**Form 3.2-1 Administrative Topics Outline**

| Facility: Nine Mile Point Unit 1 | | Date of Examination: AUG 2024 |
|---|---|---|
| Examination Level: RO | | Operating Test Number: 2024 |
| Administrative Topic (Step 1) | Activity and Associated K/A (Step 2) | Type Code (Step 3) |
| Conduct of Operations | Perform Daily Checks – RPV Level Instruments<br>K/A 2.1.19 (3.9), N1-ST-D0 | S, D |
| Conduct of Operations | Perform Heat-up Rate Determination<br>K/A 2.1.7 (4.4), N1-OP-43A | R, P, D |
| Equipment Control | Develop Clearance - RBCLC<br>K/A 2.2.15 (3.9), OP-AA-109-101 | R, D |
| Radiation Control | Radiological Requirements - Drywell<br>K/A 2.3.12 (3.2), RP-AA-203 | R, N |
| Emergency Plan | | |

**Location**:

(C)ontrol room, (S)imulator, or Class(R)oom

**Source and Source Criteria**:

(P)revious two NRC exams (no more than one JPM that is **randomly selected** from last two NRC exams)

(D)irect from bank (no more than three for ROs, no more than four for SROs and RO retakes)

(N)ew or Significantly (M)odified from bank (no fewer than one)

**Form 3.2-1 Administrative Topics Outline**

| Facility: Nine Mile Point Unit 1 | | Date of Examination: AUG 2024 |
|---|---|---|
| Examination Level: SRO | | Operating Test Number: 2024 |
| Administrative Topic (Step 1) | Activity and Associated K/A (Step 2) | Type Code (Step 3) |
| Conduct of Operations | Determine Operator Qualifications Using LMS<br>K/A 2.1.8 (4.1), OP-AA-109-101 | R, D |
| Conduct of Operations | Reactivate SRO Licenses<br>K/A 2.1.4 (3.8), OP-AA-105-102 | R, D |
| Equipment Control | Review Completed Surveillance N1-ST-M8<br>K/A 2.2.12 (4.1), N1-ST-M8 | R, N |
| Radiation Control | Determine Actions for Inoperable Stack Radiation Monitor<br>K/A 2.3.11 (4.3), CY-NM-170-301 | R, D |
| Emergency Plan | Complete Part 1 Notification Fact Sheet<br>K/A 2.4.44 (4.4), EP-CE-111, EP-CE-114 | R, N |

**Location**:

(C)ontrol room, (S)imulator, or Class(R)oom

**Source and Source Criteria**:

(P)revious two NRC exams (no more than one JPM that is **randomly selected** from last two NRC exams)

(D)irect from bank (no more than three for ROs, no more than four for SROs and RO retakes)

(N)ew or Significantly (M)odified from bank (no fewer than one)

*Form 3.2-2 Control Room/In-Plant Systems Outline*

Facility: Nine Mile Point Unit 1 Date of Examination: Aug 2024

Operating Test Number: LC1 23-1NRC

Exam Level: RO / SRO-I / **SRO-U**

| System/JPM Title | Type Code | Safety Function |
|---|---|---|
| **Control Room Systems** | | |
| a. Shift Reactor Building Operating Exhaust and Supply Fans<br>K/A 288000 A4.01 (3.2), N1-OP-10 | P, D, S, A | 9 |
| b. **Place 11 Shutdown Cooling Loop in Service**<br>**K/A 205000 A4.01 (3.9), N1-OP-4** | **P, D, S, A, L** | 4 |
| c. Secure a Reactor Recirculation Pump, Pump Trips<br>K/A 202001 A4.01 (4.0), N1-OP-1, N1-SOP-1.3 | D, A, S | 1 |
| d. Sequential Loss of Service Water<br>K/A 295018 AA1.01 (3.3), N1-SOP-18.1 | D, A, S | 8 |
| e. Swap from EPR to MPR at power<br>K/A 241000 A1.14 (3.7), N1-OP-31, N1-SOP-31.2 | D, S | 3 |
| **f. Vent Torus through RBEVS**<br>**K/A 223001 K1.17 (3.2), N1-EOP-4.1** | **N, E, S, EN** | **5** |
| **g. Adjust MVAR followed by Grid Disturbance**<br>**K/A 700000 AA2.11 (3.5), N1-OP-31, N1-SOP-31.2** | **N, A, E, S** | **6** |
| h. Lineup and Inject Containment Spray Raw Water into Core Spray Loop 11 (RO only)<br>K/A 209001, AA4.01 (4.3), N1-EOP-1, Att. 5 | D, S, EN | 2 |
| **In-Plant Systems** | | |
| i. **Transfer Reactor Trip Bus 141 to I&C Bus 130A**<br>K/A 295003 AA1.01 (4.0), N1-SOP-30.2 | **D, E, R** | **7** |

| | | |
|---|---|---|
| j. **Lineup Liquid Poison to inject to RPV from the Demin Tank**<br>**K/A 295031 EA1.08 (3.9), N01-EOP-1, Att. 12** | **M, E, R** | **2** |
| k. Lineup Lake Water to Supply the Emergency Condenser Makeup Tanks using the Electric Fire Pump<br>K/A 295016 AA1.09 (4.0), N1-SOP-21.2 | D, E, A, R | 4 |

| Code | License Level Criteria | | |
|---|---|---|---|
| | **RO** | **SRO-I** | **SRO-U** |
| (A)lternate path | 4–6 | 4–6 | 2–3 |
| (C)ontrol room | | | |
| (D)irect from bank | ≤ 9 | ≤ 8 | ≤ 4 |
| (E)mergency or abnormal in-plant | ≥ 1 | ≥ 1 | ≥ 1 |
| (EN)gineered safety feature (for control room system) | ≥ 1 | ≥ 1 | **≥ 1** |
| (L)ow power/shutdown | ≥ 1 | ≥ 1 | ≥ 1 |
| (N)ew or (M)odified from bank (must apply to at least one alternate path JPM) | ≥ 2 | ≥ 2 | ≥ 1 |
| (P)revious two exams (randomly selected) | ≤ 3 | ≤ 3 | ≤ 2 |
| (R)adiologically controlled area | ≥ 1 | ≥ 1 | ≥ 1 |
| (S)imulator | | | |

**Form 3.3-1 Scenario Outline**

| | | | |
|---|---|---|---|
| Facility: | Nine Mile Point Unit 1 | Scenario #: | NRC-1 |
| Scenario Source: | New | Op. Test #: | 2024 |
| Examiners: | | Applicants/ Operators: | |
| Initial Conditions: | The plant is operating at approximately 100% power. Containment Spray Pump 111 is out of service for maintenance. | | |
| Turnover: | Cross-tie PB 16A to PB 16B per N1-OP-30 starting at H.21.1.c. PB-11 will remain aligned to reserve power. Then, reduce reactor power approximately 5% using Reactor Pumps per provided instructions. | | |
| Critical Tasks: | CT-1: Given an inadvertently open ERV at power, close the ERV or insert a manual scram prior to Torus temperature exceeding 110°F, in accordance with N1-SOP-1.4.<br><br>CT-2: Given the plant with a primary system discharging outside the primary and secondary containments, which cannot be isolated, and the offsite release rate is projected to exceed the E-Plan "General Emergency" level, the crew will perform an RPV blowdown before the offsite release rate reaches the Emergency Plan “General Emergency” level in accordance with N1-EOP-RR (1-2), and N1-EOP-C2 (1-2). | | |

| Event No. | Malf. No. | Event Type* | Event Description |
|---|---|---|---|
| 1 | N/A | N – BOP, SRO | Cross-tie PB 16A to PB 16B<br>N1-OP-30 |
| 2 | N/A | R - ATC | Reduce reactor power with Reactor Recirculation Pumps<br>N1-OP-43B |
| 3 | RR72 | I – All<br>MC - BOP<br>TS-SRO | Respond to Flow Oscillations on RRP #13, Discharge valve fails to close<br>N1-SOP-1.5, N1-SOP-1.3; Technical Specification |
| 4 | AD05A | C – All<br>TS – SRO | Stuck Open ERV 111<br>**(2020 Scenario 3)** N1-SOP-1.4, Technical Specifications |
| 5 | EG13 | C = All | Rising Generator Stator Water Cooling Conductivity<br>N1-ARP-A2, N1-SOP-31.1, N1-SOP-1, N1-EOP-RPV |
| 6 | ED27 | C-All<br>MC-ATC | Power Board 12 fails to Fast Transfer, FW Pump 11 Trips<br>**(2020 Scenario 1)** N1-SOP-30.2 |
| 7 | MS12 | M – All | Main Steam Line Break in Turbine Building with Fuel Element Failure<br>N1-EOP-SC/RR |
| 8 | MS13A/C | C – All<br>MC - BOP | MSIV fail to close and isolate the steam leak to the Turbine Building<br>**(2020 Scenario 1)** N1-EOP-SC/RR, N1-EOP-C2 |
| 9 | HV07E/F | C – ATC | Turbine Building Ventilation Exhaust Fans Trip<br>**(2020 Scenario 1)** N1-EOP-SC/RR, N1-EOP-C2 |

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Tech Spec, (MC)Manual Control

| Facility: **Nine Mile Point Unit 1** Scenario No.: **NRC-1** Op-Test No.: **2024** | | |
|---|---|---|
| 1. Malfunctions after EOP entry (1-2)<br>**Events 6, 7, 8, 9** | 4 | |
| 2. Abnormal events (2-4)<br>**Events 3, 4, 5, 6** | 4 | |
| 3. Major transients (1-2)<br>**Event 7** | 1 | |
| 4. EOPs entered/requiring substantive actions (1-2)<br>**EOP-RPV, EOP-SC/RR** | 2 | |
| 5. Entry into a contingency EOP with substantive actions (≥1 per scenario set)<br>**EOP-C2** | 1 | |
| 6. Pre-identified critical tasks (≥2) | 2 | |

## Critical Task Justifications

| ***Critical Task (#1)*** | **Given an inadvertently open ERV at power, close the ERV or insert a manual scram prior to Torus temperature exceeding 110°F, in accordance with N1-SOP-1.4.** |
|---|---|
| *CT-1.0 Justification:* | |
| *Safety Significance* | *A manual Reactor scram is required before Torus temperature exceeds 110°F. This reduces the rate of energy production and thus heat input to the Torus. Additionally, this allows evaluating the success of the Reactor scram before boron injection would be required due to Torus temperature in the event of a failure to scram. Closing the ERV prior to the need for the scram avoids the need for these more substantial actions, prevents challenging the plant with a scram, and stops heat input to the Torus.* |
| *Initiating Cue:* | *ERV position, ERV acoustic monitors, ERV tailpipe temperature, Torus temperature, Reactor pressure, and steam flow indicate an open ERV. N1-SOP-1.4 provides direction to close the ERV or scram the Reactor.* |
| *Success Path:* | *The crew will perform the actions of N1-SOP-1.4 to successfully close the affected ERV.* |
| *Performance Feedback:* | *ERV position, ERV acoustic monitors, ERV tailpipe temperature, Torus temperature, Reactor pressure, and steam flow will provide performance feedback regarding success of crew actions to close the ERV. Control rod position and Reactor power will provide performance feedback regarding success of crew actions to scram the Reactor.* |
| *Measurable Performance Standard:* | *Pulling ERV fuses, directing an operator to pull ERV fuses in the field, rotating the depressing the manual scram pushbuttons, and/or Mode Switch to SHUTDOWN will provide observable actions for the evaluation team.* ***Bounding Criteria:*** *Based on procedure requirements of N1-SOP-1.4 to scram the reactor prior to Torus temperature exceeding 110°F.* |

| ***Critical Task (#2)*** | **Given the plant with a primary system discharging outside the primary and secondary containments, which cannot be isolated, and the offsite release rate is projected to exceed the E-Plan "General Emergency" level, the crew will perform an RPV blowdown before the offsite release rate reaches the Emergency Plan "General Emergency" level in accordance with N1-EOP-RR (1-2), and N1-EOP-C2 (1-2).** |
|---|---|
| *CT-2.0 Justification:* | |
| *Safety Significance* | *With an unisolable main steam isolation valve and a steam leak into the Turbine Building, a primary system is discharging outside both the primary and secondary containment. This discharge is approaching E-Plan general emergency levels which is an indication that all 3 fission product barriers have been breached and that dose to the public is in progress. Blowing down the RPV will both take away the driving head and send the radionuclides through the Torus to be "scrubbed", thus reducing the release to the public.* |
| *Initiating Cue:* | *The Offsite release rate has already exceeded the "Alert" level of the Emergency Plan and release rate calculations are projected to exceed the "General Emergency" Emergency Plan level.* |
| *Success Path:* | *The crew will perform a RPV blowdown in accordance with N1-EOP-C2 (1-2) to lower pressure within the capacity of all available injection sources.* |
| *Performance Feedback:* | *Emergency Condenser and ERV instrumentation will provide indication that these systems are functioning properly once placed in service. Multiple Reactor pressure indicators and annunciators will provide performance feedback regarding the success of the blowdown.* |
| *Measurable Performance Standard:* | *Manipulation of Emergency Condenser valve and ERV control switches will provide observable actions for the evaluation team.* ***Bounding Criteria:*** *Based on procedural requirements to perform an RPV Blowdown prior to the offsite release rate reaches the Emergency Plan "General Emergency" level in accordance with N1-EOP-RR (1-2)* |

**Form 3.3-1 Scenario Outline**

| | | | |
|---|---|---|---|
| Facility: | Nine Mile Point Unit 1 | Scenario #: | NRC-2 |
| Scenario Source: | New | Op. Test #: | 2024 |
| Examiners: | | Applicants/ Operators: | |
| Initial Conditions: | The plant is operating at approximately 90% power. Containment Spray Pump 111 is out of service for maintenance. | | |
| Turnover: | Place RBCLC pump 13 in service and place RBCLC pump 12 in standby per N1-OP-11. Then, raise power to 100% utilizing Reactor Recirculation pumps per given instructions. | | |
| Critical Tasks: | CT-1: Given an un-isolable RWCU leak outside primary containment and one general area temperature approaching a maximum safe limit, the crew will insert a manual scram prior to any general area he field report indicating one general area temperature exceeding a max safe limit in accordance with N1-EOP-SC (1-2).<br><br>CT-2: Given an un-isolable RWCU leak outside primary containment with two general area temperature above the maximum safe limit, the crew will execute N1-EOP-C2, RPV Blowdown (1-2), in accordance with N1-EOP-SC (1-2) within 15 minutes of receiving the field report of two general area temperatures exceeding max safe limits. | | |

| Event No. | Malf. No. | Event Type* | Event Description |
|---|---|---|---|
| 1 | N/A | N – BOP, SRO | Place RBCLC pump 13 in service and place RBCLC pump 12 to standby.<br>N1-OP-11 |
| 2 | N/A | R – ATC | Raise reactor power with Reactor Recirculation Pumps<br>N1-OP-1 |
| 3 | CW04C | C - BOP, SRO | RBCLC pump 13 trips<br>ARP |
| 4 | RM51A | I – SRO<br>TS-SRO | Main Steam Radiation Monitor Failure<br>ARP F1-2-7, Technical Specifications |
| 5 | ED06 | C – All | Power Board 101 Fault<br>N1-SOP-1.3, Technical Specifications |
| 6 | RP10 | C – All<br>MC - BOP<br>TS - SRO | CRD Flow Control Valve (FCV) fails closed<br>N1-SOP-5.2; Technical Specification |
| 7 | CU11 | M – All | RWCU break in the Secondary Containment requiring reactor scram; RWCU Isolation Valves fail to fully isolate<br>N1-EOP-RPV, N1-EOP-C2, N1-EOP-SC |
| 8 | OVR – DI-4113, 4114 | C – All<br>MC - ATC | RPS pushbuttons fail to scram the reactor; Mode Switch and ARI are successful in scramming the reactor<br>N1-SOP-1 |
| 9 | TC12 | C - All | All TBV's fail closed |

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Tech Spec, (MC)Manual Control

| Facility: **Nine Mile Point Unit 1** Scenario No.: **NRC-2** Op-Test No.: **2024** | | |
|---|---|---|
| 1. Malfunctions after EOP entry (1-2)<br>**Events 8, 9** | 2 | |
| 2. Abnormal events (2-4)<br>**Events 3, 5, 6** | 3 | |
| 3. Major transients (1-2)<br>**Event 7** | 1 | |
| 4. EOPs entered/requiring substantive actions (1-2)<br>**EOP-RPV, EOP-SC** | 2 | |
| 5. Entry into a contingency EOP with substantive actions (≥1 per scenario set)<br>**EOP-C2** | 1 | |
| 6. Pre-identified critical tasks (≥2) | 2 | |

## Critical Task Justifications

| | |
|---|---|
| ***Critical Task (#1)*** | **Given an un-isolable RWCU leak outside primary containment and one general area temperature approaching a maximum safe limit, the crew will insert a manual scram prior to any general area he field report indicating one general area temperature exceeding a max safe limit in accordance with N1-EOP-SC (1-2).** |
| *CT-1.0 Justification:* | |
| *Safety Significance* | With an un-isolable primary system discharging outside of Primary Containment resulting in general area temperature approaching the maximum safe limit, the Reactor must be scrammed. This reduces the rate of energy production and thus the heat input, radioactivity release, and break flow into the Secondary Containment. This also ensures the Reactor is shutdown prior to need for a blowdown. |
| *Success Path:* | The crew will be able to scram the reactor utilizing the Mode Switch and ARI pushbuttons per N1-SOP-1 |
| *Initiating Cue:* | *Annunciators, field reports, and RWCU flow and temperature changes, failure of isolation valves to isolate, will provide indication of the leak coming from the RWCU system.* |
| *Measurable Performance Standard:* | *Manipulation of Reactor Scram Pushbuttons, Mode Switch, and ARI pushbuttons, will provide observable actions for the evaluation team.* ***Bounding Criteria****: Before exceeding any maximum safe temperature in accordance with N1-EOP-SC..* |
| *Performance Feedback:* | *Control rod position and Reactor power indications will provide performance feedback regarding the success rod insertion.* |

| | |
|---|---|
| ***Critical Task (#2)*** | **Given an un-isolable RWCU leak outside primary containment and two general area temperatures above the maximum safe limit, within 15 minutes the crew will commence an RPV Blowdown in accordance with N1-EOP-C2.** |
| *CT-2.0 Justification:* | |
| *Safety Significance* | An un-isolable primary system discharging outside of Primary Containment resulting in two general area temperatures above the maximum safe limit indicates a wide-spread problem posing a direct and immediate threat to Secondary Containment. A blowdown minimizes flow through the break, rejects heat to the suppression pool in preference to outside the containment, and places the primary system in the lowest possible energy state. |
| *Success Path:* | The crew will open 4 ERVs in accordance with N1-EOP-SC (1-2), Secondary Containment Control for Modes 1 and 2, and N1-EOP-C2 (1-2), RPV Blowdown for Modes 1 and 2, to rapidly lower the energy released into the Secondary Containment. |
| *Initiating Cue:* | *Multiple annunciators will provide indications of a primary system discharging into Secondary Containment. Field reports will provide indication that two general area temperature limits are above the maximum safe value. N1-EOP-SC (1-2) provides direction to perform an RPV Blowdown per N1-EOP-C2 (1-2).* |
| *Measurable Performance Standard:* | *The crew will manually initiate Emergency Condensers and open ERVs. This will provide observable actions for the evaluation team.* ***Bounding Criteria:*** *Based on procedural requirements in N1-EOP-SC (1-2) to perform an RPV Blowdown once 2 or more General areas have exceeded their Maximum Safe Values. Time requirement based on Operations representative recommendation.* |
| *Performance Feedback:* | *Emergency Condenser and ERV instrumentation will provide indication that these systems are functioning properly once placed in service. Multiple Reactor pressure indicators and annunciators will provide performance feedback regarding the success of the blowdown.* |

**Form 3.3-1 Scenario Outline**

| | | | |
|---|---|---|---|
| Facility: | Nine Mile Point Unit 1 | Scenario #: | NRC-3 |
| Scenario Source: | New | Op. Test #: | 2024 |
| Examiners: | | Applicants/ Operators: | |
| Initial Conditions: | The plant is operating at approximately 1% power. Containment Spray Pump 111 is out of service for maintenance. | | |
| Turnover: | Place Feedwater Booster Pump #13 in service per N1-OP-16. Raise reactor power to greater than the POAH. Establish and maintain a 75°F/Hr HUR. | | |
| Critical Tasks: | CT-1: Given indications of an Emergency Condenser tube leak, the crew will isolate the Emergency Condenser within 15 minutes of receiving indications of the tube leak per N1-ARP-K1-1-4.<br><br>CT-2: Given a coolant leak inside the Containment, lowering Reactor water level, and failure of injection systems to automatically inject to the Reactor, the crew will inject to the Reactor to restore and maintain Reactor water level above -84", in accordance with N1-EOP-RPV. Reactor water level must not be below -84" for more than 10 minutes. | | |

| Event No. | Malf. No. | Event Type* | Event Description |
|---|---|---|---|
| 1 | N/A | N – BOP, SRO | Place Feedwater Booster Pump #13 in service<br>N1-OP-16 |
| 2 | N/A | R – ATC | Raise reactor power with rods to establish a 75ºF/hr HUR<br>N1-OP-43A |
| 3 | RD42 | C – ATC, SRO | Control Rod Double notches during startup,<br>N1-OP-5 |
| 4 | RM73/ HV04 | I – BOP, SRO<br>**TS - SRO** | RB Vent rad monitor fails upscale, RB vent FTI<br>ARP; Technical Specification(TS) |
| 5 | NM11A | I – ATC, SRO | IRM 11 Fails Downscale,<br>N1-ARP-F2, N1-OP-38B |
| 6 | EC06 | C – BOP, SRO<br>**TS – SRO** | Emergency Condenser #11 Tube Leak - Isolable<br>**(2022 Scenario 4)** ARP; Technical Specifications |
| 7 | RP10 | I – BOP, SRO<br>MC - BOP | Running TBCLC pump trips. TBCLC pump fails to auto start<br>ARP; Technical Specification (TS) |
| 8 | CU11 | M – All | Loss of all Turbine Building Closed Loop Cooling<br>N1-SOP-24.1,N1-EOP-RPV |
| 9 | LP01 | C – All<br>MC - ATC | Loss of Coolant Leak into the Primary Containment<br>N1-EOP-PC, |
| 10 | FW01, CS06 | C – All<br>M - BOP | All Condensate Pumps trip and all Core Spray pumps fail to auto start<br>**(2022 Scenario 4)** N1-ARP-H3, N1-EOP-1, Attachment 4 |

