscale alarms are operable.
- Total core flow is 102 Mlbm/hr, steady.

Then, the 1A Reactor Recirc Pump (RRP) trips. The following conditions now exist:

- APRMs indicate 62%, steady.
- Total core flow is 50 Mlbm/hr, steady.
- The cause of the 1A RRP trip has been determined and corrected.

**Note:** The Power/Flow Map and Core Flow vs. Core Pressure Drop curve are provided on the following pages.

Which one of the following describes the required direction to be given in accordance with the Off Normal (ON) procedures?

A. Insert control rods using RMCS.

B. Raise total core flow using 1B RRP.

C. Raise total core flow by re-starting 1A RRP.

D. Place the Reactor Mode Switch in SHUTDOWN.

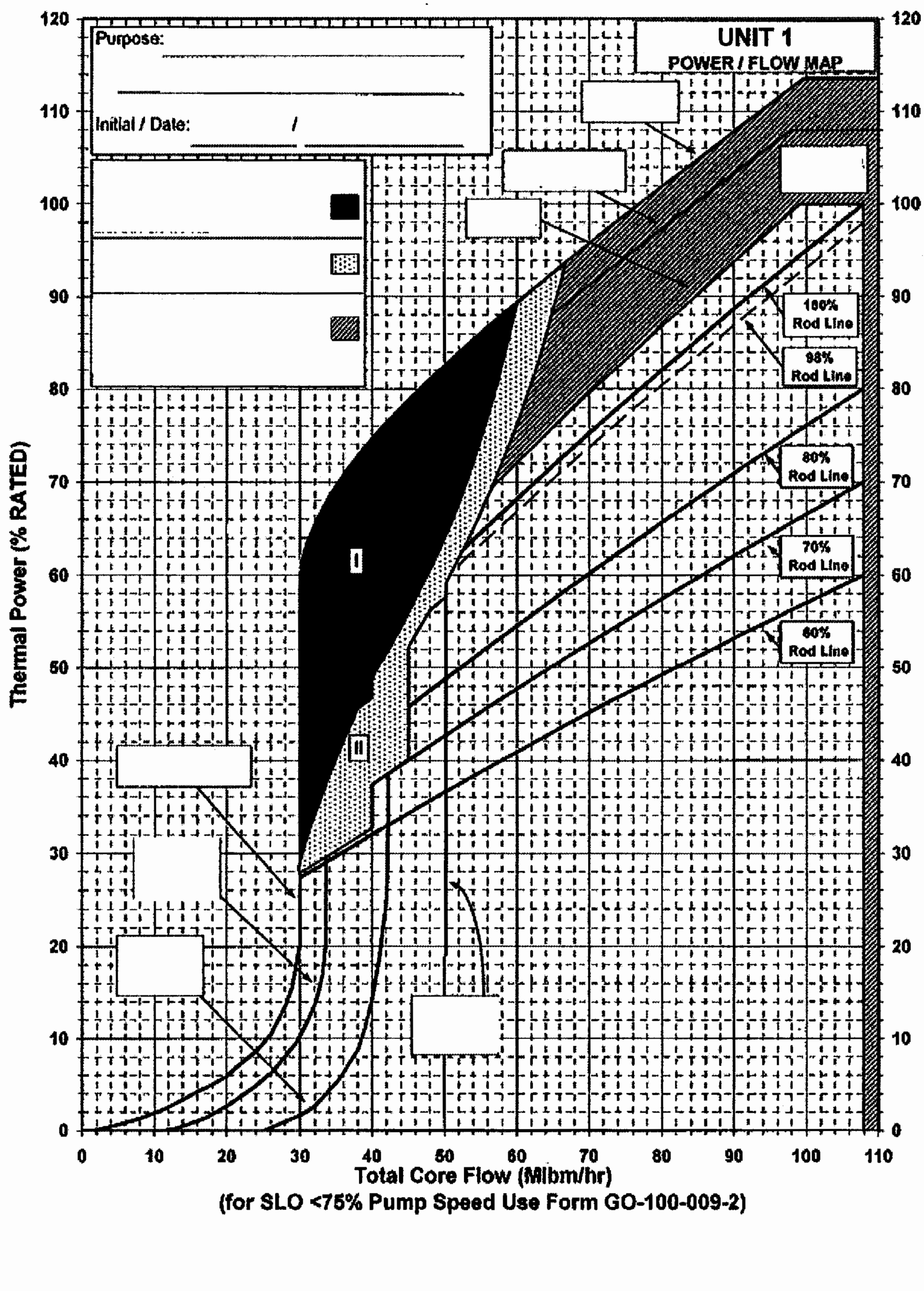
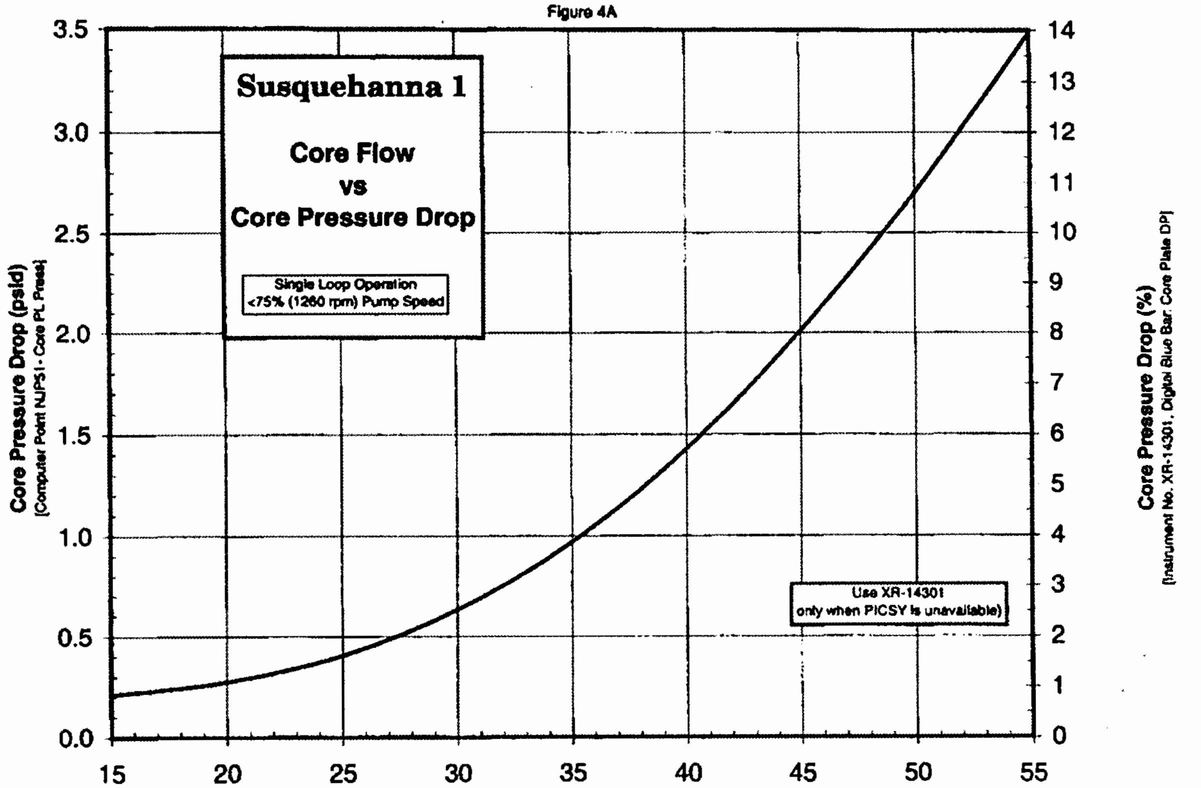