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Tech Spec, (MC)Manual Control

| Facility: **Nine Mile Point Unit 1** Scenario No.: **NRC-3** Op-Test No.: **2024** | | |
|---|---|---|
| 1. Malfunctions after EOP entry (1-2)<br>**Events 9 & 10** | 2 | |
| 2. Abnormal events (2-4)<br>**Events 3, 4, 5, 6, 7** | 5 | |
| 3. Major transients (1-2)<br>**Event 8** | 1 | |
| 4. EOPs entered/requiring substantive actions (1-2)<br>**EOP-RPV, EOP-SC** | 2 | |
| 5. Entry into a contingency EOP with substantive actions (≥1 per scenario set) | 0 | |
| 6. Pre-identified critical tasks (≥2) | 2 | |

## Critical Task Justifications

| | |
|---|---|
| ***Critical Task (#1)*** | **Given indications of an Emergency Condenser tube leak, the crew will isolate the Emergency Condenser within 15 minutes of receiving indications of the tube leak per N1-ARP-K1-1-4.** |
| *CT-1.0 Justification:* | |
| *Safety Significance* | *The EC tube leak results in a radiological release to the atmosphere. The crew will isolate the EC steam line to terminate the release before an emergency declaration is required.* |
| *Success Path:* | *The crew will isolate the affected EC in accordance with N1-ARP-K1 to minimize the release or radioactivity to the public.* |
| *Initiating Cue:* | *Multiple annunciators will provide indications of a tube leak in the Emergency Condenser. Rising water temperatures and rad levels on the Emergency Condenser Vents will provide indication on the affected Emergency Condenser. Field reports will provide indication of steam coming from the EC vent on the North side of the Reactor Building. N1-ARP-K1 provides direction to isolate the Emergency Condenser.* |
| *Measurable Performance Standard:* | *The crew will manually isolate the EC's will provide observable actions for the evaluation team.* ***Bounding Criteria:*** *Time based on Operations Representative recommendation to allow the crew time to evaluate the validity of the tube leak and to take the actions required in N1-ARP-K1-1-4.* |
| *Performance Feedback:* | *Emergency Condenser pressure lowering, lowering Rad Levels on the EC vents will provide performance feedback regarding the successful isolation of the Emergency Condenser.* |

| | |
|---|---|
| ***Critical Task (#2)*** | **Given a coolant leak inside the Containment, lowering Reactor water level, and failure of injection systems to automatically inject to the Reactor, the crew will inject to the Reactor to restore and maintain Reactor water level above -84", in accordance with N1-EOP-RPV. Reactor water level must not be below -84" for more than 10 minutes.** |
| *CT-2.0 Justification:* | |
| *Safety Significance* | *Maintaining Reactor water level above -84 inches ensures adequate core cooling through the preferred method of core submergence. This protects the integrity of the fuel cladding.* |
| *Success Path:* | *The crew will inject with all available low-pressure preferred and alternate injection sources (Details E1 and E2) in accordance with N1-EOP-RPV (1-2).* |
| *Initiating Cue:* | *Multiple Reactor water level indicators and annunciators will provide indications of lowering Reactor water level. N1-EOP-RPV (1-2) provides multiple procedure steps directing injection with preferred and alternate injection systems.* |
| *Measurable Performance Standard:* | *Manipulation of pumps and/or valves in the preferred or alternate injection system(s) will provide observable actions for the evaluation team. As well as operating ERVs for depressurization.* ***Bounding Criteria:*** *Time requirement based on Operations representative recommendation to allow RPV pressure to lower to within the capacity of the low-pressure injection sources.* |
| *Performance Feedback:* | *Multiple Reactor water level indicators and annunciators will provide performance feedback regarding the success of injection with preferred and alternate injection systems.* |

Training Id: **2024 NRC Admin RO COO1** Revision: **0.0**

Title: **N1-ST-D0 DAILY CHECKS (Partial)**

**Approvals:**

| | Signature / Printed Name | Date |
|---|---|---|
| Developed By | Patrick Ryan | 02/15/2024 |
| Validated By | Ken Graham | 03/29/24 |
| Facility Reviewer | Dave Mason | |

Approximate Duration: 10 minutes

**Documentation of Performance:**

Performer: _____

Evaluator: _____

Start Time: _____ Stop Time: _____ Completion Time _____

Grade: **Pass / Fail**

Comments: _____

Evaluators Signature: _____ Date: _____

## References

1. NUREG 1123, 2.1.7 (4.4)
2. N1-ST-DO, Daily Checks

Instructor Information

## A. JPM Information

1. Description

   a. This JPM tests the operator's ability to record and interpret plant surveillance data and identify inoperable equipment. Control room RPV water level readings are taken for N1-ST-DO. Out of spec readings and inoperable equipment are identified by the operator.

2. Task Information:

   a. NS-REL-01001, Perform System Surveillances.

   b. K/A 2.1.19 (3.9)

3. Evaluation / Task Criteria

| | |
|---|---|
| Evaluation Method | Perform |
| Evaluation Location | Simulator |
| Time Critical Task | No |
| Alternate Path | No |
| LOD >1.0 | Yes |

4. Recommended Start Location

   a. Simulator

5. JPM Setup (if required)

   a. Provide copies of N1-ST-DO
   b. Initialize simulator to IC-239
   c. Plant is operating at approximately 15% power
   d. Verify the following overrides are set
      1) **AO-523**, 1F3/36-09, FINAL = 73.00
      2) **AO-524**, 1F3/36-10, FINAL = 75.00
      3) **AO-552**, 1F3/ID59A, FINAL = 68.00 *(GEMAC Channel 11)*
      4) **AO-554**, 1F9/ID59B, FINAL = 76.00 *(GEMAC Channel 12)*
      5) **AO-1125**, 6K27A/1D59C, FINAL = 68.00 *(RPV Level Indication on K-panel)*

6) **AO-929**, 1E9/1D59D, FINAL = 68.00 *(RPV Level indication on E-console)*
7) **AO=568**, 1F14/1D14 PEN 2, FINAL = 68.00 *(RPV Level Trend Recorder on F=panel)*

e. Take the simulator to run, verify plant parameters, then return simulator to freeze
f. Verify average Drywell temperature is less than 130°F
g. Verify total recirculation flow is 38 Mlbm/hr
h. Verify FWLC is selected to column 11
i. Turn off PI monitor
j. Ensure a calculator is available.

## B. Read Before Every JPM Performance

1. For the performance of this JPM, I will act as all those you need to talk to. Prior to providing direction to perform this task, I will provide you with the initial conditions and answer any questions. During task performance, I will identify the steps to be simulated, or discuss and provide cues as necessary. (Note, read the next only if conducting a plant JPM). With the exception of accessing panels, no plant equipment will be physically manipulated. Repositioning of devices will be simulated by discussion and acknowledged by my cues.

## C. Read Before Each Evaluated JPM

1. This evaluated JPM is a measure of your ability to perform this task independently. The US has determined that a verifier is not available and that additional verification will not be provided.

| | |
|---|---|
| **INITIAL CONDITIONS** | Given:<br>• The plant is operating at approximately 15% power<br>• N1-ST-DO Daily Checks are in progress<br>• Another operator will perform independent verifications once section 6.4.2 is complete<br>• The plant process computer is NOT available<br><br>**Evaluator:** *Ask trainee if he/she has any questions after presenting initial conditions* |

| | |
|---|---|
| **INITIATING CUE** | **(Operators Name)**, complete N1-ST-DO steps 6.4.2.1 through 6.4.2.10. When completed, report findings and provide completed sections to SRO. |

| | |
|---|---|
| **START TIME** | |

| | **PERFORMANCE** | **ACT. CODE**<br>P / S / NA | **EVALUATOR** |
|---|---|---|---|
| 1. | Provide repeat back of initiating cue<br><br>**Cue:** *Acknowledge repeat back providing correction if necessary.* | P | SAT / UNSAT<br><br>**STD:** Proper communications used. |
| 2. | Obtain a copy of the reference procedure and review / utilize the correct section of the procedure.<br><br>**Note:** Completed N1-ST-DO sections are attached as JPM key. Small differences in recorded values may occur due to instrument interpolation. | P | SAT / UNSAT<br><br>**STD:** N1-ST-DO sections 6.4.2 obtained. Remainder of procedure referenced if necessary. |

| | PERFORMANCE | ACT. CODE<br>P / S / NA | EVALUATOR |
|---|---|---|---|
| 3. | Record Reactor Water Level Low-Low alarm status<br><br>**Note:** Step 6.4.2.2 is N/A due to no alarms present. | P | SAT / UNSAT<br><br>**STD:** Determines no reactor water level low-low alarms are present. Marks alarms clear for step 6.4.2.1 and N/A for step 6.4.2.2. |
| 4. | Records Feedwater Level Column and Reactor Pressure Column selection<br><br>**Note:** Step 6.4.2.5 is N/A due to no level or pressure column locked out. | P | SAT / UNSAT<br><br>**STD:** Determines Channel 11 selected for both Feedwater Level and Reactor Pressure. Marks Channel 11 on steps 6.4.2.3 and 6.4.2.4. Determines no columns are locked out and marks 6.4.2.5 N/A. |
| 5. | Records vessel level indications and total recirculation flow<br><br>**Note:** Small differences in recorded values may occur due to instrument interpolation. | P | SAT / UNSAT<br><br>**STD:** Determines multiple reactor water level indications and total recirculation flow in accordance with step 6.4.2.6. |
| 6. | Calculates Yarway indicators LI-36-09 and LI-36-10 variance<br><br>**Note:** Small differences in recorded values may occur due to instrument interpolation. | P | SAT / UNSAT<br><br>**STD:** Determines Yarway indication variance in accordance with step 6.4.2.6. |

| | PERFORMANCE | ACT. CODE<br>P / S / NA | EVALUATOR |
|---|---|---|---|
| 7. | Verifies Yarway indicators LI-36-09 and LI-36-10 RPV level indication is less than 10 inches<br><br>**Note:** Small differences in recorded values may occur due to instrument interpolation. However, grossly inaccurate interpolation would be required to determine the Yarway indicators inoperable. | P | ***PASS / FAIL**<br><br>**STD:** Determines Yarway indication variance is less than 10 inches in accordance with step 6.4.2.8. |
| 8. | Calculates GEMAC Channel 11 and Channel 12 level meter variance<br><br>**Note:** Small differences in recorded values may occur due to instrument interpolation. | P | SAT / UNSAT<br><br>**STD:** Determines GEMAC indication variance in accordance with step 6.4.2.9. |
| 9. | Determines GEMAC 11/12 variance as a function of total recirculation flow<br><br>**Note:** Small differences in recorded values may occur due to instrument interpolation. However, grossly inaccurate interpolation would be required to determine the GEMAC indicators inoperable. | P | ***PASS / FAIL**<br><br>**STD:** Determines GEMAC indication variance is greater than 6 inches at the given total recirculation flow. Selects INOP Range for step 6.4.2.10 |
| 9. | Provides completed sections to SRO. Notifies SRO GEMAC Channel 11/12 variance readings are out of specification. | P | SAT / UNSAT<br><br>**STD:** Proper communications used. |

| | |
|---|---|
| **TASK STANDARD** | Control room readings are taken. Out of spec readings are identified to SRO and inoperable equipment is identified by the candidate. |

| | |
|---|---|
| **STOP TIME** | |

# JPM Handout

| | |
|---|---|
| **INITIAL CONDITIONS** | Given:<br>• The plant is operating at approximately 15% power.<br>• N1-ST-DO, Daily Checks, is in progress.<br>• Another operator will perform independent verifications once section 6.4.2 is complete.<br>• The plant process computer is NOT available. |

| | |
|---|---|
| **INITIATING CUE** | **(Operators Name)**, complete N1-ST-DO steps 6.4.2.1 through 6.4.2.10. When completed, report findings and provide completed sections to SRO. |

6.4. **IF** an approved SOMS printout has **not** been attached,
**THEN RECORD** the following Plant Data,
**OTHERWISE MARK** these steps N/A:

6.4.1. **INDICATE** Mode Switch Position:

- [X] RUN
- [ ] STARTUP
- [ ] SHUTDOWN
- [ ] REFUEL

PMR

6.4.2. Control Room Reactor Water Level Checks: **[T/S]**

1. **RECORD** Reactor Water Level Low-Low Alarm status:

   - [ ] Alarm(s) In
   - [X] Alarm(s) Clear

   PMR

2. **IF** Low-Low Level Alarms are present,
   **THEN RECORD** reason in Remarks,
   **OTHERWISE MARK** this step N/A.

   N/A

3. **RECORD** Feedwater Level Column selection:

   - [X] 11
   - [ ] 12

   PMR

4. **RECORD** Reactor Pressure Column selection:

   - [X] 11
   - [ ] 12

   PMR

5. **IF** any Level Column
   **OR** Pressure Column is locked out,
   **THEN RECORD** reason in Remarks,
   **OTHERWISE MARK** this step N/A.

   N/A

6.4.2 (Continued)

6. **RECORD** the following levels:

- E Panel Vessel Level Indicator:

  68 inches
- K Panel Vessel Level Indicator:

  68 inches
- K Panel Flange Level Indicator:

  - 3 feet
- Reactor Vessel Level Recorder:

  68 inches
- Wide Range Level Indicator:

  5 feet
- Ch 11 GEMAC Level Column ID59A:

  68
- Ch 12 GEMAC Level Column ID59B:

  76
- Total Recirc Flow: 38 Mlb/Hr
- Yarway RPS Level Column LI-36-09:

  74 inches
- Yarway RPS Level Column LI-36-10:

  76 inches

PMR

6.4.2 (Continued)

**NOTE**

Yarway RPS Level Column references: 3.6.2a(4), 4.6.2a(4), 3.6.2k(1), 4.6.2k(1), Operability Evaluation OpEval# ENGDOC-21-000006 from IR 04415668.

7. **CALCULATE** Yarway indicators LI-36-09 **AND** LI-36-10 variance:

74 inches (LI-36-09) – 76 inches (LI-36-10)

= - 2 inches variance

PMR

IV

8. **IF** Yarway indicators LI-36-09 and LI-36-10 RPV level indication deviation is greater than 10 inches, **THEN PERFORM** the following, **OTHERWISE MARK** these steps N/A:

- **REVIEW** previous readings for the instrument(s). ..... ☐
- **IF** a single reading is the cause for the variance value being greater than 10 inches, **THEN VERIFY** that reading, **AND IF** variance is still greater than 10 inches, **THEN CONSIDER** the indicator OPERABLE **AND GENERATE** an IR to have the calibration checked, **OTHERWISE MARK** this step N/A. ..... ☐
- **IF** variance is grossly greater than 10 inches, **THEN CONSIDER** the indicator INOPERABLE, **OTHERWISE MARK** this step N/A. ..... ☐
- **INITIATE** the required Technical Specification actions (AR 02031163). ..... ☐
- **SEE** 2.4.9 TS Guidance for Level Indication Deviation for additional guidance. ..... ☐

N/A

6.4.2 (Continued)

9. **CALCULATE** Ch 12
**AND** Ch 11 GEMAC Level Meter variance:

76 inches (Ch 12) – 68 inches (Ch 11)

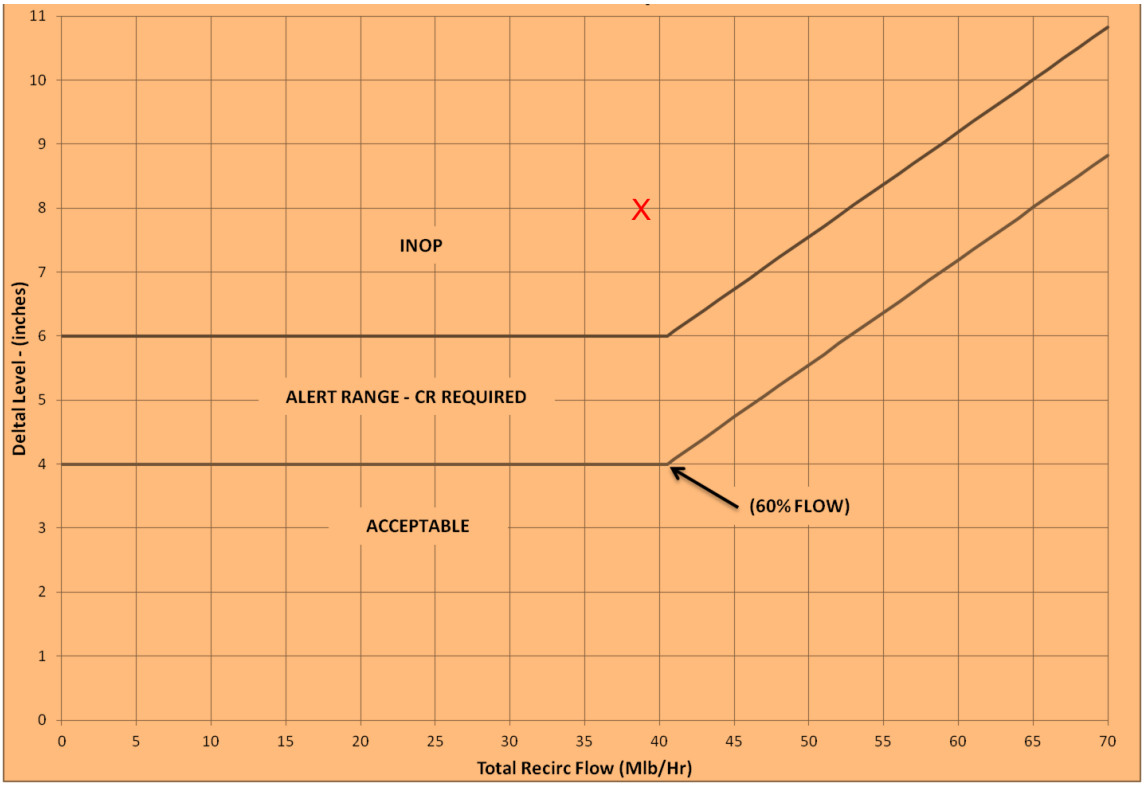
= 8 inches variance PMR

IV

**NOTE**

Ignore negative sign, if any.

10. **DETERMINE** GEMAC 11/12 Variance as a function of Total Recirc Flow:




**AND CHECK** applicable result:

☐ **ACCEPTABLE** Range, **no** action required.

☒ **INOP** Range,
**DECLARE** GEMAC 11 **AND/OR** 12 INOP

☐ **ALERT** Range,
**GENERATE** IR for Engineering Evaluation. PMR

6.4.2 (Continued)

11. **IF** GEMAC 11/12 Variance as a function of Total Recirc Flow is **not** acceptable,
**THEN PERFORM** the following,
**OTHERWISE MARK** these steps N/A:

- **REVIEW** previous readings for the instrument(s). ..... ☐
- **IF** a single reading is the cause for the variance value being greater than required,
**THEN VERIFY** that reading,
**AND IF** variance is still greater required,
**THEN CONSIDER** the indicator OPERABLE
**AND GENERATE** an IR to have the calibration checked,
**OTHERWISE MARK** this step N/A. ..... ☐
- **IF** variance is grossly greater than required,
**THEN CONSIDER** the indicator INOPERABLE,
**OTHERWISE MARK** this step N/A. ..... ☐
- **INITIATE** the required Technical Specification actions (AR 02031163). ..... ☐
- **SEE** 2.4.9 TS Guidance for Level Indication Deviation for additional guidance..... ☐ _____

Training Id: **2024 NRC Admin RO COO2** Revision: **0.0**

Title: **Perform Heat-up Rate Determination**

**Approvals:**

| | Signature / Printed Name | Date |
|---|---|---|
| Developed By | Patrick Ryan | 2/15/2024 |
| Validated By | Tim Heagerty | 3/29/2024 |
| Facility Reviewer | Dave Mason | |

Approximate Duration: 15 minutes

**Documentation of Performance:**

Performer: _____

Evaluator: _____

Start Time: _____ Stop Time: _____ Completion Time _____

Grade: **Pass / Fail**

Comments: _____

Evaluators Signature: _____ Date: _____

## References

1. N1-OP-43A, Plant Startup

Instructor Information

## A. JPM Information

1. Description

   a. This JPM tests the operator's ability to record and calculate heat-up rate data. An out of spec heat-up rate will be identified.

2. Task Information:

   a. K/A 2.1.7 (4.4)

3. Evaluation / Task Criteria

| | |
|---|---|
| Evaluation Method | Perform |
| Evaluation Location | Classroom |
| Time Critical Task | No |
| Alternate Path | No |
| LOD >1.0 | Yes |

4. Recommended Start Location

   a. Classroom

5. JPM Setup (if required)

   1. Prepare a picture of RRP 11 suction temperature gauge indicating approximately 529°F.
   2. Provide each candidate a calculator.
   3. Prepare N1-OP-43A Attachment 17 with the following data (next page):

| Date/Time | RRP Suction Temp | ΔT/15 min | ΔT/15 min Limits | Heatup Rate Limits |
|---|---|---|---|---|
| Today 0900 | 350 | N/A | N/A | N/A |
| Today 0915 | 370 | 20 | SAT | N/A |
| Today 0930 | 380 | 10 | SAT | N/A |
| Today 0945 | 400 | 20 | SAT | N/A |
| Today 1000 | 415 | 15 | SAT | SAT |
| Today 1015 | 435 | 20 | SAT | SAT |
| Today 1030 | 470 | | | |
| Today 1045 | 490 | | | |
| Today 1100 | | | | |

## B. Read Before Every JPM Performance

1. For the performance of this JPM, I will act as all those you need to talk to. Prior to providing direction to perform this task, I will provide you with the initial conditions and answer any questions. During task performance, I will identify the steps to be simulated, or discuss and provide cues as necessary. (Note, read the next only if conducting a plant JPM). With the exception of accessing panels, no plant equipment will be physically manipulated. Repositioning of devices will be simulated by discussion and acknowledged by my cues.

## C. Read Before Each Evaluated JPM

1. This evaluated JPM is a measure of your ability to perform this task independently. The US has determined that a verifier is not available and that additional verification will not be provided.

| | |
|---|---|
| **INITIAL CONDITIONS** | Given:<br>• A plant startup is in progress.<br>• Current time is 1100 Today.<br>• N1-OP-43A Attachment 17 is provided with heat-up data for the last hour.<br>• All RRP suction temperature readings have been taken from Recirc loop 11.<br>• The picture provided below shows the current Recirc loop 11 suction temperature indication<br>**Evaluator:** *Ask trainee if he/she has any questions after presenting initial conditions* |

| | |
|---|---|
| **INITIATING CUE** | **(Operators Name)**, obtain the time 1100 reading for N1-OP-43A Attachment 17, then complete the attachment for times 1030, 1045, and 1100. When completed, report findings and provide completed sections. |

| | |
|---|---|
| **START TIME** | |

| | **PERFORMANCE** | **ACT. CODE**<br>P / S / NA | **EVALUATOR** |
|---|---|---|---|
| 1. | Provide repeat back of initiating cue<br>**Cue:** *Acknowledge repeat back providing correction if necessary.* | P | SAT / UNSAT<br>**STD:** Proper communications used. |
| 2. | Obtain a copy of the reference procedure and review / utilize the correct section of the procedure.<br>**Note:** A completed N1-OP-43A Attachment 17 is provided as a JPM key. Small differences in recorded values may occur due to instrument interpolation. | P | SAT / UNSAT<br>**STD:** N1-ST-43A attachment 17 obtained, rest of procedure referenced if necessary. |
| 3. | Record current RRP suction temperature | P | *** PASS / FAIL**<br>**STD:** Records current RRP suction temperature (517°F) |

| | PERFORMANCE | ACT. CODE<br>P / S / NA | EVALUATOR |
|---|---|---|---|
| 4. | Calculate and record ΔT/15 min for times 1030-1100 | P | SAT / UNSAT<br><br>**STD:** Calculates and records ΔT/15 min:<br><br>1030: 470°F - 435°F = 35°F<br><br>1045: 490°F - 470°F = 20°F<br><br>1100: 517°F - 490°F = 27°F |
| 5. | Analyze and record ΔT Limits for ΔT/15 min | P | SAT / UNSAT<br><br>**STD:** Analyzes and records ΔT Limits for ΔT/15 min:<br><br>1030: UNSAT<br><br>1045: SAT<br><br>1100: UNSAT |
| 6. | Calculate hourly ΔTs for times 1030-1100<br><br>**Note:** There is no column on N1-OP-43A Attachment 17 to record this step. It is provided to document an intermediate step taken to arrive at JPM Step #7. If the candidate does not provide these intermediate values, the JPM step may be graded "NA". | P | SAT / UNSAT / NA<br><br>**STD:** Calculates ΔTs:<br><br>1030: 470°F - 380°F = 90°F<br><br>1045: 490°F - 400°F = 90°F<br><br>1100: 517°F - 415°F = 102°F |

| | PERFORMANCE | ACT. CODE<br>P / S / NA | EVALUATOR |
|---|---|---|---|
| 7. | Analyze and record Heat-up Rate Limits for 100°F/hr limits. | P | *** PASS / FAIL**<br><br>**STD:** Determines SAT for 1030 |
| | | P | *** PASS / FAIL**<br><br>**STD:** Determines SAT for 1045 |
| | | P | *** PASS / FAIL**<br><br>**STD:** Determines UNSAT for 1100 |
| 8. | Provides completed attachment to examiner and informs of issues. | P | SAT / UNSAT<br><br>**STD:** Proper communications used. |

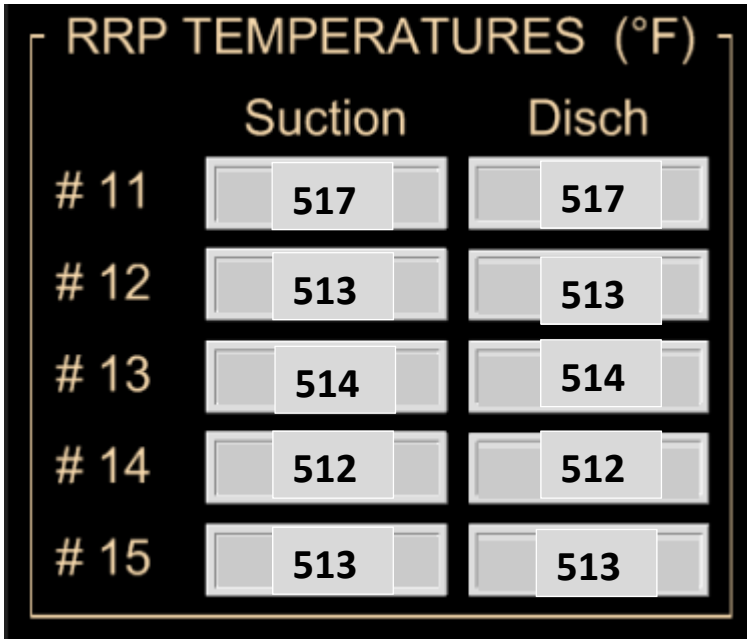
| | |
|---|---|
| **TASK STANDARD** | Current plant heat-up data taken per N1-OP-43A Attachment 17. Current and past heat-up rates calculated. Candidate correctly determines the current RRP 11 suction temperature and calculates heat up rates in accordance with N1-OP-43A. Candidate determines 100°F/hr limit is satisfactory for 1030 and 1045, and is unsatisfactory at 1100. |

| | |
|---|---|
| **STOP TIME** | |

# JPM Handout

| | |
|---|---|
| **INITIAL CONDITIONS** | Given:<br>• A plant startup is in progress.<br>• Current time is 1100 Today.<br>• N1-OP-43A Attachment 17 is provided with heat-up data for the last hour.<br>• All RRP suction temperature readings have been taken from Recirc loop 11.<br>• The picture provided below shows the current Recirc loop 11 suction temperature indication. |

| | |
|---|---|
| **INITIATING CUE** | **(Operators Name)**, obtain the time 1100 reading for N1-OP-43A Attachment 17, then complete the attachment for times 1030, 1045, and 1100. When completed, report findings and provide completed sections. |



Attachment 17, Heatup Rate Monitoring

| Date/Time (T+15 min) | RRP Suction Temp (°F) | ΔT/15 min[I] | ΔT Limits ΔT/15 min ≤25 °F | Heatup Rate Limits[H] (PTLR) ≤100 °F/hr | Init. | Init (SRO) |
|---|---|---|---|---|---|---|
| Today 0900 | 350 | N/A | N/A | N/A | RO | SRO |
| Today 0915 | 370 | 20 | SAT | N/A | RO | SRO |
| Today 0930 | 380 | 10 | SAT | N/A | RO | SRO |
| Today 0945 | 400 | 20 | SAT | N/A | RO | SRO |
| Today 1000 | 415 | 15 | SAT | | RO | SRO |
| Today 1015 | 435 | 20 | SAT | | RO | SRO |
| Today 1030 | 470 | | | | RO | SRO |
| Today 1045 | 490 | | | | | |
| Today 1100 | | | | | | |
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[H] Heatup Rate Limitations are performed on an hour ΔT.

[I] ΔT/15 min is calculated by subtracting the previous temperature from the current temperature.

Attachment 17, Heatup Rate Monitoring

| Date/Time (T+15 min) | RRP Suction Temp (°F) | ΔT/15 min[I] | ΔT Limits ΔT/15 min ≤25 °F | Heatup Rate Limits[H] (PTLR) ≤100 °F/hr | Init. | Init (SRO) |
|---|---|---|---|---|---|---|
| Today 0900 | 350 | N/A | N/A | N/A | RO | SRO |
| Today 0915 | 370 | 20 | SAT | SAT | RO | SRO |
| Today 0930 | 380 | 10 | SAT | SAT | RO | SRO |
| Today 0945 | 400 | 20 | SAT | SAT | RO | SRO |
| Today 1000 | 415 | 15 | SAT | SAT | RO | SRO |
| Today 1015 | 435 | 20 | SAT | SAT | RO | SRO |
| Today 1030 | 470 | 35 | UNSAT | SAT | RO | SRO |
| Today 1045 | 490 | 20 | SAT | SAT | | |
| Today 1100 | 517 | 27 | UNSAT | UNSAT | | |
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[H] Heatup Rate Limitations are performed on an hour ΔT.

[I] ΔT/15 min is calculated by subtracting the previous temperature from the current temperature.

Training Id: **2024 NRC Admin RO COO2** Revision: **0.0**

Title: **Perform Heat-up Rate Determination**

**Approvals:**

| | Signature / Printed Name | Date |
|---|---|---|
| Developed By | Patrick Ryan | 2/15/2024 |
| Validated By | Tim Heagerty | 3/29/2024 |
| Facility Reviewer | Dave Mason | |

Approximate Duration: 15 minutes

**Documentation of Performance:**

Performer: _____

Evaluator: _____

Start Time: _____ Stop Time: _____ Completion Time _____

Grade: **Pass / Fail**

Comments: _____

Evaluators Signature: _____ Date: _____

## References

1. N1-OP-43A, Plant Startup

Instructor Information

## A. JPM Information

1. Description

   a. This JPM tests the operator's ability to record and calculate heat-up rate data. An out of spec heat-up rate will be identified.

2. Task Information:

   a. K/A 2.1.7 (4.4)

3. Evaluation / Task Criteria

| | |
|---|---|
| Evaluation Method | Perform |
| Evaluation Location | Classroom |
| Time Critical Task | No |
| Alternate Path | No |
| LOD >1.0 | Yes |

4. Recommended Start Location

   a. Classroom

5. JPM Setup (if required)

   1. Prepare a picture of RRP 11 suction temperature gauge indicating approximately 529°F.
   2. Provide each candidate a calculator.
   3. Prepare N1-OP-43A Attachment 17 with the following data (next page):

| Date/Time | RRP Suction Temp | ΔT/15 min | ΔT/15 min Limits | Heatup Rate Limits |
|---|---|---|---|---|
| Today 0900 | 350 | N/A | N/A | N/A |
| Today 0915 | 370 | 20 | SAT | N/A |
| Today 0930 | 380 | 10 | SAT | N/A |
| Today 0945 | 400 | 20 | SAT | N/A |
| Today 1000 | 415 | 15 | SAT | SAT |
| Today 1015 | 435 | 20 | SAT | SAT |
| Today 1030 | 470 | | | |
| Today 1045 | 490 | | | |
| Today 1100 | | | | |

## B. Read Before Every JPM Performance

1. For the performance of this JPM, I will act as all those you need to talk to. Prior to providing direction to perform this task, I will provide you with the initial conditions and answer any questions. During task performance, I will identify the steps to be simulated, or discuss and provide cues as necessary. (Note, read the next only if conducting a plant JPM). With the exception of accessing panels, no plant equipment will be physically manipulated. Repositioning of devices will be simulated by discussion and acknowledged by my cues.

## C. Read Before Each Evaluated JPM

1. This evaluated JPM is a measure of your ability to perform this task independently. The US has determined that a verifier is not available and that additional verification will not be provided.

| | |
|---|---|
| **INITIAL CONDITIONS** | Given:<br>• A plant startup is in progress.<br>• Current time is 1100 Today.<br>• N1-OP-43A Attachment 17 is provided with heat-up data for the last hour.<br>• All RRP suction temperature readings have been taken from Recirc loop 11.<br>• The picture provided below shows the current Recirc loop 11 suction temperature indication<br>**Evaluator:** *Ask trainee if he/she has any questions after presenting initial conditions* |

| | |
|---|---|
| **INITIATING CUE** | **(Operators Name)**, obtain the time 1100 reading for N1-OP-43A Attachment 17, then complete the attachment for times 1030, 1045, and 1100. When completed, report findings and provide completed sections. |

| | |
|---|---|
| **START TIME** | |

| | **PERFORMANCE** | **ACT. CODE**<br>P / S / NA | **EVALUATOR** |
|---|---|---|---|
| 1. | Provide repeat back of initiating cue<br>**Cue:** *Acknowledge repeat back providing correction if necessary.* | P | SAT / UNSAT<br>**STD:** Proper communications used. |
| 2. | Obtain a copy of the reference procedure and review / utilize the correct section of the procedure.<br>**Note:** A completed N1-OP-43A Attachment 17 is provided as a JPM key. Small differences in recorded values may occur due to instrument interpolation. | P | SAT / UNSAT<br>**STD:** N1-ST-43A attachment 17 obtained, rest of procedure referenced if necessary. |
| 3. | Record current RRP suction temperature | P | *** PASS / FAIL**<br>**STD:** Records current RRP suction temperature (517°F) |

| | PERFORMANCE | ACT. CODE<br>P / S / NA | EVALUATOR |
|---|---|---|---|
| 4. | Calculate and record ΔT/15 min for times 1030-1100 | P | SAT / UNSAT<br><br>**STD:** Calculates and records ΔT/15 min:<br><br>1030: 470°F - 435°F = 35°F<br><br>1045: 490°F - 470°F = 20°F<br><br>1100: 517°F - 490°F = 27°F |
| 5. | Analyze and record ΔT Limits for ΔT/15 min | P | SAT / UNSAT<br><br>**STD:** Analyzes and records ΔT Limits for ΔT/15 min:<br><br>1030: UNSAT<br><br>1045: SAT<br><br>1100: UNSAT |
| 6. | Calculate hourly ΔTs for times 1030-1100<br><br>**Note:** There is no column on N1-OP-43A Attachment 17 to record this step. It is provided to document an intermediate step taken to arrive at JPM Step #7. If the candidate does not provide these intermediate values, the JPM step may be graded "NA". | P | SAT / UNSAT / NA<br><br>**STD:** Calculates ΔTs:<br><br>1030: 470°F - 380°F = 90°F<br><br>1045: 490°F - 400°F = 90°F<br><br>1100: 517°F - 415°F = 102°F |

| | PERFORMANCE | ACT. CODE<br>P / S / NA | EVALUATOR |
|---|---|---|---|
| 7. | Analyze and record Heat-up Rate Limits for 100°F/hr limits. | P | *** PASS / FAIL**<br><br>**STD:** Determines SAT for 1030 |
| | | P | *** PASS / FAIL**<br><br>**STD:** Determines SAT for 1045 |
| | | P | *** PASS / FAIL**<br><br>**STD:** Determines UNSAT for 1100 |
| 8. | Provides completed attachment to examiner and informs of issues. | P | SAT / UNSAT<br><br>**STD:** Proper communications used. |

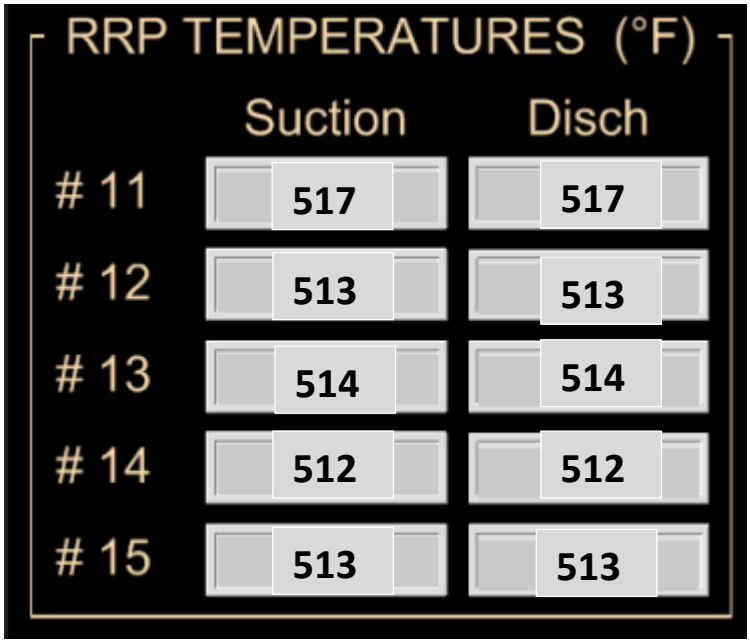
| | |
|---|---|
| **TASK STANDARD** | Current plant heat-up data taken per N1-OP-43A Attachment 17. Current and past heat-up rates calculated. Candidate correctly determines the current RRP 11 suction temperature and calculates heat up rates in accordance with N1-OP-43A. Candidate determines 100°F/hr limit is satisfactory for 1030 and 1045, and is unsatisfactory at 1100. |

| | |
|---|---|
| **STOP TIME** | |

# JPM Handout

| | |
|---|---|
| **INITIAL CONDITIONS** | Given:<br>• A plant startup is in progress.<br>• Current time is 1100 Today.<br>• N1-OP-43A Attachment 17 is provided with heat-up data for the last hour.<br>• All RRP suction temperature readings have been taken from Recirc loop 11.<br>• The picture provided below shows the current Recirc loop 11 suction temperature indication. |

| | |
|---|---|
| **INITIATING CUE** | **(Operators Name)**, obtain the time 1100 reading for N1-OP-43A Attachment 17, then complete the attachment for times 1030, 1045, and 1100. When completed, report findings and provide completed sections. |



Training Id: **2024 NRC Admin RO EC** Revision: **0.0**

Title: **Develop Clearance – RBCLC Pump 13**

**Approvals:**

| | Signature / Printed Name | Date |
|---|---|---|
| Developed By | Patrick Ryan | 1/24/2024 |
| Validated By | Ken Graham | 3/29/2024 |
| Facility Reviewer | | |

Approximate Duration: 25 minutes

**Documentation of Performance:**

Performer: _____

Evaluator: _____

Start Time: _____ Stop Time: _____ Completion Time _____

Grade: **Pass / Fail**

Comments: _____

Evaluators Signature: _____ Date: _____

## References

1. EWDs C-19436-C Sheets 1 and 3
2. P&ID C-18022-C Sheet 2
3. OP-AA-109-101-F-07 Worker Tagout (WTO) Form
4. OP-AA-109-101 Personnel and Equipment Tagout Process
5. N1-OP-11 Reactor Building Closed Loop Cooling System
6. NUREG 1123 2.2.15 (3.9)

Instructor Information

## A. JPM Information

1. Description

   a. This JPM tests the operator's ability to develop a tagout in accordance with OP-AA-109-101 Personnel and Equipment Tagout Process.

2. Task Information:

   a. K/A 2.2.15 (3.9)

3. Evaluation / Task Criteria

| | |
|---|---|
| Evaluation Method | Perform |
| Evaluation Location | Classroom |
| Time Critical Task | No |
| Alternate Path | No |
| LOD >1.0 | Yes |

4. Recommended Start Location

   a. Classroom

5. JPM Setup (if required)

   a. Verify copies of the following reference material provided to the candidate:

      1) EWD C-19436-C Sheets 1 and 3

      2) P&ID C-18022-C Sheet 2

      3) OP-AA-109-101 Personnel and Equipment Tagout Process

      4) OP-AA-109-101-F-07 Worker Tagout (WTO) Form

## B. Read Before Every JPM Performance

1. For the performance of this JPM, I will act as all those you need to talk to. Prior to providing direction to perform this task, I will provide you with the initial conditions and answer any questions. During task performance, I will identify the steps to be simulated, or discuss and provide cues as necessary. (Note, read the next only if conducting a plant JPM). With the exception of accessing panels, no plant equipment will be physically manipulated. Repositioning of devices will be simulated by discussion and acknowledged by my cues.