UNIT 1
POWER / FLOW MAP
Purpose:
Initial / Date: /
I
II
100% Rod Line
98% Rod Line
80% Rod Line
70% Rod Line
60% Rod Line
Thermal Power (% RATED)
Total Core Flow (Mlbm/hr)
(for SLO <75% Pump Speed Use Form GO-100-009-2)
0
10
20
30
40
50
60
70
80
90
100
110
120

Figure 4A




[Single Loop Operation - for use on Power/Flow Map only; Not valid for thermal limit calculations]

Ref: NFE-B-NA-068 Rev. 22

Proposed Answer A

Applicant References None

Explanation The given conditions require entry into multiple Off Normal (ON) procedures, including ON-164-002, Loss of Reactor Recirculation Flow, ON-156-001, Unanticipated Reactivity Change, and ON-178-002, Core Flux Oscillations. Plant conditions following the trip of the 1A RRP are in Region 2 of the P/F map. With the OPRMs inoperable and conditions in Region 2 of the P/F map, ON-178-002 requires action to exit Region 2. Raising speed on the 1B RRP is not an allowable option because it is already operating at the upper limit cautioned in ON-178-002.

A Correct.

B Incorrect - Raising speed on the 1B RRP is not an allowable option in these conditions because it is already operating at the upper limit cautioned in ON-178-002.

C Incorrect – Starting a RRP is not one of the authorized methods for rapidly exiting Region 2 of the P/F Map in ON-178-002. Additionally, ON-164-001 does not allow restart of a RRP with rod line above 60% per ON-164-002.

D Incorrect - Since APRMs are steady, there are no observed core flux oscillations, therefore a Reactor scram is NOT required. Additionally, with rod line >60%, trip of 1B RRP would require a Reactor scram per ON-164-002. The same limit does not apply to trip of 1A RRP.

10CFR55 43.5

| | | |
|---|---|---|
| Technical References | **ON-164-002, ON-178-002** | |
| Learning Objectives | **14908** | |
| Question Source | **Bank** | **LOC23 NRC #76** |
| Previous NRC Exam | **Yes** | **LOC23 NRC #76** |
| Comments | | |