## C. Read Before Each Evaluated JPM

1. This evaluated JPM is a measure of your ability to perform this task independently. The US has determined that a verifier is not available and that additional verification will not be provided.

| | |
|---|---|
| **INITIAL CONDITIONS** | Given:<br>• The plant is operating at 100% power<br>• RBCLC Pumps 11 and 12 are operating<br>• RBCLC Pump 13 is shutdown<br>• Maintenance is required to replace RBCLC Pump 13 shaft seal under work order C91234567<br>• The pressure switch is not required to be tagged out for this maintenance<br>• eSOMS is unavailable<br>• Operations has determined that the WTO process is appropriate for this task<br><br>**Evaluator:** *Ask trainee if he/she has any questions after presenting initial conditions* |

| | |
|---|---|
| **INITIATING CUE** | **(Operators Name)**, identify the components required to be tagged for RBCLC Pump 13 shaft seal replacement. Record the required components and their tagged positions starting on page 2 of OP-AA-109-101-F-07 Worker Tagout (WTO) Form. |

| | |
|---|---|
| **START TIME** | |

| | **PERFORMANCE** | **ACT. CODE**<br>P / S / NA | **EVALUATOR** |
|---|---|---|---|
| 1. | Provide repeat back of initiating cue<br><br>**Cue:** *Acknowledge repeat back providing correction if necessary.* | P | SAT / UNSAT<br><br>**STD:** Proper communications used. |
| 2. | Obtain a copy of the reference procedure and review / utilize the correct section of the procedure.<br><br>**Note:** OP-AA-109-101 provides guidance on tagging standards and isolations. If eSOMS is unavailable, then the Worker Tagout (WTO) process shall be used per OP-AA-109-101-F07. | P | SAT / UNSAT<br><br>**STD:** OP-AA-109-101 obtained and referenced as necessary. |
| 3. | Obtains and references appropriate drawings and/or procedures to determine appropriate isolations. | P | SAT / UNSAT<br>**STD:** P&ID C-18022-C sheet 2 and EWD C-19436-C sheets 1 and 3 referenced as necessary. |

| | PERFORMANCE | ACT. CODE<br>P / S / NA | EVALUATOR |
|---|---|---|---|
| 4. | Determines 70-03 Reactor Building Closed Loop Cooling Pump 13 Control Switch INFO/CAUTION tagged in PULL-TO-LOCK position<br><br>**Note:** Due to unavailability of eSOMS and other electronic databases, exact component IDs, names, and tagging positions may vary. | P | SAT / UNSAT<br>**STD:** In accordance with OP-AA-109-101. |
| 5. | Determines 70-03 Reactor Building Closed Loop Cooling Pump 13 motor breaker TRIP fuses tagged in PULLED or INSTALLED IN OFF position.<br><br>**Note:** Due to unavailability of eSOMS and other electronic databases, exact component IDs, names, and tagging positions may vary. Candidate may elect any type of tag (if any) for the trip fuses since they do not provide any protection for the work to be performed. If the operator does not place a tag on this component, mark this step as N/A. | P | SAT / UNSAT / N/A<br>**STD:** In accordance with OP-AA-109-101. |
| 6. | Determines 70-03 Reactor Building Closed Loop Cooling Pump 13 motor breaker CLOSE fuses tagged in PULLED or INSTALLED IN OFF position.<br><br>**Note:** Due to unavailability of eSOMS and other electronic databases, exact component IDs, names, and tagging positions may vary. Candidate may elect any type of tag (if any) for the close fuses since they do not provide any protection for the work to be performed. If the operator does not place a tag on this component, mark this step as N/A. | P | SAT / UNSAT / N/A<br>**STD:** In accordance with OP-AA-109-101. |

| | PERFORMANCE | ACT. CODE<br>P / S / NA | EVALUATOR |
|---|---|---|---|
| 7. | Determines 70-03 Reactor Building Closed Loop Cooling Pump 13 motor breaker DANGER tagged in RACKED OUT or DISCONNECTED position.<br><br>**Note:** Due to unavailability of eSOMS and other electronic databases, exact component IDs, names, and tagging positions may vary. | p | ***PASS / FAIL**<br>**STD:** In accordance with OP-AA-109-101. |
| 8. | Determines 70-09, BV – 13 RBCLC Pump Discharge DANGER tagged in CLOSED position.<br><br>**Note:** Due to unavailability of eSOMS and other electronic databases, exact component IDs, names, and tagging positions may vary. | P | ***PASS / FAIL**<br>**STD:** In accordance with OP-AA-109-101. |
| 9. | Determines 70-141, BV – 13 RBCLC Pump Suction DANGER tagged in CLOSED position.<br><br>**Note:** Due to unavailability of eSOMS and other electronic databases, exact component IDs, names, and tagging positions may vary. | P | ***PASS / FAIL**<br>**STD:** In accordance with OP-AA-109-101. |
| 10. | Determines 70-500, Drain – 13 RBCLC Pump tagged in OPEN position.<br><br>**Note:** Due to unavailability of eSOMS and other electronic databases, exact component IDs, names, and tagging positions may vary. Candidate may elect any type of tag (if any) for the drain valve since it does not provide any protection for the work to be performed. | P | SAT / UNSAT / N/A<br>**STD:** In accordance with OP-AA-109-101. |

| | PERFORMANCE | ACT. CODE<br>P / S / NA | EVALUATOR |
|---|---|---|---|
| 11. | Determines 70-501, Vent – 13 RBCLC Pump tagged in OPEN position and UNCAPPED.<br><br>**Note:** Due to unavailability of eSOMS and other electronic databases, exact component IDs, names, and tagging positions may vary. Candidate may elect any type of tag (if any) for the vent valve since it does not provide any protection for the work to be performed. | P | SAT / UNSAT / N/A<br>**STD:** In accordance with OP-AA-109-101. |
| 12. | Provides completed Worker Tagout (WTO) Form to examiner | P | SAT / UNSAT<br><br>**STD:** Proper communications used. |

| | |
|---|---|
| **TASK STANDARD** | The following isolations for Reactor Building Closed Loop Cooling Pump 13 identified and recorded on OP-AA-109-101-F-07:<br>70-03, RBCLC Pump Breaker danger tagged RACKED OUT or DISCONNECTED<br>70-09, BV – 13 RBCLC Pump Discharge danger tagged CLOSED<br>70-141, BV – 13 RBCLC Pump Suction danger tagged CLOSED |

| | |
|---|---|
| **STOP TIME** | |

# JPM Handout

| | |
|---|---|
| **INITIAL CONDITIONS** | Given:<br>• The plant is operating at 100% power<br>• RBCLC Pumps 11 and 12 are operating<br>• RBCLC Pump 13 is shutdown<br>• Maintenance is required to replace RBCLC Pump 13 shaft seal under work order C91234567<br>• The pressure switch is not required to be tagged out for this maintenance<br>• eSOMS is unavailable<br>• Operations has determined that the WTO process is appropriate for this task |

| | |
|---|---|
| **INITIATING CUE** | **(Operators Name)**, identify the components required to be tagged for RBCLC Pump 13 shaft seal replacement. Record the required components and their tagged positions starting on page 2 of OP-AA-109-101-F-07 Worker Tagout (WTO) Form. |

Training Id: **2024 NRC Admin RO RC** Revision: **0.0**

Title: **Application of Radiation Exposure Limits – Drywell**

**Approvals:**

| | Signature / Printed Name | Date |
|---|---|---|
| Developed By | Patrick Ryan | 2/13/2024 |
| Validated By | Ken Graham | 03/29/24 |
| Facility Reviewer | Dave Mason | |

Approximate Duration: 20 minutes

**Documentation of Performance:**

Performer: _____

Evaluator: _____

Start Time: _____ Stop Time: _____ Completion Time _____

Grade: **Pass / Fail**

Comments: _____

Evaluators Signature: _____ Date: _____

## References

1. RP-AA-203
2. NISP-RP-002

Instructor Information

## A. JPM Information

1. Description

   a. This JPM tests the operator's ability to evaluate radiological survey data against exposure limits in order to determine availability to perform operator tasks.

2. Task Information:

   a. S-RAP-RPP0703-00002, Authorize extension of emergency exposure limits

   b. K/A 2.3.12 (3.2/3.7), Knowledge of radiation exposure limits under normal or emergency conditions.

3. Evaluation / Task Criteria

| | |
|---|---|
| Evaluation Method | Perform |
| Evaluation Location | Classroom |
| Time Critical Task | No |
| Alternate Path | No |
| LOD >1.0 | Yes |

4. Recommended Start Location

   a. Classroom

5. JPM Setup (if required)

   a. Provide a copy of the Drywell survey map for each operator performing the JPM.
   b. Provide a copy of RP-AA-203 and NISP-RP-002 to each candidate.
   c. Ensure calculators are available for the applicant's use.

## B. Read Before Every JPM Performance

1. For the performance of this JPM, I will act as all those you need to talk to. Prior to providing direction to perform this task, I will provide you with the initial conditions and answer any questions. During task performance, I will identify the steps to be simulated, or discuss and provide cues as necessary. (Note, read the next only if conducting a plant JPM). With the exception of accessing panels, no plant equipment will be physically manipulated. Repositioning of devices will be simulated by discussion and acknowledged by my cues.

## C. Read Before Each Evaluated JPM

1. This evaluated JPM is a measure of your ability to perform this task independently. The US has determined that a verifier is not available and that additional verification will not be provided.

| | |
|---|---|
| **INITIAL CONDITIONS** | Given:<br>• The plant is shut down for a refueling outage<br>• All Reactor Recirculation Pumps are secured<br>• All Reactor Recirculation Pump discharge valves are open<br>• The discharge valves for Reactor Recirculation Pumps 11, 12, and 13 require manual closure<br>• All dose rates were taken at the Recirculation Pump Discharge Valve Handwheels<br>• Assume no radiation exposure is received in transit to and from the job location<br><br>**Evaluator:** *Ask trainee if he/she has any questions after presenting initial conditions* |

| | |
|---|---|
| **INITIATING CUE** | **(Operators Name)**, given supporting information, determine the dose required for each discharge valve closure. The discharge valves are to be closed in the order stated. Determine which discharge valves may be closed, if any, without exceeding your administrative dose limit. Your current dose is 1800 mrem. |

| | |
|---|---|
| **START TIME** | |

| | **PERFORMANCE** | **ACT. CODE**<br>P / S / NA | **EVALUATOR** |
|---|---|---|---|
| 1. | Provide repeat back of initiating cue<br><br>**Cue:** *Acknowledge repeat back providing correction if necessary.* | P | SAT / UNSAT<br><br>**STD:** Proper communications used. |
| 2. | Using the provided survey map, calculate the respective dose for each discharge valve closure as follows: | | |
| 2a. | Dose for Reactor Recirculation Pump 11 discharge valve: (1/3 hour x 150 mrem/hr) = 50 mrem | P | *** PASS / FAIL**<br>**STD:** Determines dose for closing discharge valve 11 is 50 mrem. |
| 2b. | Dose for Reactor Recirculation Pump 12 discharge valve: (1/3 hour x 300 mrem/hr) = 100 mrem | P | *** PASS / FAIL**<br>**STD:** Determines dose for closing discharge valve 12 is 100 mrem. |

| | PERFORMANCE | ACT. CODE<br>P / S / NA | EVALUATOR |
|---|---|---|---|
| 2c. | Dose for Reactor Recirculation Pump 13 discharge valve:<br>(1/3 hour x 225 mrem/hr) = 75 mrem | P | *** PASS / FAIL**<br>**STD:** Determines dose for closing discharge valve 13 is 75 mrem. |
| 3. | Determines remaining dose the operator is allowed to receive.<br>2000 mrem – 1800 mrem = 200 mrem | P | SAT / UNSAT<br>**STD:** Determines 200 mrem remain prior to exceeding administrative dose limits. |
| 4. | Determines which jobs can be completed without exceeding administrative dose limits. | P | *** PASS / FAIL**<br>**STD:** Determine that discharge valves for Reactor Recirculation Pumps 11 and 12 can be closed without exceeding the administrative dose limit.<br>Discharge Valve 11 – 50 mrem<br>Discharge Valve 12 – 100 mrem<br>Discharge Valve 13 – 75 mrem<br>Total dose for all three discharge valves would be 225 mrem. The operator can only close discharge valves 11 and 12 with exceeding the administrative dose limit. |
| 12. | Provides completed handout to examiner | P | SAT / UNSAT<br>**STD:** Proper communications used. |

| | |
|---|---|
| **TASK STANDARD** | Calculates the dose for each job and determines that only Reactor Recirculation Pump 11 and 12 discharge valves can be closed without exceeding the administrative dose limit of 2000 mrem/yr. |

| | |
|---|---|
| **STOP TIME** | |

# JPM Handout

| | |
|---|---|
| **INITIAL CONDITIONS** | Given:<br>• The plant is shut down for a refueling outage<br>• All Reactor Recirculation Pumps are secured<br>• All Reactor Recirculation Pump discharge valves are open<br>• The discharge valves for Reactor Recirculation Pumps 11, 12, and 13 require manual closure<br>• Assume no radiation exposure is received in transit to and from the job location |

| | |
|---|---|
| **INITIATING CUE** | **(Operators Name)**, given supporting information, determine the dose required for each discharge valve closure. The discharge valves are to be closed in the order stated. Determine which discharge valves may be closed, if any, without exceeding your administrative dose limit. Your current dose is 1800 mrem. |

1. **<u>Reactor Recirculation Pump 11 Discharge Valve</u>**
    a. Total time for discharge valve closure: 20 minutes
    b. Determine dose required for valve closure:

2. **<u>Reactor Recirculation Pump 12 Discharge Valve</u>**
    a. Total time for discharge valve closure: 20 minutes
    b. Determine dose required for valve closure:

3. **<u>Reactor Recirculation Pump 13 Discharge Valve</u>**
    a. Total time for discharge valve closure: 20 minutes
    b. Determine dose required for valve closure:

4. **<u>Determine which discharge valves may be closed without exceeding administrative dose limit:</u>**

Constellation. **RADIOLOGICAL SURVEY MAP** 

| DATE | TIME | RX POWER | H2 SCFM |
|---|---|---|---|
| Today | 0600 | U1[0.00] U2[100.00]% | 0 |

| STATION | UNIT | BLG | ELEV | ROOM # | MAP # |
|---|---|---|---|---|---|
| NMP | 1 | Drywell | 237 | N/A | 1DW237 |

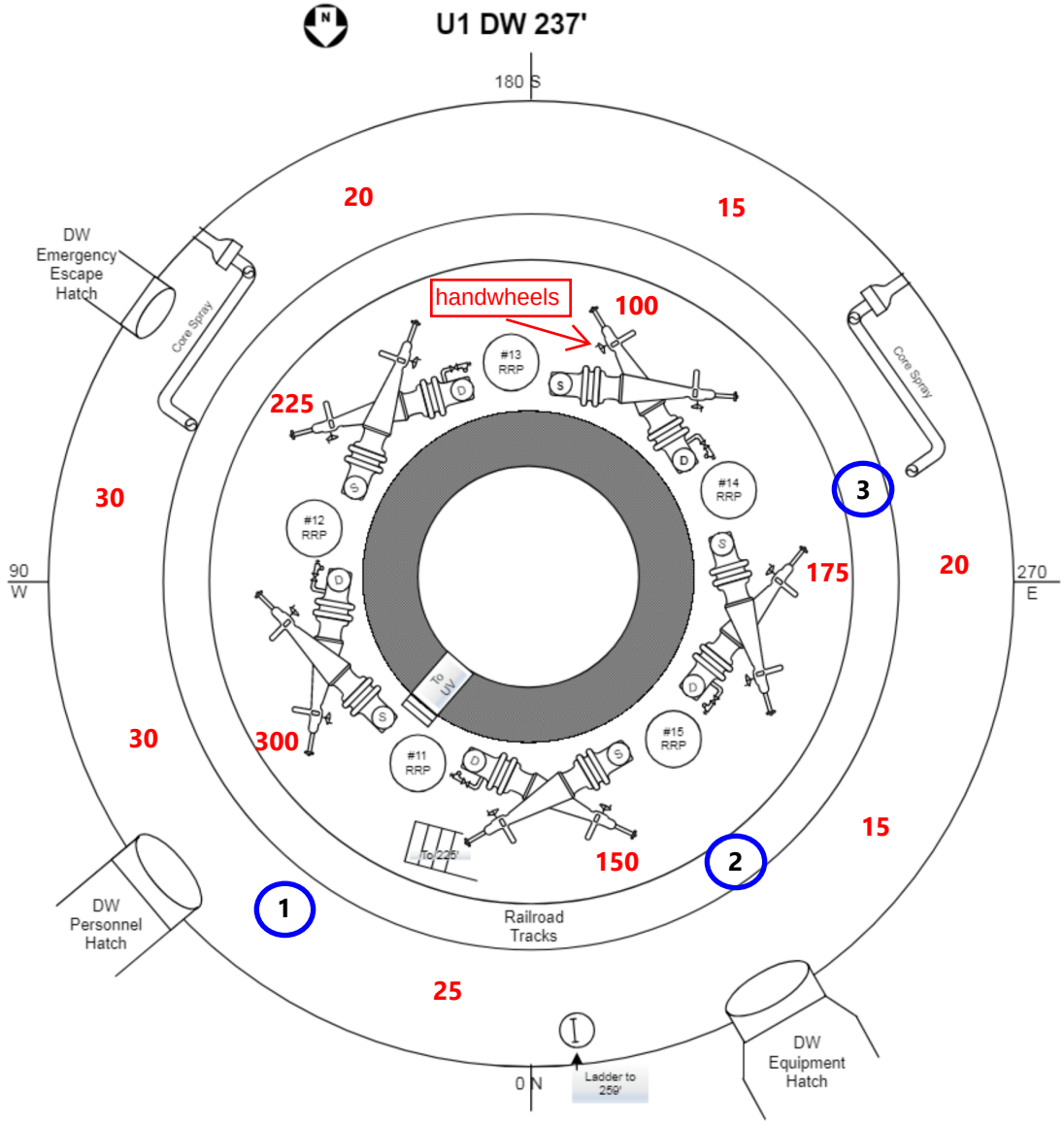
**AREA DESCRIPTION COMPONENT**: Open Area

**RADIATION AREA RANGE**: 10 TO 900 mRem/hr

**AIRBORNE RESULTS**: < 0.02 UID (Unidentified DAC)

**SURVEY #**: 2024-999999

*Unless otherwise noted on the survey, all radiation dose rates are in mRem/hr and contamination units are in dpm/100cm2 (beta/gamma contamination)




**NOTES:** N/A

**CONTAMINATION SURVEY UNIT KEY:** LAS dpm/ LAS | LAS ncpm/LAS | mRad/hr/100cm2 | dpm/100cm2

| # | LOCATION/AREA | LEVEL | UNITS | # | LOCATION/AREA | LEVEL | UNITS | # | LOCATION/AREA | LEVEL | UNITS |
|---|---|---|---|---|---|---|---|---|---|---|---|
| 1 | Grating | 5k | | 2 | Grating | 10k | | 3 | Grating | 20k | |

**ALPHA CONTAMINATION RESULTS**

BG/A Ratio Average:

| # | DPM / 100CM2 | RATIO (B&G:A) |
|---|---|---|
| 1 | < 6 | > 30000 |

**ELECTRON CAPTURE RESULTS**

| # | LEVEL | UNIT |
|---|---|---|

**SURVEYORS:** Radiation Protection

## SURVEY INSTRUMENT INFORMATION

| INST. TYPE | INST. NUMBER | CAL DUE | MDA* |
|---|---|---|---|
| | | | |
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| | | | |

* INSTRUMENT MDA IS IN DPM UNLESS OTHERWISE SPECIFIED

| SURVEY # | APPROVED BY | EMPLOYEE ID # |
|---|---|---|
| 2024-999999 | RP Supervisor | 999999 |

| DATE | TIME | SURVEY TYPE |
|---|---|---|
| Today | 0600 | Initial |

| RWP # | WO # | SURVEY DOSE |
|---|---|---|
| NM-1-24-9999 | N/A | 12.0 |

| TEMPLATE # | DESCRIPTION |
|---|---|
| 1DW237 | Open Area |

| NEXT DUE DATE | RX POWER | H2 SCFM |
|---|---|---|
| N/A | U1[0.00] U2[100.00]% | 0 |

| PEER CHECK BY | EMPLOYEE ID # | TIME |
|---|---|---|
| Radiation Protection | 111999 | 0630 |

| CORRECTION FOR SURVEY # |
|---|
| N/A |

| SURVEYOR NAME | EMPLOYEE ID # |
|---|---|
| Radiation Protection | 191919 |
| | |
| | |
| | |
| | |
| | |

## NOTES:

N/A

**COMMON POSTINGS, ABBREVIATIONS & SYMBOLS:** (VHRA)-Very High Rad Area (LHRA)-Locked High Rad Area (HRA)-High Rad Area (KO)-Keep Out (RA)-Radiation Area (RCA)-Radiologically Controlled Area (RM)-Radioactive Materials (ARA)-Airborne Radioactivity Area (CA)-Contaminated Area (HCA)-High Contamination Area (DRPBA)-DRP Buffer Area (L2AA)-Level 2 Alpha Area (L3AA)-Level 3 Alpha Area (RPB)- Radiation Protection Brief Required For Entry (NMR)-Neutron Monitoring Required (RPE)- Radiation Protection Escort Required For Entry (RWP)- Radiation Work Permit Required For Entry (DRFE)-Dosimetry Required For Entry (CRP7)-Contact RP Prior To Working Above 7FT (NEDS)-No Eating Drinking Smoking (RPSRPTE)- Radiation Protection Survey Required Prior To Entry (ND)-Non Detectable (NDB)-No Detectable Beta (t:#)-Telemetry (#)-Gamma Dose Rate(mRem/hr) (#B)-Beta Dose Rate(mRad/hr) (#N)-Neutron Dose Rate(mRem/hr) (*#/#)-Dose Rate @ Contact/30cm (H|C|W|K)-Head|Chest|Waist|Knee (K)-Kilo x 1000 (LAS)-Large Area Smear (DRP)-Discrete Radioactive Particle (DPM)-Disintegrations per Minute (NCPM)-Net Counts per Minute ( # )-Smear Location ( ..... )-Radiological Boundary ( SOP )-Step Off Pad ( △ )-Air Sample Location

Training Id: **2024 NRC Admin SRO COO1** Revision: **0.0**

Title: **Reactivate SRO Licenses In Accordance With OP-AA-105-102**

**Approvals:**

| | Signature / Printed Name | Date |
|---|---|---|
| Developed By | Patrick Ryan | 1/26/2024 |
| Validated By | Branden Dempsey | 04/01/2024 |
| Facility Reviewer | Dave Mason | |

Approximate Duration: 15 minutes

**Documentation of Performance:**

Performer: _____

Evaluator: _____

Start Time: _____ Stop Time: _____ Completion Time _____

Grade: **Pass / Fail**

Comments: _____

Evaluators Signature: _____ Date: _____

## References

1. NUREG 1123, 2.1.4 (3.8)
2. OP-AA-105-102 NRC Active License Maintenance

Instructor Information

## A. JPM Information

1. Description

   a. This JPM tests the operator's ability to evaluate license reactivation documentation.

2. Task Information:

   a. S-ODP-TQS0101-00002, Reactivate an inactive license

   b. K/A 2.1.4 (3.8), Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, "no-solo" operation, maintenance of active license status, 10CFR55, etc.

3. Evaluation / Task Criteria

| | |
|---|---|
| Evaluation Method | Perform |
| Evaluation Location | Classroom |
| Time Critical Task | No |
| Alternate Path | No |
| LOD >1.0 | Yes |

4. Recommended Start Location

   a. Classroom

5. JPM Setup (if required)

   a. Provide copy of OP-AA-105-102
   b. If multiple operators are going to perform the JPM at the same time, ensure each operator has a copy of OP-AA-105-102.
   c. Provide marked up copies of OP-AA-105-102, attachment 2.

## B. Read Before Every JPM Performance

1. For the performance of this JPM, I will act as all those you need to talk to. Prior to providing direction to perform this task, I will provide you with the initial conditions and answer any questions. During task performance, I will identify the steps to be simulated, or discuss and provide cues as necessary. (Note, read the next only if conducting a plant JPM). With the exception of accessing panels, no plant equipment will be physically manipulated. Repositioning of devices will be simulated by discussion and acknowledged by my cues.

## C. Read Before Each Evaluated JPM

1. This evaluated JPM is a measure of your ability to perform this task independently. The US has determined that a verifier is not available and that additional verification will not be provided.

| | |
|---|---|
| **INITIAL CONDITIONS** | Given:<br>• Due to recent SRO attrition, several SRO licenses are being reactivated.<br>• Two inactive license holders are completing the requirements to reactivate their licenses.<br>• OP-AA-105-102, Attachment 2, Reactivation of License Log is completed up to the Shift Manager review.<br><br>**Evaluator:** *Ask trainee if he/she has any questions after presenting initial conditions* |

| | |
|---|---|
| **INITIATING CUE** | **(Operators Name)**, perform the Shift Manager review of OP-AA-105-102, Attachment 2, Reactivation of License Log, and return it to me. |

| | |
|---|---|
| **START TIME** | |

| | **PERFORMANCE** | **ACT. CODE**<br>P / S / NA | **EVALUATOR** |
|---|---|---|---|
| 1. | Provide repeat back of initiating cue<br><br>**Cue:** *Acknowledge repeat back providing correction if necessary.* | P | SAT / UNSAT<br><br>**STD:** Proper communications used. |
| 2. | Obtain a copy of the reference procedure and review / utilize the correct section of the procedure. | P | SAT / UNSAT<br><br>**STD:** OP-AA-105-102 obtained |
| 3. | Reviews shift hours for SRO #1 | P | SAT / UNSAT<br>**STD:** Reviews OP-AA-105-102 attachment 2 |
| 4. | Determines SRO #1 does not meet the requirements for license reactivation. | P | ***PASS / FAIL**<br><br>**STD:** Recognize that not all required shift hours were completed in the same calendar quarter, as required in step 4.2.2. |

| | PERFORMANCE | ACT. CODE<br>P / S / NA | EVALUATOR |
|---|---|---|---|
| 5. | Reviews shift hours for SRO #2 | P | SAT / UNSAT<br>**STD:** Reviews OP-AA-105-102 attachment 2 |
| 6. | Determines SRO #2 does not meet the requirements for license reactivation. | P | ***PASS / FAIL**<br><br>**STD:** Recognize that not all required shift hours were completed in a shift position required by technical specifications. |
| 9. | Reports SRO reactivation status | P | SAT / UNSAT<br><br>**STD:** Proper communications used. |

| | |
|---|---|
| **TASK STANDARD** | Shift Manager review of attachment 2, Reactivation of License Log, is complete recognizing neither SRO meets the requirements for license reactivation |

| | |
|---|---|
| **STOP TIME** | |

# JPM Handout

| | |
|---|---|
| **INITIAL CONDITIONS** | Given:<br>• Due to recent SRO attrition, several SRO licenses are being reactivated.<br>• Two inactive license holders are completing the requirements to reactivate their licenses.<br>• OP-AA-105-102, Attachment 2, Reactivation of License Log is completed up to the Shift Manager review. |

| | |
|---|---|
| **INITIATING CUE** | **(Operators Name)**, perform the Shift Manager review of OP-AA-105-102, Attachment 2, Reactivation of License Log, and return it to me. |

Training Id: **2024 NRC Admin SRO COO2** Revision: **0.0**

Title: **Determine Operator Qualifications Using LMS**

**Approvals:**

| | Signature / Printed Name | Date |
|---|---|---|
| Developed By | Patrick Ryan | 1/29/2024 |
| Validated By | Brandon Dempsey | 4/1/2024 |
| Facility Reviewer | Dave Mason | 2/24/2024 |

Approximate Duration: 15 minutes

**Documentation of Performance:**

Performer: _____

Evaluator: _____

Start Time: _____ Stop Time: _____ Completion Time _____

Grade: **Pass / Fail**

Comments: _____

Evaluators Signature: _____ Date: _____

## References

1. NUREG 1123, 2.1.8 (4.1)

Instructor Information

## A. JPM Information

1. Description

   a. This JPM tests the senior operator's ability to evaluate an operator's qualifications to perform specific tasks.

2. Task Information:

   a. NS-FLDSUP-001 – Lead Prejob Briefs for Operator Teams

   b. K/A 2.1.8 (4.1), Ability to coordinate personnel activities outside the control room.

3. Evaluation / Task Criteria

| | |
|---|---|
| Evaluation Method | Perform |
| Evaluation Location | Classroom |
| Time Critical Task | No |
| Alternate Path | No |
| LOD >1.0 | Yes |

4. Recommended Start Location

   a. Classroom

5. JPM Setup (if required)

   a. Provide copies of LMS printouts for each applicant
   b. Provide copies of OP-AA-109-101

## B. Read Before Every JPM Performance

1. For the performance of this JPM, I will act as all those you need to talk to. Prior to providing direction to perform this task, I will provide you with the initial conditions and answer any questions. During task performance, I will identify the steps to be simulated, or discuss and provide cues as necessary. (Note, read the next only if conducting a plant JPM). With the exception of accessing panels, no plant equipment will be physically manipulated. Repositioning of devices will be simulated by discussion and acknowledged by my cues.

## C. Read Before Each Evaluated JPM

1. This evaluated JPM is a measure of your ability to perform this task independently. The US has determined that a verifier is not available and that additional verification will not be provided.

| | |
|---|---|
| **INITIAL CONDITIONS** | Given:<br>• A maintenance activity is being planned.<br>• Qualifications need to be verified for the following aspects of the maintenance activity:<br>o Write the tagout for a system (which will include disabling fire suppression systems)<br>o Accept and Release the tagout<br>o Hang/remove tags and align systems in accordance with the tagout<br>o Perform duties as the fire watch while the tagout is active<br><br>**Evaluator:** *Ask trainee if he/she has any questions after presenting initial conditions* |

| | |
|---|---|
| **INITIATING CUE** | **(Operators Name)**, given the LMS sheets and the tasks required, determine which operator(s) is/are qualified to perform the tasks. |

| | |
|---|---|
| **START TIME** | |

| | **PERFORMANCE** | **ACT. CODE**<br>P / S / NA | **EVALUATOR** |
|---|---|---|---|
| 1. | Provide repeat back of initiating cue<br><br>**Cue:** *Acknowledge repeat back providing correction if necessary.* | P | SAT / UNSAT<br><br>**STD:** Proper communications used. |
| 2. | Obtains and reviews the tasks and the LMS sheets and determines who is qualified to perform the tasks as shown on the KEY.<br><br>**Note**: Applicant can evaluate the operators in any order. | P | SAT / UNSAT<br><br>**STD:** LMS sheets and task information obtained and reviewed. |
| 2a. | Evaluates RO1 | P | ***PASS / FAIL**<br>**STD**: Determines RO1 can perform tasks #1, #2, & #3. |

| | PERFORMANCE | ACT. CODE<br>P / S / NA | EVALUATOR |
|---|---|---|---|
| 2b. | Evaluates RO2 | P | ****PASS / FAIL**<br><br>**STD:** Determines RO2 can perform tasks #1, #3, & #4. |
| 2c. | Evaluates EO1 | P | ***PASS / FAIL**<br>**STD**: Determines EO1 can perform tasks #3 and #4 |
| 2d. | Evaluates EO2 | P | ***PASS / FAIL**<br>**STD:** Determines EO2 can perform only task #4 |
| 9. | Reports qualification status | P | SAT / UNSAT<br><br>**STD:** Proper communications used. |

| | |
|---|---|
| **TASK STANDARD** | Applicant determines which Operators are qualified to perform the tasks using LMS. |

| | |
|---|---|
| **STOP TIME** | |

# JPM Handout

| | |
|---|---|
| **INITIAL CONDITIONS** | Given:<br>• A maintenance activity is being planned.<br>• Qualifications need to be verified for the following aspects of the maintenance activity:<br>o Write the tagout for a system (which will include disabling fire suppression systems)<br>o Accept and Release the tagout<br>o Hang/remove tags and align systems IAW the tagout<br>o Perform duties as the fire watch while the tagout is active |

| | |
|---|---|
| **INITIATING CUE** | **(Operators Name)**, given the LMS sheets and the tasks required, determine which operator is qualified to perform the tasks. |

| **Task Number** | **Task Description** |
|---|---|
| 1 | Write a tagout for a system (which will include disabling fire suppression systems) |
| 2 | Accept and Release the tagout |
| 3 | Hang/remove the tags and align the systems IAW the tagout |
| 4 | Perform the duties of a fire watch while the tagout is active |

| **Is the Operator Qualified to Perform the Task? (Yes / No)** | | | | |
|---|---|---|---|---|
| **Operator** | **Task 1** | **Task 2** | **Task 3** | **Task 4** |
| RO 1 | | | | |
| RO 2 | | | | |
| EO 1 | | | | |
| EO 2 | | | | |

*******Answer Key*******
**Do Not Provide to Applicant**

| Task Number | Task Description |
|---|---|
| 1 | Write a tagout for a system (which will include disabling fire suppression systems)<br><br>Qualification Required: N-NM-SA-TAG-WRITER-PREPARER |
| 2 | Accept and Release the tagout<br><br>Qualification Required: N-AN-SA-TAG-TO-HOLDER |
| 3 | Hang/remove the tags and align the systems IAW the tagout<br><br>Qualification Required: N-AN-SA-TAG-APPLIER-REMOVER |
| 4 | Perform the duties of a fire watch while the tagout is active<br><br>Qualification Required: N-AN-SA-FIREWATCH |

| Is the Operator Qualified to Perform the Task? (Yes / No) | | | | |
|---|---|---|---|---|
| **Operator** | **Task 1** | **Task 2** | **Task 3** | **Task 4** |
| RO 1 | Yes | Yes | Yes | No |
| RO 2 | Yes | No | Yes | Yes |
| EO 1 | No | No | Yes | Yes |
| EO 2 | No | No | No | Yes |

*******Answer Key*******
**Do Not Provide to Applicant**

Training Id: **2024 NRC Admin SRO EC** Revision: **0.0**

Title: **Review and Approval of Completed Surveillance Test, N1-ST-M8, Reactor Building Emergency Ventilation System Operability Test**

**Approvals:**

| | Signature / Printed Name | Date |
|---|---|---|
| Developed By | Patrick Ryan | 1/29/2024 |
| Validated By | Brandon Dempsey | 4/1/2024 |
| Facility Reviewer | Dave Mason | |

Approximate Duration: 20 minutes

**Documentation of Performance:**

Performer: _____

Evaluator: _____

Start Time: _____ Stop Time: _____ Completion Time _____

Grade: **Pass / Fail**

Comments: _____

Evaluators Signature: _____ Date: _____

## References

1. N1-ST-M8 Reactor Building Emergency Ventilation System Operability Test
2. Unit 1 Technical Specifications
3. NUREG 1123, 2.2.12 (4.1)

Instructor Information

## A. JPM Information

1. Description

   a. This JPM tests the operator's ability to use and comply with facility surveillance procedures. Given a partial copy of N1-ST-M8, Reactor Building Emergency Ventilation System Operability Test, the candidate will review the results and identify any discrepancies.

2. Task Information:

   a. N1-261000-01015 Perform or Support Performance of N1-ST-M8 Reactor Building Emergency Ventilation System Operability Test (Branching Task)

   b. K/A 2.2.12 (4.1) Knowledge of Surveillance Procedures

3. Evaluation / Task Criteria

| | |
|---|---|
| Evaluation Method | Perform |
| Evaluation Location | Classroom |
| Time Critical Task | No |
| Alternate Path | No |
| LOD >1.0 | Yes |

4. Recommended Start Location

   a. Classroom

5. JPM Setup (if required)

   a. Provide marked up copies of N1-ST-M8, Reactor Building Emergency Ventilation System Operability Test
   b. Verify Technical Specifications are available
   c. Verify calculators are available

## B. Read Before Every JPM Performance

1. For the performance of this JPM, I will act as all those you need to talk to. Prior to providing direction to perform this task, I will provide you with the initial conditions and answer any questions. During task performance, I will identify the steps to be simulated, or discuss and provide cues as necessary. (Note, read the next only if conducting a plant JPM). With the exception of accessing panels, no plant equipment will be physically manipulated. Repositioning of devices will be simulated by discussion and acknowledged by my cues.

## C. Read Before Each Evaluated JPM

1. This evaluated JPM is a measure of your ability to perform this task independently. The US has determined that a verifier is not available and that additional verification will not be provided.

| | |
|---|---|
| **INITIAL CONDITIONS** | Given:<br>• The plant is operating at 100% power.<br>• N1-ST-M8 has just been completed up to the Operations Review.<br>• You are the Shift Manager<br><br>**Evaluator:** *Ask trainee if he/she has any questions after presenting initial conditions* |

| | |
|---|---|
| **INITIATING CUE** | **(Operators Name)**, using the completed surveillance provided, complete Section 10.1 Operations Review and Section 10.2 SM Review.<br><br>Report findings and any required actions to the examiner. |

| | |
|---|---|
| **START TIME** | |

| | **PERFORMANCE** | **ACT. CODE**<br>P / S / NA | **EVALUATOR** |
|---|---|---|---|
| 1. | Provide repeat back of initiating cue<br><br>**Cue:** *Acknowledge repeat back providing correction if necessary.* | P | SAT / UNSAT<br><br>**STD:** Proper communications used. |
| 2. | Obtain a copy of the reference procedure and review / utilize the correct section of the procedure.<br><br>**Note**: Evaluator is to provide the marked up copy of N1-ST-M8, Reactor Building Emergency Ventilation System Operability Test, to the candidate | P | SAT / UNSAT<br><br>**STD:** N1-ST-M8 obtained. General Test Methods, References / Commitments and the Precaution and Limitation reviewed. |
| 3. | Determines status of the completed surveillance.<br><br>Reviews N1-ST-M8, Reactor Building Emergency Ventilation System Operability Test and identifies the following: | | |

| | PERFORMANCE | ACT. CODE<br>P / S / NA | EVALUATOR |
|---|---|---|---|
| 3a. | Performs Operations Review step 10.1.1.<br><br>**Note**: May inform examiner verbally of issues | P | ***PASS / FAIL**<br><br>**STD:** Identifies a mathematical error at step 8.1.9.c for the sum of Train 11 Filter D/P's. Actual sum of Train 11 Filer D/P's is greater than 6"H20. Step 10.1.1.c should be marked UNSAT. |
| 3b.. | Performs Operations Review step 10.1.1<br><br>**Note**: May inform examiner verbally of issues | P | ***PASS / FAIL**<br><br>**STD**: Identifies RBEVS Loop 12 flow is less than 1440 cfm. Step 10.1.2.b should be marked UNSAT. |
| 4. | Identifies actions required for components being declared inoperable per the Acceptance Criteria<br><br>**Note**: May inform examiner verbally of issues | P | SAT / UNSAT<br><br>**STD:** The following actions are required per step 10.2 SM Review: Enter component in ESL, enter applicable TS LCO, make entry in SM log, place EST on component, write an IR, notify U1 SOS or designee. |

| | |
|---|---|
| **TASK STANDARD** | Determination of the status of this surveillance has been made with the following errors identified:<br>• Train 11 Filter D/P is greater than 6"H20. Train 11 is inoperable<br>• Train 12 flow is less than 1440 cfm. Train 12 is inoperable |

| | |
|---|---|
| **STOP TIME** | |

# JPM Handout

| | |
|---|---|
| **INITIAL CONDITIONS** | Given:<br>• The plant is operating at 100% power.<br>• N1-ST-M8 has just been completed up to the Operations Review.<br>• You are the Shift Manager |

| | |
|---|---|
| **INITIATING CUE** | **(Operators Name)**, using the completed surveillance provided, complete Section 10.1 Operations Review and Section 10.2 SM Review.<br><br>Report findings and any required actions to the examiner. |

Training Id: **2024 NRC Admin SRO EP** Revision: **0.0**

Title: **Emergency Event Classification & Notification**

**Approvals:**

| | Signature / Printed Name | Date |
|---|---|---|
| Developed By | Patrick Ryan | 1/26/2024 |
| Validated By | Brandon Dempsey | 4/1/2024 |
| Facility Reviewer | Dave Mason | |

Approximate Duration: 20 minutes

**Documentation of Performance:**

Performer: _____

Evaluator: _____

Start Time: _____ Stop Time: _____ Completion Time _____

Grade: **Pass / Fail**

Comments: _____

Evaluators Signature: _____ Date: _____

## References

1. N1-EOP-C5 (1-2) RPV Control - ATWS
2. N1-EOP-PC Primary Containment Control (1-2)
3. EP-AA-1013, Addendum 3, Appendix 1, NMP Unit 1 EAL Wallboard
4. EP-AA-111, Emergency Classification and Protective Action Recommendations
5. NUREG 1123, 2.4.41 (4.6)

Instructor Information

## A. JPM Information

1. Description

   a. This JPM tests the SRO's ability to classify an event and perform the station specific Part 1 Notification Form.

2. Task Information:

   a. NS-EP101-03018, Prepare and Approve a Part 1 Notification Fact Sheet for an Emergency Classification.

   b. K/A 2.4.41 (4.6), Knowledge of the emergency action level thresholds and classifications.

3. Evaluation / Task Criteria

| | |
|---|---|
| Evaluation Method | Perform |
| Evaluation Location | Classroom |
| Time Critical Task | Yes |
| Alternate Path | No |
| LOD >1.0 | Yes |

4. Recommended Start Location

   a. Classroom

5. JPM Setup (if required)

   a. Provide sufficient copies of the following:
      1) EP-AA-112-100-F-01 Shift Emergency Director Checklist
      2) EP-AA-114-F-02 BWR Release in Progress Determination Guidance
      3) EP-CE-114-100-F-05 Part 1 Notification Facts Sheet
      4) EP-AA-112-100-F-06 ERO Notification Form
      5) EP-AA-112-F-09 Emergency Public Address Announcements
      6) EPA-28 Simplified Classification Sequence
      7) EP-AA-1013 Addendum 3 Nine Mile Point Unit 1 Emergency Classification Technical Bases
      8) EP-AA-1013 Addendum 3 Appendix 1 NMP Unit 1 Wallboard
      9) N1-EOP-PC

## B. Read Before Every JPM Performance

1. For the performance of this JPM, I will act as all those you need to talk to. Prior to providing direction to perform this task, I will provide you with the initial conditions and answer any questions. During task performance, I will identify the steps to be simulated, or discuss and provide cues as necessary. (Note, read the next only if conducting a plant JPM). With the exception of accessing panels, no plant equipment will be physically manipulated. Repositioning of devices will be simulated by discussion and acknowledged by my cues.

## C. Read Before Each Evaluated JPM

1. This evaluated JPM is a measure of your ability to perform this task independently. The US has determined that a verifier is not available and that additional verification will not be provided.

| | |
|---|---|
| **INITIAL CONDITIONS** | Given:<br>• The plant was operating at 100% power when a main steam line break caused an automatic closure of all Main Steam Isolation Valves<br>• An automatic reactor SCRAM was generated from the MSIV closure, however multiple control rods failed to insert<br>• All manual and ARI actions to shut down the reactor have been unsuccessful<br>• Reactor pressure is stable at 900 psig with the following pressure control methods<br>o Both ECs in service<br>o 4 ERVs open<br>• Torus water level is 12.5 feet and rising slowly<br>• Torus water temperature is 120°F and rising slowly<br><br>**Evaluator:** *Ask trainee if he/she has any questions after presenting initial conditions* |

| | |
|---|---|
| **INITIATING CUE** | The time that conditions are available in the Control Room is upon completion of the initiating cue.<br><br>**(Operators Name)**, classify this event and complete the station specific initial notification form in accordance with EP-AA-112-100-F-01 SHIFT EMERGENCY DIRECTOR CHECKLIST. |

| | |
|---|---|
| **START TIME** | |

| | **PERFORMANCE** | **ACT. CODE**<br>P / S / NA | **EVALUATOR** |
|---|---|---|---|
| 1. | Provide repeat back of initiating cue<br><br>**Cue:** *Acknowledge repeat back providing correction if necessary.* | P | SAT / UNSAT<br><br>**STD:** Proper communications used. |
| 2. | Obtain a copy of the reference procedure and review / utilize the correct section of the procedure. | P | SAT / UNSAT<br><br>**STD:** Obtains EAL Flowchart and Shift Emergency Director binder. |
| **Evaluator Note** | • Steps may not be performed in the order exactly as listed in the JPM.<br>• The Operator may choose to utilize the Simplified Classification Sequence guide NMP EP AID EPA-28. | | |

| | PERFORMANCE | ACT. CODE<br>P / S / NA | EVALUATOR |
|---|---|---|---|
| 3. | Starts EP-AA-112-100-F-01, Shift Emergency Director Checklist | P | SAT / UNSAT<br>**STD:** Maintains place keeping throughout performance of this checklist |
| 4. | If the event is classified as an SITE AREA EMERGENCY, then perform the following:<br><br>RECORD the EAL and declaration time | P | ***PASS / FAIL**<br><br>**STD**: Classifies the event as a **SITE AREA EMERGENCY MS3 and/or FS1**<br><br>**Critical Step Justification**: Emergency classification and declaration shall be completed as soon as possible but no later than 15 from the time indications an EAL threshold being met or exceeded are available in the Control Room. The time written on the ED Checklist or SED job aid will count as declaration time if no announcement is made.<br><br>The time difference below must be 15 minutes or less:<br><br>**JPM Start Time**:<br><br>_____<br><br>**Declaration Time**:<br><br>_____ |

| | PERFORMANCE | ACT. CODE<br>P / S / NA | EVALUATOR |
|---|---|---|---|
| 5. | ANNOUNCE the event classification, possible escalation paths, and declaration time to the Control Room staff.<br><br>**Cue**: If asked for a peer check, report that the STA/IA is not available for peer check<br><br>DECLARE the event by announcing the following (from EPA-28):<br>- I am declaring a(n) (EAL Classification) at (time) due to (brief reason). Escalation criteria is(are) (criteria) and I am assuming the role as Shift ED." | P | SAT/UNSAT<br><br>STD: Announces to Control Room personnel the EAL classification, possible escalation paths, and declaration times. |
| 6. | Calls Shift Communicators and Shift dose Assessors to their respective Control Rooms | P | SAT / UNSAT<br><br>**STD**: Uses Gai-Tronics to call personnel to Control Room |
| 7. | Security Related Events, GO TO Step 4.1 Security Related Events.<br><br>**Note**: Determines NO Security Event is in progress (N/A for this scenario) | P | SAT / UNSAT<br><br>**STD:** Determines that this is not a Security Event, step is N/A |
| 8. | If all the ERO has not been activated, then DIRECT activation of the ERO Notification using 1, "Actual Event Respond to Facility" or 3, "Actual Event Alternative Facility Response" as appropriate, per EP-AA-112-100-F-06.<br><br>**Cue**: As Shift Communicator, acknowledge direction to notify ERO<br><br>**Note**: Record time SED directs the Shift Communicator to activate the ERO as recorded in the SED Checklist | P | SAT/UNSAT<br><br>**STD**: Accurately fills out and completes ERONS Notification Details (CNG)<br>• Circles NMP<br>• Circles "Actual Event Respond to Facility"<br><br>**Note**: Operator may only call U2 and direct them to activate ERONs. This satisfies the requirement. |

| | PERFORMANCE | ACT. CODE<br>P / S / NA | EVALUATOR |
|---|---|---|---|
| 9. | SELECT the Emergency Public Address Announcements from the form and DIRECT performance of the public address announcement within 15 minutes of the event classification using EP-AA-112-F-09<br><br>**Cue**: As Shift Communicator, acknowledge direction. Wait 1 minute and then report that Station announcement has been made. | P | SAT / UNSAT<br><br>**STD**: Directs Shift communicator to makes Emergency PA Announcement |

| | PERFORMANCE | ACT. CODE<br>P / S / NA | EVALUATOR |
|---|---|---|---|
| 10. | DIRECT performance of State/Local notifications within 15 minutes of the event classification as required per the Notifications procedure.<br><br>• Notification Procedure: (Tab 3)<br>• Notification form: (Tab 4)<br>• Release in Progress Determination Guidance: (Tab 21)<br>• Release in Progress Determination Guidance Technical Basis: (Tab 26)<br><br>**Cue**: If operators asks for a peer check, report that the STA/IA is not available for peer check<br><br>Provides the completed form to Shift Communicator and directs them to notify State and Local.<br><br>**Cue**: As Shift Communicator, acknowledge direction and record time Part 1 Notification received.<br><br>**Note**: The accurate declaration "MS3 and/or FS1" is not required for a passing grade determination. The requirement is the Shift Manager make the correct event classification. In this case the SM/ED needs to declare a "SITE AREA EMERGENCY" to receive credit. (per TQ-AA-155, Attachment 6, Step 2.4) | P | ***PASS / FAIL**<br><br>**STD**: Accurately fills in and completes all information inside SHADED boxes of Part 1 Notifications.<br><br>Marks "Nine Mile Point Unit 1" on first row<br>Step 2: B<br>Step 3: C<br>Step 4: Date /Time<br>Step 5: A<br>Step 6: A<br>This notification is required to be completed within 15 minutes of the Emergency Declaration. This means that the state and local are required to be notified within 15 minutes.<br><br>**Critical Step Justification**: This information is required to be accurate when providing transmitting this information to outside organizations and the NRC<br><br>The time difference below must be 15 minutes or less:<br><br>**Declaration Time**:<br>_____<br><br>**Notification Time**:<br>_____ |

| | PERFORMANCE | ACT. CODE<br>P / S / NA | EVALUATOR |
|---|---|---|---|
| 11. | The Operator may perform the following steps before completing the Part 1 Notification Fact Sheet. These steps are NOT required to be completed for Satisfactory completion of the JPM | | |
| 12. | DIRECT performance of required NRC notifications immediately following notification of the appropriate State and Local agencies:<br>• Notification Procedure: (Tab 20)<br>• Notification Form: (Tab 10) | P | SAT/UNSAT<br><br>**STD**: Accurately fills out and completes the Reactor Plant Event Notification Worksheet (NRC Form 361) |
| 13. | If a release is in progress, then:<br><br>**Note**: Operator should determine there is NO release in progress based on initial conditions and this step is N/A | P | SAT / UNSAT<br><br>**STD**: Determines that there is not a Release in Progress, step is N/A |
| 14. | If the emergency involves any type or size LOCA, then ENSURE Control Room(s) emergency ventilation is initiated.<br><br>**Note**: Operator should determine there is NO LOCA progress based on initial conditions and this step is N/A | P | SAT / UNSAT<br><br>**STD**: Determines there is no LOCA in progress, this step is N/A |
| 15. | Determines if SAFER FLEX Equipment is deemed necessary.<br><br>**Note**: Operator should determine there is no need for SAFER FLEX equipment, this step is N/A | P | SAT / UNSAT<br><br>**STD**: Determines there is no need for SAFER FLEX equipment, this step is N/A |
| **Termination Cue** | Once the Part 1 Notification Fact Sheet is completed and the Shift communicator is Directed to make State and Local notifications, report “At this time, the JPM is completed” | | |

| | |
|---|---|
| **TASK STANDARD** | A SITE AREA EMERGENCY (MS3 or FS1) has been classified and NMP Notification Fact Sheet-Part 1 is completed. |

| | |
|---|---|
| **STOP TIME** | |

# JPM Handout

| | |
|---|---|
| **INITIAL CONDITIONS** | Given:<br>• The plant was operating at 100% power when a main steam line break caused an automatic closure of all Main Steam Isolation Valves<br>• An automatic reactor SCRAM was generated from the MSIV closure, however multiple control rods failed to insert<br>• All manual and ARI actions to shut down the reactor have been unsuccessful<br>• Reactor pressure is stable at 900 psig with the following pressure control methods<br>  o Both ECs in service<br>  o 4 ERVs open<br>• Torus water level is 12.5 feet and rising slowly<br>• Torus water temperature is 120°F and rising slowly |

| | |
|---|---|
| **INITIATING CUE** | The time that conditions are available in the Control Room is upon completion of the initiating cue.<br><br>**(Operators Name)**, classify this event and complete the station specific initial notification form in accordance with EP-AA-112-100-F-01 SHIFT EMERGENCY DIRECTOR CHECKLIST. |

Emergency Meteorology Report

| Last 15 Minute Emergency Meteorology Report Data | | | | | | | |
|---|---|---|---|---|---|---|---|
| Data from Nine Mile Point Met System | | | | | | | |
| Date: XX/XX/XXXX | | | | Time (Local): XX:XX:XX | | | |
| Elevated | | | | Ground | | | |
| 200' | Wind Speed (Main) | 37 | (mph) | 30' | Wind Speed (Main) | 32 | (mph) |
| 200' | Wind Dir From (Main) | 352 | (deg) | 30' | Wind Dir From (Main) | 350 | (deg) |
| 200' | Delta Temperature | 0.05 | (deg F) | 100' | Delta Temperature | 0.7 | (deg F) |
| | Stability Class | E | | | Stability Class | C | |
| 30' | Air Temperature | 50 | (deg F) | | Precipitation (15 min) | 1.02 | (in) |

Training Id: **2024 NRC Admin SRO RC** Revision: **0.0**

Title: **Determine Actions for Inoperable Stack Radiation Monitor**

**Approvals:**

| | Signature / Printed Name | Date |
|---|---|---|
| Developed By | Patrick Ryan | 1/26/2024 |
| Validated By | Brandon Dempsey | 4/1/2024 |
| Facility Reviewer | Dave Mason | |

Approximate Duration: 20 minutes

**Documentation of Performance:**

Performer: _____

Evaluator: _____

Start Time: _____ Stop Time: _____ Completion Time _____

Grade: **Pass / Fail**

Comments: _____

Evaluators Signature: _____ Date: _____

## References

1. N1-ARP-H3
2. N1-OP-50B
3. NUREG 1123, 2.3.11 (4.3)
4. CY-NM-170-301 Offsite Dose Calculation Manual

Instructor Information

## A. JPM Information

1. Description

   a. This JPM tests the operator's ability to determine actions required for a failed gaseous radiation monitor per the Offsite Dose Calculation Manual.

2. Task Information:

   a. NS-PS115-03001 Maintain cognizance of entry into technical specification action statements.

   b. K/A 2.3.11 (4.3) Ability to control radiation releases

3. Evaluation / Task Criteria

| | |
|---|---|
| Evaluation Method | Perform |
| Evaluation Location | Classroom |
| Time Critical Task | No |
| Alternate Path | No |
| LOD >1.0 | Yes |

4. Recommended Start Location

   a. Classroom

5. JPM Setup (if required)

   a. Provide copy of N1-ARP-H1-3-8
   b. Provide copy of CY-NM-170-301

## B. Read Before Every JPM Performance

1. For the performance of this JPM, I will act as all those you need to talk to. Prior to providing direction to perform this task, I will provide you with the initial conditions and answer any questions. During task performance, I will identify the steps to be simulated, or discuss and provide cues as necessary. (Note, read the next only if conducting a plant JPM). With the exception of accessing panels, no plant equipment will be physically manipulated. Repositioning of devices will be simulated by discussion and acknowledged by my cues.

## C. Read Before Each Evaluated JPM

1. This evaluated JPM is a measure of your ability to perform this task independently. The US has determined that a verifier is not available and that additional verification will not be provided.

| | |
|---|---|
| **INITIAL CONDITIONS** | Given:<br>• The plant is operating at 100% power.<br>• Annunciator H1-3-8, STACK GAS MONITORING TROUBLE alarmed.<br>• Computer Point C061, OGESMS RAD MON STATUS - TRBL alarmed.<br>• OGESMS radiation monitor RN-10A indicates downscale.<br>• RN-10A input to H1-3-8 has been defeated per S-ARP-001.<br>• Annunciator H1-3-8 is now clear.<br><br>**Evaluator:** *Ask trainee if he/she has any questions after presenting initial conditions* |

| | |
|---|---|
| **INITIATING CUE** | **(Operators Name)**, determine the appropriate actions, compensatory measures, and reporting requirements in response to ARP H1-3-8. Record your findings on the provided attachment. |

| | |
|---|---|
| **START TIME** | |

| | **PERFORMANCE** | **ACT. CODE**<br>P / S / NA | **EVALUATOR** |
|---|---|---|---|
| 1. | Provide repeat back of initiating cue<br><br>**Cue:** *Acknowledge repeat back providing correction if necessary.* | P | SAT / UNSAT<br><br>**STD:** Proper communications used. |
| 2. | Obtain a copy of the reference procedure and review / utilize the correct section of the procedure. | P | SAT / UNSAT<br><br>**STD:** CY-NM-170-301 and N1-ARP-H1-3-8 obtained |
| 3. | Requests Chemistry to implement CY-NM-170-309 OGESMS OFF-NORMAL PROCEDURES | P | SAT / UNSAT<br>**STD:** In accordance with N1-ARP-H3-1-8 |

| | PERFORMANCE | ACT. CODE<br>P / S / NA | EVALUATOR |
|---|---|---|---|
| 4. | Determines OGESMS is inoperable and Secondary Containment is required. Determines 8 Drywell and Torus Vent and Purge Valves (201-07, 201-08, 201-09, 201-10, 201-16, 201-17, 201-31, and 201-32) must be closed and administratively tagged.<br><br>**Note:** This action may be determined using either N1-ARP-H1-3-8 or CY-NM-170-301. | P | SAT / UNSAT<br>**STD:** In accordance with N1-ARP-H3-1-8 |
| 5. | Reviews CY-NM-170-301 ODCM 3.6.14 | P | SAT / UNSAT<br>**STD:** In accordance with CY-NM-170-301 |
| 6. | Determines DLCO per 3.6.14.b<br>- With less than the minimum number of radioactive gaseous process and effluent monitoring instrumentation channels functional, take the action shown in Table D 3.6.14-2. Restore the instruments to functional status within 30 days or outline in the next Radioactive Effluent Release Report the cause of the nonfunctionality and how the instruments were or will be restored to functional status. | P | ***PASS / FAIL**<br><br>**STD:** In accordance with CY-NM-170-301 step 3.6.14.b. |
| 7. | Reviews CY-NM-170-301 Table D 3.6.14-2 | P | SAT / UNSAT<br>**STD:** In accordance with CY-NM-170-301 |

| | PERFORMANCE | ACT. CODE<br>P / S / NA | EVALUATOR |
|---|---|---|---|
| 8. | Determines ODCM Table D 3.6.14-2 note (a)(1) applies<br>- With the number of channels functional 1 less than required by the minimum channels functional requirement, effluent releases via this pathway may continue provided:<br>(a) The nonfunctional channel is placed in the tripped condition,<br>OR<br>(b) Vent and Purge valves are closed and administratively controlled,<br>OR<br>(c) Primary containment integrity is not required. | P | ***PASS / FAIL**<br><br>**STD:** In accordance with CY-NM-170-301 step 3.6.14.b. |
| 9. | Reports DLCO actions | P | SAT / UNSAT<br><br>**STD:** Proper communications used. |

| | |
|---|---|
| **TASK STANDARD** | Determine actions per ARP H1-3-8 and CY-NM-107-301 Offsite Does Calculation Manual in response to stack radiation monitor downscale condition. |

| | |
|---|---|
| **STOP TIME** | |

# JPM Handout

| | |
|---|---|
| **INITIAL CONDITIONS** | Given:<br>• The plant is operating at 100% power.<br>• Annunciator H1-3-8, STACK GAS MONITORING TROUBLE alarmed.<br>• Computer Point C061, OGESMS RAD MON STATUS - TRBL alarmed.<br>• OGESMS radiation monitor RN-10A indicates downscale.<br>• RN-10A input to H1-3-8 has been defeated per S-ARP-001.<br>• Annunciator H1-3-8 is now clear. |

| | |
|---|---|
| **INITIATING CUE** | **(Operators Name)**, determine the appropriate actions, compensatory measures, and reporting requirements in response to ARP H1-3-8. Record your findings on the provided attachment. |

**Attachment 1**

**ANSWER KEY**

| Determine the appropriate actions in response to these conditions. | |
|---|---|
| Action | Required By (Document) |
| Request Chemistry implement CY-NM-170-309 OGESMS OFF-NORMAL PROCEDURES.<br><br>Verify the 8 Drywell and Torus Vent and Purge valves CLOSED and administratively tag. | ARP H1-3-8 |
| ODCM Table D 3.6.14-2 note (a)(1) applies - With the number of channels functional 1 less than required by the minimum channels functional requirement, effluent releases via this pathway may continue provided:<br><br>(a) The nonfunctional channel is placed in the tripped condition,<br><br>OR<br><br>(b) Vent and Purge valves are closed and administratively controlled,<br><br>OR<br><br>(c) Primary containment integrity is not required. | ODCM Table D 3.6.14-2 |
| DLCO 3.6.14.b requires - Restore the instruments to functional status within 30 days or outline in the next Radioactive Effluent Release Report the cause of the non-functionality and how the instruments were or will be restored to functional status. | ODCM DLCO 3.6.14.b |

**Attachment 2**

**JPM Scorecard for Applicant Use**

| Determine the appropriate actions in response to these conditions. | |
|---|---|
| Action | Required By (Document) |
| | |

Training Id: **2023 NRC NMP1 Simulator JPM S-A** Revision: **0.0**

Title: **Shift Reactor Building Operating Exhaust and Supply Fans (Alternate Path)**

**Approvals:**

| | Signature / Printed Name | Date |
|---|---|---|
| Developed By | Mike Dougherty | 02/08/24 |
| Validated By | Tim Heagerty | 03/29/24 |
| Facility Reviewer | Dave Mason | |

Approximate Duration: 15 minutes

**Documentation of Performance:**

Performer: _____

Evaluator: _____

Start Time: _____ Stop Time: _____ Completion Time _____

Grade: **Pass / Fail**

Comments: _____

Evaluators Signature: _____ Date: _____

## References

1. N1-OP-10, Reactor Building Heating, Cooling, and Ventilating System
2. NUREG 1123 K/A 288000 A4.01, (3.1/2.9)

Instructor Information

## A. JPM Information

1. Description

   a. This JPM tests the operator's ability to operate the Reactor Building Ventilation System.

   b. This JPM is considered alternate path, because one of the newly started fans displays abnormal operating parameters which should prompt the operator to restore reactor building ventilation back to a functioning lineup.

   c. Critical steps are annotated in the Evaluator standard column with a bolded ***PASS/FAIL**.

2. Task Information:

   a. N1-288001-01002

   b. K/A 288000 A4.01, (3.2)

3. Evaluation / Task Criteria

| | |
|---|---|
| Evaluation Method | Perform |
| Evaluation Location | Simulator |
| Time Critical Task | No |
| Alternate Path | Yes |
| LOD >1.0 | Yes |

4. Recommended Start Location

   a. Unit 1 Simulator

5. Simulator Setup (if required)

   a. Initialize simulator to IC-231 (244)
   b. RX Building Supply and Exhaust Fans #11 in service in slow speed.
   c. RB VENT JPM SETUP
      - Event trigger – set TRG 3 to ZDHVF02T==0
      - Overrides assigned to TRG 3 (Delete overrides when Supply Fan 12 switch is taken to STOP.)
        - DI-5485, INOP 2L11/202-02A POS A, FINAL = On
          *(RB Supply Fan 12 & Inlet Damper)*
        - AO-1257, 2L1/202-02C, Analog Value = 1
          *(React Bldg supply Fan 12 AMP meter)*
        - DO-9525, OFF 202-02 G, DV = OFF
          *(Reactor Bldg Supply Fan 12 Green light OFF)*
        - DO-9526, OFF 202-02 R, DV = ON
          *(Reactor Bldg Supply Fan 12 SLOW Red light ON)*
        - DO-9512, OFF 202-04 G, DV = OFF, DT = 10 sec
          *(Reactor Bldg Supply Fan 12 Inlet Damper Green light OFF)*
        - DO-9513, OFF 202-04 R, DV = ON
          *(Reactor Bldg Supply Fan 12 Inlet Damper Red light ON)*

6. JPM Setup (if required)

   a. No steps need to be marked up.

## B. Read Before Every JPM Performance

1. For the performance of this JPM, I will act as all those you need to talk to. Prior to providing direction to perform this task, I will provide you with the initial conditions and answer any questions. During task performance, I will identify the steps to be simulated, or discuss and provide cues as necessary. (Note, read the next only if conducting a plant JPM). With the exception of accessing panels, no plant equipment will be physically manipulated. Repositioning of devices will be simulated by discussion and acknowledged by my cues.

## C. Read Before Each Evaluated JPM

1. This evaluated JPM is a measure of your ability to perform this task independently. The US has determined that a verifier is not available and that additional verification will not be provided.

| | |
|---|---|
| **INITIAL CONDITIONS** | Given:<br>• Reactor Building Exhaust Fan #11, and Reactor Building Supply Fan # 11 are in service.<br>• All Reactor Building Ventilation Fans in service are operating in slow speed.<br><br>**Evaluator:** *Ask trainee if he/she has any questions after presenting initial conditions* |

| | |
|---|---|
| **INITIATING CUE** | **(Operators Name)**, Shift Reactor Building Ventilation to place Reactor Building Exhaust and Supply Fans #12 in service IAW N1-OP-10 Section F.1.0 and F.2.0. |

| | |
|---|---|
| **START TIME** | |

| | **PERFORMANCE** | **ACT. CODE**<br>P / S / NA | **EVALUATOR** |
|---|---|---|---|
| 1. | Provide repeat back of initiating cue<br><br>**Cue:** Acknowledge repeat back providing correction if necessary | P | SAT / UNSAT<br><br>**STD:** Proper communications used |
| 2. | Obtain a copy of the reference procedure and review / utilize the correct section of the procedure | P | SAT / UNSAT<br><br>**STD:** N1-OP-10 obtained. Precautions & limitations reviewed & section F.1.0 and F.2.0 referenced. |
| 3. | Verify operating RX Building supply and exhaust fans in SLOW.<br><br>**Role Play**: If contacted as Chemistry that Operations are shifting Reactor building Ventilation Fans, acknowledge the report. | P<br>(F.1.2.a) | SAT / UNSAT<br><br>**STD:** Visually observe REACTOR BLDG SUPPLY FAN 11 and REACTOR BLDG EXHAUST FAN 11 fans in SLOW red slow light illuminated. |

| | PERFORMANCE | ACT. CODE<br>P / S / NA | EVALUATOR |
|---|---|---|---|
| 4. | Start REACTOR BLDG EXHAUST FAN 12 on SLOW | P<br>(F.1.2.b) | ***PASS / FAIL**<br><br>**STD:** Rotate REACTOR BLDG EXHAUST FAN 12 control switch CW to the slow position observe red slow light illuminated, green light off. |
| 5. | Confirm damper 202-07, REACTOR BLDG EXHAUST FAN 12 OUTLET DAMPER open. | P<br>(F.1.2.c) | SAT / UNSAT<br><br>**STD:** Observe 202-07 open red light on, green light off. |
| 6. | Stop REACTOR BLDG EXHAUST FAN 11. | P<br>(F.1.2.d) | ***PASS / FAIL**<br><br>**STD:** Rotate REACTOR BLDG EXHAUST FAN 11 control switch CCW to the Off position. |
| 7. | Confirm damper 202-08, REACTOR BLDG EXHAUST FAN 11 OUTLET DAMPER closed. | P<br>(F.1.2.e) | SAT / UNSAT<br><br>**STD:** Observe REACTOR BLDG EXHAUST FAN 11 OUTLET DAMPER closed green light on, red light off. |
| 8. | Confirm normal system flow. | P<br>(F.1.3) | SAT / UNSAT<br><br>**STD:** Observe annunciator L1-2-5 RB VENT EXH FLOW LOW is clear. |
| 9. | Start REACTOR BLDG SUPPLY FAN 12 on SLOW. | P<br>(F.2.2.b) | ***PASS / FAIL**<br><br>**STD:** Rotate REACTOR BLDG SUPPLY FAN 12 control switch CW to the Slow position observe red slow light illuminated, green light off. |
| 10. | Confirm damper FCV 202-04, REACTOR BLDG SUPPLY FAN 12 INLET DAMPER open. | P<br>(F.2.2.c) | SAT / UNSAT / NA<br><br>**STD:** Observe 202-04 open red light on, green light off |

| | PERFORMANCE | ACT. CODE P / S / NA | EVALUATOR |
|---|---|---|---|
| **Alternate Path** | **Note:** Candidate may not notice amps are low and continue in procedure to secure Fan 11.<br>There are 2 success paths.<br><br>1. Low amps identified after Supply Fan 12 start. Candidate performs steps 11,12,13 and 18, with step 13 a critical step.<br>2. Low amps not identified after Supply Fan 12 start. Candidate performs steps 14,15,16,17,18, with steps 16 and 17 as critical step. | | |
| 11. | Identify low amps on REACTOR BLDG SUPPLY FAN 12. | P | SAT / UNSAT / NA<br><br>**STD:** Observe low amps on Fan 12 (5-10 amps) with Fan 11 amps remaining at normal (~30 amps). |
| 12. | Inform SM of low amps on REACTOR BLDG SUPPLY FAN 12.<br><br>**Cue:** As SM, concur and allow Fan 11 to remain in service. | P | SAT / UNSAT / NA<br><br>**STD:** Recommend Fan 11 remains in service and Fan 12 be shutdown. |
| 13. | Secure REACTOR BLDG SUPPLY FAN 12.<br><br>**Note:** Booth operator, delete all overrides when Supply Fan 12 is secured. | P | ***PASS / FAIL / NA**<br><br>**STD:** Rotate REACTOR BLDG SUPPLY FAN 12 control switch CCW to the Off position. |
| 14. | Secure REACTOR BLDG SUPPLY FAN 11<br><br>**Cue:** If directed from L1-2-4 report the following, as the operator in the field:<br>• Heating Unit is tripped<br>• Inlet Filter differential pressure is normal | P<br>(F.2.2.d) | SAT / UNSAT / NA<br><br>**STD:** Rotate REACTOR BLDG SUPPLY FAN 11 control switch CCW to the Off position and observe red slow light off, green light illuminated.<br>Observe annunciator L1-3-4 and L1-2-4 alarm<br>Observe RB dp indication rises indicating high negative dp |

| | PERFORMANCE | ACT. CODE<br>P / S / NA | EVALUATOR |
|---|---|---|---|
| 15. | Report abnormal indication to SM<br><br>**Cue:** As SM, concur and allow Fan 11 to be returned to service and Fan 12 to be shutdown. | | SAT / UNSAT / NA<br><br>**STD:** Recommend returning Fan 11 to service and securing Fan 12 |
| 16. | Start REACTOR BLDG SUPPLY FAN 11 on SLOW.<br><br>*Annunciators L1-3-4 and L1-2-4 clear*<br>*RB dp indication returns to normal* | | ***PASS / FAIL / NA**<br><br>**STD:** Rotate REACTOR BLDG SUPPLY FAN 11 control switch CW to the Slow position observe red slow light illuminated, green light off. |
| 17. | Secure REACTOR BLDG SUPPLY FAN 12<br><br>**Note:** Booth operator, delete all overrides when Supply Fan 12 is secured.<br><br>**Note:** Applicant may reference the OP to secure fan 12 | | ***PASS / FAIL / NA**<br><br>**STD:** Rotate REACTOR BLDG SUPPLY FAN 12 control switch CCW to the Off position and observe red slow light off, green light illuminated. |
| 18. | Notify US/SM that REACTOR BLDG SUPPLY FAN 11 has been returned to service with normal dp and REACTOR BLDG supply FAN 12 is secured.<br><br>**Cue:** Acknowledge report. | P | SAT / UNSAT / NA<br><br>**STD:** Proper communications used. |
| **Evaluator Note:** | **Cue:** *Your task is complete.* | | |

| | |
|---|---|
| **TASK STANDARD** | While placing Supply Fan 12 into service, abnormal operation was identified on Supply Fan 12. Secured Supply Fan 12 and placed Supply Fan 11 in service. Supply Fan 11 and Exhaust Fan 12 are running. |

| | |
|---|---|
| **STOP TIME** | |

# JPM Handout

| | |
|---|---|
| **INITIAL CONDITIONS** | Given:<br>• Reactor Building Exhaust Fan #11, and Reactor Building Supply Fan # 11 are in service. |

| | |
|---|---|
| **INITIATING CUE** | **(Operators Name)**, Shift Reactor Building Ventilation to place Reactor Building Exhaust and Supply Fans #12 in service IAW N1-OP-10 Section F.1.0 and F.2.0. |

Training Id: **2023 NRC NMP1 Simulator JPM S-B** Revision: **0.0**

Title: **Place 11 Shutdown Cooling Loop in Service (Alternate Path)**

**Approvals:**

| | Signature / Printed Name | Date |
|---|---|---|
| Developed By | Mike Dougherty | 02/08/24 |
| Validated By | Tim Heagerty | 03/29/24 |
| Facility Reviewer | Dave Mason | |

Approximate Duration: 20 minutes

**Documentation of Performance:**

Performer: _____

Evaluator: _____

Start Time: _____ Stop Time: _____ Completion Time _____

Grade: **Pass / Fail**

Comments: _____

Evaluators Signature: _____ Date: _____

## References

1. N1-OP-4, Shutdown Cooling System
2. NUREG 1123 K/A 205000, A4.01, (3.9)

Instructor Information

## A. JPM Information

1. Description

   a. This JPM tests the operator's ability to operate the SDC system. The operator will attempt to put SDC system 11 in service.

   b. This JPM is considered alternate path because shortly after a SDC pump is started, it will trip. The operator will be expected to make a recommendation, based on plant conditions, to place 12 SDC loop in service.

   c. Critical steps are annotated in the Evaluator standard column with a bolded ***Pass/Fail**.

2. Task Information:

   a. N1-205000-01002

   b. K/A 205000, A4.01, (3.9)

3. Evaluation / Task Criteria

| | |
|---|---|
| Evaluation Method | Perform |
| Evaluation Location | Simulator |
| Time Critical Task | No |
| Alternate Path | Yes |
| LOD >1.0 | Yes |

4. Recommended Start Location

   a. Unit 1 Simulator

5. Simulator Setup (if required)

   a. Initialize simulator to IC 232
   b. Verify SDC temperature recorder on.
   c. Update 70-49, 70-54, and 70-58 status to “4”.
   d. Verify recirc suction temperature is displayed (computer point A427).
   e. Verify SDC is removed from service (TCVs closed, pumps secured, IVs closed, inlet BVs closed).
   f. Verify SDC Pump 13 in PTL and info/caution tagged
   g. Verify SDC 13 suction valve is info/caution tagged (38-05)
   h. Verify Malfunction SC01C inserted
   i. Verify Malfunction SC01A on TRG 1 with 30 second TD
   j. Verify TRG 1 condition is “ZDSCPSTR(1)==1”
   k. Verify the following remotes are preset:
      - SC01 = close
      - SC02 = installed
      - SC03 = close
   l. Verify remotes SC04A(B)(C) = 100% are preset.

6. JPM Setup (if required)

   a. N1-OP-4, E.3.1 marked complete.

## B. Read Before Every JPM Performance

1. For the performance of this JPM, I will act as all those you need to talk to. Prior to providing direction to perform this task, I will provide you with the initial conditions and answer any questions. During task performance, I will identify the steps to be simulated, or discuss and provide cues as necessary. (Note, read the next only if conducting a plant JPM). With the exception of accessing panels, no plant equipment will be physically manipulated. Repositioning of devices will be simulated by discussion and acknowledged by my cues.

## C. Read Before Each Evaluated JPM

1. This evaluated JPM is a measure of your ability to perform this task independently. The US has determined that a verifier is not available and that additional verification will not be provided.

| | |
|---|---|
| **INITIAL CONDITIONS** | Given:<br>• The Reactor is shutdown with all control rods inserted.<br>• Reactor pressure is less than 120 psig.<br>• N1-OP-4 has been completed through step E.3.1 for placing Shutdown Cooling in service.<br>• Shutdown Cooling has been filled and vented within the last 24 hours and does not require further filling or venting.<br>• Shutdown Cooling IV fuses/breakers have been installed.<br>• Cooling Water has been placed in service per section E.2.0<br>• SDC pump 13 is OOS for maintenance.<br>• Reactor water level is being monitored by another operator.<br><br>**Evaluator:** *Ask trainee if he/she has any questions after presenting initial conditions* |

| | |
|---|---|
| **INITIATING CUE** | **(Operators Name)**, place Shutdown Cooling loop 11 in service per N1-OP-4 starting at step E.3.2. |

| | |
|---|---|
| **START TIME** | |

| | **PERFORMANCE** | **ACT. CODE** P / S / NA | **EVALUATOR** |
|---|---|---|---|
| 1. | Provide repeat back of initiating cue<br><br>**Cue:** Acknowledge repeat back providing correction if necessary | P | SAT / UNSAT<br><br>**STD:** Proper communications used |
| 2. | Obtain a copy of the reference procedure and review / utilize the correct section of the procedure | P | SAT / UNSAT<br><br>**STD:** N1-OP-4 obtained, precautions & limitations reviewed, & section E.3.0 referenced |
| 3. | Open the following valves: (Step 3.2) | | |

| | PERFORMANCE | ACT. CODE<br>P / S / NA | EVALUATOR |
|---|---|---|---|
| 3a. | 38-03, SDC VALVE 11 PUMP SUCTION | P<br>(E.3.2) | ***PASS / FAIL**<br><br>**STD:** Opens 38-03 by rotating control switch CW to OPEN |
| 3b. | 38-04, SDC VALVE 12 PUMP SUCTION | P<br>(E.3.2) | ***PASS / FAIL**<br><br>**STD:** Opens 38-04 by rotating control switch CW to OPEN |
| 3c. | 38-05, SDC VALVE 13 PUMP SUCTION<br><br>**Note:** Candidate may request an SQR one-time procedure change be processed before proceeding.<br><br>**Cue:** If SQR one-time procedure change is requested, inform the Operator that the step can be left open and to proceed. | P<br>(E.3.2) | SAT / UNSAT<br><br>**STD:** Recognizes 38-05 is tagged closed |
| 4. | Fill AND vent the SDC System by performing the following: | P<br>(E.3.3) | SAT / UNSAT<br><br>**STD:** Determines fill and vent not required due to initial conditions |
| 5 | If RCS pressure is less than 120 psig or RCS pressure will be less than 120 psig with one hour Then mark step E.3.5 N/A<br><br>***Note**: The operator should mark E.3.5 as N/A due to RCS pressure being less than 120 psig as directed by step E.3.4.* | P<br>(E.3.4) | SAT / UNSAT<br><br>**STD:** Determines RCS pressure is less than 120 psig due to initial conditions or simulator setup |

| | PERFORMANCE | ACT. CODE<br>P / S / NA | EVALUATOR |
|---|---|---|---|
| 6. | Verify all rods inserted UNLESS directed by EOPs | P<br>(E.3.6) | SAT / UNSAT<br><br>**STD:** Determines all rods are inserted per initial conditions or simulator setup |
| 7. | IF SDC IV fuses AND breakers have NOT been installed, THEN perform the following: | P<br>(E.3.7) | SAT / UNSAT<br><br>**STD:** Determines SDC IV fuses and breakers have been installed per initial conditions or simulator setup |
| 8. | Open IV-38-02, SDC SYSTEM IN IV 12 (OUTSIDE) | P<br>(E.3.8) | ***PASS / FAIL**<br><br>**STD:** Opens 38-02 by rotating control switch CW to OPEN |
| 9. | WHEN Reactor pressure is less than 120 psig, un-isolate Shutdown Cooling System as follows: | P<br>(E.3.9) | SAT / UNSAT<br><br>**STD:** Determines Reactor pressure is less than 120 psig per initial conditions or simulator setup |
| 9a | Monitor Reactor vessel level | P<br>(E.3.9.a) | SAT / UNSAT<br><br>**STD:** Determines another operator is monitoring Reactor water level per initial conditions |
| 9b | IF Reactor vessel level begins to go down WHILE un-isolating Shutdown Cooling System, THEN isolate Shutdown Cooling System AND determine cause of reduction in Reactor Vessel level | P<br>(E.3.9.b) | SAT / UNSAT<br><br>**STD:** Determines Reactor water level is stable in subsequent steps |

| | PERFORMANCE | ACT. CODE<br>P / S / NA | EVALUATOR |
|---|---|---|---|
| 9c | Open 38-13, SDC SYSTEM OUT IV 1 (INSIDE) | P<br>(E.3.9.c) | ***PASS / FAIL**<br><br>**STD:** Opens 38-13 by rotating control switch CW to OPEN |
| 9d | Open 38-01, SDC SYTEM IN IV 11 (INSIDE) | P<br>(E.3.9.d) | ***PASS / FAIL**<br><br>**STD:** Opens 38-01 by rotating control switch CW to OPEN |
| 10. | If system was NOT in-service within last 24 hours OR system was NOT filled AND vented with 24 hours, then vent shutdown cooling pumps using the following valves: | P<br>(E.3.10) | SAT / UNSAT<br><br>**STD:** Determines venting not required due to initial conditions |
| 11. | If system was NOT in-service within last 24 hours OR system was NOT filled AND vented with 24 hours, then vent heat exchangers using the following valves: | P<br>(E.3.11) | SAT / UNSAT<br><br>**STD:** Determines venting not required due to initial conditions |
| 12. | Verify open the following valves:<br><br>• 38-134, SDC PUMP RECIRC VALVE 11<br><br>• 38-131, SDC PUMP RECIRC VALVE 12<br><br>• 38-128, SDC PUMP RECIRC VALVE 13<br><br>**Note:** Candidate may disregard 38-128 because the associated pump is tagged out. | P<br>(E.3.12) | SAT / UNSAT<br><br>**STD:** Observes green light off, red light on for all |
| 13. | Start 11 SDC Pump as follows:<br>(Step E.3.13) | **Note:** Pump Seal was NOT discovered de-staged (E.3.13.b) | |

| | PERFORMANCE | ACT. CODE<br>P / S / NA | EVALUATOR |
|---|---|---|---|
| 13a | Verify Cooling Water in Service per N1-OP-4 Section E | P<br>(E.3.13.a) | SAT / UNSAT<br><br>**STD:** Determines this is already complete per the initial conditions. |
| 13b | If respective pump seal was discovered de-staged then prior to placing SDC Loop in service, fill and vent associated SDC Loop per Section H.<br><br>**Role Play**: if asked, the pump seal was **NOT** discovered de-staged. | P<br>(E.3.13.b) | SAT / UNSAT<br><br>**STD:** Determines this is N/A based on information provided by the evaluator. |
| 13b | Verify SD COOLING TCV 11 manual controller selected for zero output demand | P<br>(E.3.13.c.1)) | SAT / UNSAT<br><br>**STD:** Observes SD COOLING TCV 11 manual controller selected for zero output demand |
| 13c | Place 11 SDC Pump Control Switch in START UNTIL 11 SDC PUMP running light is lit | P<br>(E.3.13.c.2)) | ***PASS / FAIL**<br><br>**STD:** Starts SDC pump 11 by rotating control switch CW to START |
| **Alternate Path** | 11 SDC pump will fail, initiating the alternate path. | | |
| 13d | Open 38-09, SD COOLING TCV 11, approximately 10%:<br><br>**NOTE:** 1 minute after 11 SDC pump in service, verify tripped 11 SDC pump (event trigger). Alarm K3-1-1, SDC Pump 11 trip, comes in on the pump trip. The alarm response procedure gives direction to start another pump. | P<br>(E.3.13.c.3)) | ***PASS / FAIL / NA**<br><br>**STD:** Opens 38-09 by rotating knurled knob CW<br><br>**Note:** Step may be NA if pump trips prior to executing. |

| | PERFORMANCE | ACT. CODE<br>P / S / NA | EVALUATOR |
|---|---|---|---|
| | **Cue:** If applicant does not make a recommendation to place another SDC pump in service, ask "What would you recommend?".<br><br>**Role Play:** If candidate recommends placing **SDC Loop 12** back in service, direct them to place SDC Loop 12 back into service.<br><br>**Role Play:** If candidate asks about normalizing **SDC loop 11**, tell them to leave SDC loop 11 as is. | | |
| 14. | Start 12 SDC Pump as follows:<br>(E.3.13.d) | **Note:** Pump Seal was NOT discovered de-staged (E.3.13.b) | |
| 14a | Verify SD COOLING TCV 12 manual controller selected for zero output demand | P<br>(E.3.13.d.1)) | SAT / UNSAT<br><br>**STD:** Observes SD COOLING TCV 12 manual controller selected for zero output demand |
| 14b | Place 12 SDC Pump Control Switch in START UNTIL 12 SDC PUMP running light is lit<br>(Step 3.11.2.o) | P<br>(E.3.13.d.2)) | ***PASS / FAIL**<br><br>**STD:** Starts SDC pump 12 by rotating control switch CW to START |
| 14c. | Open 38-10, SD COOLING TCV 12, approximately 10%d SDC Loop: | P<br>(E.3.13.d.3)) | ***PASS / FAIL**<br><br>**STD:** Opens 38-10 by rotating knurled knob CW |
| 15. | If reactor water flashing occurs in SDC System, reduce reactor water flow via selected SDC cooling TCV and maximize RBCLC cooling water flow.<br><br>**Cue:** No flashing observed. | P<br>(E.3.13.f) | SAT / UNSAT<br><br>**STD:** Pump amp indicator observed as steady. |

| | PERFORMANCE | ACT. CODE<br>P / S / NA | EVALUATOR |
|---|---|---|---|
| 16. | Adjust 38-10, SDC COOLING TCV 12 for gradual warmup of the system and to achieve desired cooldown rate of less than or equal to $75^0$F/hr.<br><br>**NOTE:** Operator will monitor temperatures on panel recorder.<br><br>**NOTE:** Operator may choose not to make any further adjustments based on the rate at which the loop is warming up with 38-10 10% open.<br><br>**Cue:** Tell operator no further adjustments in the system are required. | P<br>(E.3.13.g) | SAT / UNSAT<br><br>**STD:** Rotate knob on controller 38-10 CW or CCW as required for a gradual warmup as indicated on 38-136B. |
| 18. | Inform SM SDC Loop 12 in service<br><br>**Cue:** Acknowledge report | P | SAT / UNSAT |
| **Evaluator Note:** | **Cue:** *Your task is complete.* | | |

| | |
|---|---|
| **TASK STANDARD** | While placing SDC loop into service the applicant recognizes failure of Shutdown Cooling Pump 11. The applicant secures Shutdown Cooling Pump 11 and placed Shutdown Cooling Loop 12 in service. |

| | |
|---|---|
| **STOP TIME** | |

# JPM Handout

| | |
|---|---|
| **INITIAL CONDITIONS** | Given:<br>• The Reactor is shutdown with all rods in.<br>• Reactor pressure is less than 120 psig.<br>• N1-OP-4 has been completed through step E.3.1 for placing Shutdown Cooling in service.<br>• Shutdown Cooling has been filled and vented within the last 24 hours and does not require further filling or venting.<br>• Shutdown Cooling IV fuses/breakers have been installed.<br>• Cooling Water has been placed in service per section E.2.0<br>• SDC pump 13 is OOS for maintenance.<br>• Reactor water level is being monitored by another operator. |

| | |
|---|---|
| **INITIATING CUE** | **(Operators Name)**, place Shutdown Cooling loop 11 in service per N1-OP-4 starting at step E.3.2. |

Constellation

Training Id: **2023 NRC NMP1 Simulator JPM S-C** Revision: **0.0**

Title: **Secure a Reactor Recirculation Pump, Pump Trips (Alternate Path)**

**Approvals:**

| | Signature / Printed Name | Date |
|---|---|---|
| Developed By | Mike Dougherty | 02/08/24 |
| Validated By | Tim Heagerty | 03/29/24 |
| Facility Reviewer | Dave Mason | |

Approximate Duration: 20 minutes

**Documentation of Performance:**

Performer: _____

Evaluator: _____

Start Time: _____ Stop Time: _____ Completion Time _____

Grade: **Pass / Fail**

Comments: _____

Evaluators Signature: _____ Date: _____

## References

1. N1-OP-1, Nuclear Steam Supply System
2. NUREG 1123 K/A 202001 A4.01, (4.0)

Instructor Information

## A. JPM Information

1. Description

   a. This JPM tests the operator's ability to operate controls associated with the Reactor Recirculation System.

   b. This JPM is considered alternate path because when the operator reduces flow on the pump to be secured, the pump will trip, requiring the operator to respond in accordance with the SOP.

   c. Critical steps are annotated in the Evaluator standard column with a bolded **Pass/Fail**.

2. Task Information:

   a. N1-202001-01002

   b. K/A 202001 A4.01, (4.0)

3. Evaluation / Task Criteria

| | |
|---|---|
| Evaluation Method | Perform |
| Evaluation Location | Simulator |
| Time Critical Task | No |
| Alternate Path | Yes |
| LOD >1.0 | Yes |

4. Recommended Start Location

   a. Unit 1 Simulator

5. Simulator Setup (if required)
    a. The reactor is in a power operating condition at approximately 90% Reactor power
    b. Initialize simulator to IC 233
    c. Verify the following malfunctions are preset:
        - RR01A, RR Pump 11 27x Short Drive Bkr Trip, **TRG 1**
        - **TRG 1** event trigger action set to "hzarrfi(1)<0.44" (Causes drive breaker to trip when individual loop flow reaches $8x10^6$ lbm/hr)

6. JPM Setup (if required)
    a. Verify Reactor power is at approximately 90%
    b. Verify sufficient copies are available of N1-OP-1 P&L's and section H.1

## B. Read Before Every JPM Performance

1. For the performance of this JPM, I will act as all those you need to talk to. Prior to providing direction to perform this task, I will provide you with the initial conditions and answer any questions. During task performance, I will identify the steps to be simulated, or discuss and provide cues as necessary. (Note, read the next only if conducting a plant JPM). With the exception of accessing panels, no plant equipment will be physically manipulated. Repositioning of devices will be simulated by discussion and acknowledged by my cues.

## C. Read Before Each Evaluated JPM

1. This evaluated JPM is a measure of your ability to perform this task independently. The US has determined that a verifier is not available and that additional verification will not be provided.

| | |
|---|---|
| **INITIAL CONDITIONS** | Given:<br>• The reactor is operating at 90% power<br>• Reactor Recirculation Pump 11 is being removed for maintenance<br>• Reactor power will be allowed to lower for the evolution<br>• Recirculation pump 11 will remain idle and unisolated<br>• Another operator is monitoring computer points during this evolution<br><br>**Evaluator:** *Ask trainee if he/she has any questions after presenting initial conditions* |

| | |
|---|---|
| **INITIATING CUE** | **(Operators Name)**, Remove Recirc Pump 11 from service in accordance with N1-OP-1, section G.2.0 |

| | |
|---|---|
| **START TIME** | |

| | PERFORMANCE | ACT. CODE<br>P / S / NA | EVALUATOR |
|---|---|---|---|
| 1. | Provide repeat back of initiating cue<br><br>**Cue:** Acknowledge repeat back providing correction if necessary | P | SAT / UNSAT<br>**STD:** Proper communications used |
| 2. | Obtain a copy of the reference procedure and review / utilize the correct section of the procedure | P | SAT / UNSAT<br>**STD:** N1-OP-1 obtained. |
| 3. | IF Recirc Pump 11 is in Local Lock, THEN continue at Step G.2.4 | P<br>(G.2.1) | SAT / UNSAT<br>**STD:** Determines Recirc Pump 11 is not in Local Lock. |
| 4. | Place RRECIRC PUMP 11 SPEED CONTROL in BAL and null out<br>Deviation Meter (top meter) | P<br>(G.2.2) | SAT / UNSAT<br>**STD:** Verifies RRECIRC PUMP 11 SPEED CONTROL in BAL and nulls out Deviation Meter (top meter) |

| | PERFORMANCE | ACT. CODE<br>P / S / NA | EVALUATOR |
|---|---|---|---|
| 5. | Place RRECIRC PUMP 11 SPEED CONTROL AUTO/BAL/MAN switch to MAN | P<br>(G.2.3) | ***PASS / FAIL**<br>**STD:** Rotates RRECIRC PUMP 11 SPEED CONTROL AUTO/BAL/MAN switch CW to MAN |
| 6. | Verify open REACTOR R PUMP 11 BYPASS VALVE<br><br>**Note:** Step G.2.5 is N/A per initial conditions | P<br>(G.2.4) | SAT / UNSAT<br>**STD:** Verifies open REACTOR R PUMP 11 BYPASS VALVE by observing red light on, green light off |
| 7. | IF present power level is NOT to be maintained, THEN reduce flow on Reactor Recirc Pump 11 to 6 to 8 x $10^6$ lbm/hr | P<br>(G.2.6) | ***PASS / FAIL**<br>**STD:** Slowly reduces Recirculation pump 11 flow to 6 to 8x$10^6$ lbm/hr by rotating knob CCW |
| **Alternate Path** | The alternate path starts when the operator is lowering pump flow. The recirc pump will trip requiring entry into N1-SOP-1.3, Recirc Pump Trip at Power. | | |
| 8. | Recognize Recirc Pump 11 trip and required entry in to N1-SOP-1.3, RECIRC PUMP TRIP AT POWER. | P | SAT / UNSAT<br>**STD:** Enters N1-SOP-1.3 |
| 9. | Verify proximity to restricted zone using P/F map for remaining loop operation | P | SAT / UNSAT<br>**STD:** Using the four loop power to flow map, confirms the restricted zone has not been entered. |

| | PERFORMANCE | ACT. CODE<br>P / S / NA | EVALUATOR |
|---|---|---|---|
| 10. | Verify ALL operating Recirculation Pumps are within the following limits:<br><br>Generator MW: 0.790 MW<br>Generator Amps: 240 A<br>RRP Flow: 16.8 x 106 lbm/hr continuous<br>Generator Frequency: 11.5 Hz to 56 Hz | P | SAT / UNSAT<br><br>**STD:** Verifies recirc pumps 12-15 are operating within required limits. |
| 11. | Notify SM that:<br>APRMs are inoperable.<br>8 hr ENS report is required | P | SAT / UNSAT<br><br>**STD:** Makes required notifications to SM. |
| 12. | Close RECIRC PUMP 11 DISCHARGE VALVE<br><br>**Note:** Power is expected to rise ~ 3% | P | ***PASS / FAIL**<br><br>**STD:** Closes REACTOR R PUMP 11 DISCHARGE VALVE by rotating control switch CCW and holding until green light is on, red light is off, observes lowering loop flow. |
| 13. | **Override Statement:**<br><br>IF RECIRC PUMP 11 DISCHARGE VALVE is closed, THEN Hold open for 2-3 seconds RECIRC PUMP 11 DISCHARGE VALVE. | P | SAT / UNSAT<br><br>**STD:** Rotates CW and holds in OPEN position for 2 to 3 seconds, REACTOR R PUMP 11 DISCHARGE VALVE, observes red light on, green light on. |
| 14. | Notify SM that APRMs are operable | P | SAT / UNSAT<br><br>**STD:** Notifies SM that APRMs are operable. |

| | PERFORMANCE | ACT. CODE<br>P / S / NA | EVALUATOR |
|---|---|---|---|
| 15. | Green Flag RECIRC PUMP 11 SB switch | P | SAT / UNSAT<br><br>**STD:** Green Flags RECIRC PUMP 11 SB switch by rotating CCW |
| 16. | Place RECIRC PUMP 11 SPEED CONTROL AUTO/BAL/MAN switch to MAN<br><br>**Note:** This step was performed earlier in the JPM | N/A | SAT / UNSAT<br><br>**STD:** Verifies RECIRC PUMP 11 SPEED CONTROL AUTO/BAL/MAN switch in MAN |
| **Evaluator Note:** | **Cue:** *Your task is complete.* | | |

| | |
|---|---|
| **TASK STANDARD** | *While attempting to shutdown Recirc Pump 11, the pump tripped as Recirc flow was being lowered. The applicant entered and executed N1-SOP-1.3, RECIRC PUMP TRIP AT POWER. Actions completed for a Recirc Pump trip at power with its associated discharge valve closed.* |

| | |
|---|---|
| **STOP TIME** | |

# JPM Handout

| | |
|---|---|
| **INITIAL CONDITIONS** | Given:<br>• The reactor is operating at 90% power<br>• Reactor Recirculation Pump 11 is being removed for maintenance<br>• Reactor power will be allowed to lower for the evolution<br>• Recirculation pump 11 will remain idle and unisolated<br>• Another operator is monitoring computer points during this evolution |

| | |
|---|---|
| **INITIATING CUE** | **(Operators Name)**, Remove Recirc Pump 11 from service in accordance with N1-OP-1, section G.2.0 |

Constellation

Training Id: **2024 NRC NMP1 Sim JPM S-D** Revision: **0.0**

Title: **Respond to a Loss of Service Water (PRA) (Alternate Path)**

**Approvals:**

| | Signature / Printed Name | Date |
|---|---|---|
| Developed By | Mike Dougherty | 02/09/24 |
| Validated By | Tim Heagerty | 03/29/24 |
| Facility Reviewer | Dave Mason | |

Approximate Duration: 10 minutes

**Documentation of Performance:**

Performer: _____

Evaluator: _____

Start Time: _____ Stop Time: _____ Completion Time _____

Grade: **Pass / Fail**

Comments: _____

Evaluators Signature: _____ Date: _____

## References

1. N1-SOP-18.1, Loss of Service Water
2. NUREG 1123 K/A 295018, AA1.01, (3.3)

Instructor Information

## A. JPM Information

1. Description

   a. This JPM tests the operator's ability to safely operate control room systems during a loss of Service Water.

   b. This JPM is considered alternate path because after the standby service water pump is started, it trips, requiring override steps in the SOP to be executed.

   c. Critical steps are annotated in the Evaluator standard column with a bolded **PASS/FAIL**.

2. Task Information:

   a. N1-SOP-18.1-01000, Respond to Service Water Failure (PRA)

   b. K/A 295018, AA1.01, (3.3)

3. Evaluation / Task Criteria

| | |
|---|---|
| Evaluation Method | Perform |
| Evaluation Location | Simulator |
| Time Critical Task | No |
| Alternate Path | Yes |
| LOD >1.0 | Yes |

4. Recommended Start Location

   a. Unit 1 Simulator

5. Simulator Setup (if required)
   a. Initialize Simulator to IC-234
   b. Verify Service Water pump 11 is red flagged and Service Water pump 12 is green flagged
   c. Verify malfunction CW02B is on trigger 1 with a 10 second time delay
   d. Verify trigger 1 command is set to zdcwpstr(4)==1 (Service Water pump 12 control switch in start)
   e. Verify malfunction CW02A is active

6. JPM Setup (if required)

   a. No steps need to be marked up.

## B. Read Before Every JPM Performance

1. For the performance of this JPM, I will act as all those you need to talk to. Prior to providing direction to perform this task, I will provide you with the initial conditions and answer any questions. During task performance, I will identify the steps to be simulated, or discuss and provide cues as necessary. (Note, read the next only if conducting a plant JPM). With the exception of accessing panels, no plant equipment will be physically manipulated. Repositioning of devices will be simulated by discussion and acknowledged by my cues.

## C. Read Before Each Evaluated JPM

1. This evaluated JPM is a measure of your ability to perform this task independently. The US has determined that a verifier is not available and that additional verification will not be provided.

| | |
|---|---|
| **INITIAL CONDITIONS** | Given:<br>• The plant is operating at approximately 100% power.<br>• Service Water pump 11 has tripped on overcurrent.<br><br>**Evaluator:** *Ask trainee if he/she has any questions after presenting initial conditions* |

| | |
|---|---|
| **INITIATING CUE** | **(Operators Name)**, enter and execute N1-SOP-18.1 for Service Water Failure. |

| | |
|---|---|
| **START TIME** | |

| | **PERFORMANCE** | **ACT. CODE** P / S / NA | **EVALUATOR** |
|---|---|---|---|
| 1. | Provide repeat back of initiating cue<br><br>**Cue:** Acknowledge repeat back providing correction if necessary | P | SAT / UNSAT<br><br>**STD:** Proper communications used |
| 2. | Obtain a copy of the reference procedure and review / utilize the correct section of the procedure | P | SAT / UNSAT<br><br>**STD:** N1-SOP-18.1 obtained. |
| 3. | Start standby Service Water pump by placing control switch on Panel H to START<br><br>**Note:** Service Water pump 12 will trip after a 10 second time delay, requiring an alternate path to execute the override for loss of both Service Water pumps | P | ***PASS / FAIL**<br><br>**STD:** Rotates Service Water pump 12 control switch CW to start, observes red light on, green light off, proper pump amps |
| Alternate Path | The alternate path starts when Service Water pump 12 trips approximately 10 seconds after the applicant starts the pump. Service Pump 12 tripping will require the applicant to perform the override for “If Neither Service Water Pump can be started”. | | |

| | PERFORMANCE | ACT. CODE<br>P / S / NA | EVALUATOR |
|---|---|---|---|
| 4. | Identifies trip of Service Water pump 12 | P | SAT / UNSAT<br><br>**STD:** Observes Service Water pump 12 green light on, red light off, pump amps drop to zero, annunciator H1-2-2 alarms |
| 5. | If neither Service Water pump can be started OR Service Water Header pressure can NOT be restored to greater than 60 psig, then performs the following steps: | P | SAT / UNSAT<br><br>**STD:** Enters override of N1-SOP-18.1 |
| 6. | Start Emergency Service Water pumps | P | ***PASS / FAIL**<br><br>**STD:** Rotates ESW pump 11 control switch CW to start, observes red light on, green light off, proper pump amps<br><br>***PASS / FAIL**<br><br>**STD:** Rotates ESW pump 12 control switch CW to start, observes red light on, green light off, proper pump amps |
| 7. | SCRAM the Reactor AND concurrently execute N1-SOP-1<br><br>**Cue:** **When** the applicant has completed actions to scram the reactor, inform them that "*another operator will perform SOP-1 actions. Continue your actions in SOP-18.1.*" | P | ***PASS / FAIL**<br><br>**STD:** Depresses the Reactor Trip 11 and 12 pushbuttons AND rotates the Reactor Mode Switch CW to Shutdown |

| | PERFORMANCE | ACT. CODE<br>P / S / NA | EVALUATOR |
|---|---|---|---|
| 8. | Initiate Emergency Condensers<br><br>**Cue:** Once the applicant has rotated the Emergency Condenser switches to the OPEN position, inform them that "*another operator will maintain Reactor pressure per N1-EOP-RPV (1-2).*" | P | ***PASS / FAIL**<br><br>**STD:** Rotates 39-05 control switch CW to open and/or rotates 39-06 control switch CW to open |
| 9. | Close MSIVs | | |
| 9a | | P | ***PASS / FAIL**<br><br>**STD:** Rotates 01-01 control switch CCW to close |
| 9b | | P | ***PASS / FAIL**<br><br>**STD:** Rotates 01-02 control switch CCW to close |
| 9c | | P | ***PASS / FAIL**<br><br>**STD:** Rotates 01-03 control switch CCW to close |
| 9d | | P | ***PASS / FAIL**<br><br>**STD:** Rotates 01-04 control switch CCW to close |
| 10. | Trip all Rx Recirculation Pumps<br><br>**Cue:** When the applicant has tripped all Reactor Recirculation pumps inform them that "*another operator will monitor RBCLC and TBCLC temperatures.*" | | |

| | PERFORMANCE | ACT. CODE<br>P / S / NA | EVALUATOR |
|---|---|---|---|
| 10a | | P | ***PASS / FAIL**<br>**STD:** Rotates RRP 11 control switch CCW to stop |
| 10b | | P | ***PASS / FAIL**<br>**STD:** Rotates RRP 12 control switch CCW to stop |
| 10c | | P | ***PASS / FAIL**<br>**STD:** Rotates RRP 13 control switch CCW to stop |
| 10d | | P | ***PASS / FAIL**<br>**STD:** Rotates RRP 14 control switch CCW to stop |
| 10e | | P | ***PASS / FAIL**<br>**STD:** Rotates RRP 15 control switch CCW to stop |
| **Evaluator Note:** | **Cue:** *Your task is complete.* | | |

| | |
|---|---|
| **TASK STANDARD** | The applicant attempted to place the standby Service Water pump 12 in service followed by the standby pump starting and then tripping 10 seconds later. The applicant entered and completed N1-SOP-18.1 actions through the override step for a sequential loss of Service Water pumps. |

| | |
|---|---|
| **STOP TIME** | |

# JPM Handout

| | |
|---|---|
| **INITIAL CONDITIONS** | Given:<br>• The plant is operating at approximately 100% power.<br>• Service Water pump 11 has tripped on overcurrent. |

| | |
|---|---|
| **INITIATING CUE** | **(Operators Name)**, enter and execute N1-SOP-18.1 for Service Water Failure. |

Training Id: **2023 NMP1 NRC Simulator JPM S-E** Revision: **0.0**

Title: **Transfer to the MPR at Power**

**Approvals:**

| | Signature / Printed Name | Date |
|---|---|---|
| Developed By | Mike Dougherty | 02/09/24 |
| Validated By | Tim Heagerty | 03/29/24 |
| Facility Reviewer | Dave Mason | |

Approximate Duration: 15 minutes

**Documentation of Performance:**

Performer: _____

Evaluator: _____

Start Time: _____ Stop Time: _____ Completion Time _____

Grade: **Pass / Fail**

Comments: _____

Evaluators Signature: _____ Date: _____

## References

1. N1-OP-31, Tandem Compound Reheat Turbine
2. NUREG 1123 K/A 241000, A1.01, (4.4)

Instructor Information

## A. JPM Information

1. Description

   a. This JPM tests the operator's ability to manually operate reactor pressure control devices in the control room.

   b. This JPM is NOT considered alternate path.

   c. Critical steps are annotated in the Evaluator standard column with a bolded **Pass/Fail**.

2. Task Information:

   a. N1-248000-01002, Transfer To/From MPR at Power

   b. K/A 241000, A1.01, (4.4)

3. Evaluation / Task Criteria

| | |
|---|---|
| Evaluation Method | Perform |
| Evaluation Location | Simulator |
| Time Critical Task | No |
| Alternate Path | No |
| LOD >1.0 | Yes |

4. Recommended Start Location

   a. Unit 1 Simulator

5. Simulator Setup (if required)

    a. Initialize simulator to IC 235 (244)
    b. Verify the EPR is in service
    c. Verify MPR Servo Stroke is 8-12% lower than the EPR Servo Stroke

6. JPM Setup (if required)

    a. Provide sufficient copies of N1-OP-31 P&Ls and section F.2.

## B. Read Before Every JPM Performance

1. For the performance of this JPM, I will act as all those you need to talk to. Prior to providing direction to perform this task, I will provide you with the initial conditions and answer any questions. During task performance, I will identify the steps to be simulated, or discuss and provide cues as necessary. (Note, read the next only if conducting a plant JPM). With the exception of accessing panels, no plant equipment will be physically manipulated. Repositioning of devices will be simulated by discussion and acknowledged by my cues.

## C. Read Before Each Evaluated JPM

1. This evaluated JPM is a measure of your ability to perform this task independently. The CRS has determined that a verifier is not available and that additional verification will not be provided.

| | |
|---|---|
| **INITIAL CONDITIONS** | Given:<br>• The reactor is operating at approximately 100% power<br>• EPR is in service<br>• Pressure control will be transferred to the MPR<br>• Thermal limits have been verified SAT per N1-OP-31 section H, Operation with One Pressure Regulator Inoperable<br>• An Equipment Operator is standing by at the Front Standard for support<br><br>**Evaluator:** *Ask trainee if he/she has any questions after presenting initial conditions* |

| | |
|---|---|
| **INITIATING CUE** | **(Operators Name)**, transfer Reactor pressure control from the EPR to the MPR in accordance with N1-OP-31, section F.2. |

| | |
|---|---|
| **START TIME** | |

| | **PERFORMANCE** | **ACT. CODE**<br>P / S / NA | **EVALUATOR** |
|---|---|---|---|
| 1. | Provide repeat back of initiating cue<br><br>**Cue:** Acknowledge repeat back providing correction if necessary | P | SAT / UNSAT<br><br>**STD:** Proper communications used |
| 2. | Obtain a copy of the reference procedure and review / utilize the correct section of the procedure | P | SAT / UNSAT<br><br>**STD:** N1-OP-31 obtained, precautions and limitation reviewed, section F.2.0 referenced |

| | PERFORMANCE | ACT. CODE<br>P / S / NA | EVALUATOR |
|---|---|---|---|
| 3. | Verify Thermal Limits satisfied per Section H. Operation with one Pressure Regulator Inoperable<br><br>**Role Play**: If asked report that Thermal Limits have been verified SAT for use of the EPR as a back up regulator.<br><br>**Note**: Already verified per initial conditions provided | P<br>(F.2.1) | SAT / UNSAT<br><br>**STD:** Applicant should mark this step as complete based on initial conditions provided. |
| 4. | Record Reactor pressure (from ID77 "Reactor Pressure Suppressed Range" on F Panel) | P<br>(F.2.2) | SAT / UNSAT<br><br>**STD:** Records Reactor pressure as read on F Panel |
| 5. | Record MPR Bushing Rotation Speed.<br><br>**Role Play**: When asked report that the MPR Bushing Speed is 31 rpm | P<br>(F.2.3) | SAT / UNSAT<br><br>**STD:** Applicant should record MPR bushing speed with information provided from the Role Play. |
| 6. | During transfer from EPR to MPR, IF pressure oscillations greater than 2 psi OR greater than plus or minus 3% APRM oscillations are observed, stop the transfer AND return control to the EPR per subsection F.3.0 | P<br>(F.2.4) | SAT / UNSAT<br><br>**STD:** Monitors Reactor pressure and power during transfer, determines no oscillations occur. |
| 7. | Slowly lower MPR setpoint, move-and-wait manner, until MPR servo position begins to move upscale (E panel) | P<br>(F.2.5) | ***PASS / FAIL**<br><br>**STD:** Rotates MPR setpoint control switch CW to lower, observes MPR servo stroke rise |

| | PERFORMANCE | ACT. CODE<br>P / S / NA | EVALUATOR |
|---|---|---|---|
| 8. | WHEN MPR servo position approaches EPR servo position, observe MPR assuming control by response of steam pressure, MPR control light and Annunciator A2-4-4 MPR in control | P<br>(F.2.6) | SAT / UNSAT<br><br>**STD:** Observes MPR assuming control based on Reactor pressure lowering, MPR yellow light on, annunciator A2-4-4 alarming |
| 9. | ASSURE MPR has taken control by slowly lowering MPR setpoint again in move-and-wait manner, UNTIL reactor pressure lowers by 1 to 2 psig<br><br>**Note:** Reactor pressure may lower by more than 1 to 2 psig due to the relatively coarse nature of the MPR setpoint control switch. | P<br>(F.2.7) | SAT / UNSAT<br><br>**STD:** Rotates MPR setpoint control switch CW to lower, observes Reactor pressure lower |
| 10. | Confirm Reactor pressure steady on MPR | P<br>(F.2.8) | SAT / UNSAT<br><br>**STD:** Observes Reactor pressure stable |
| 11. | Raise EPR to high stop, 1010 psig or raise EPR setpoint approximately 2 psig, observing NO change in Reactor pressure.<br><br>**Note:** Raising the EPR setpoint to 1010 psig is a lengthy adjustment. The EPR setpoint control switch may be released and re-gripped during this step.<br><br>**Cue:** At the evaluator's discretion, when EPR pressure setpoint is greater the 930 psig, inform the candidate, "Time compression. EPR setpoint is 1010 psig". | P<br>(F.2.9) | ***PASS / FAIL**<br><br>**STD:** Rotates EPR setpoint control switch CCW to raise, observes EPR setpoint rise to 1010 psig or approximately 2 psig, releases EPR setpoint control switch when EPR setpoint reaches 1010 psig or EPR setpoint raised by approximately 2 psig. |

| | PERFORMANCE | ACT. CODE<br>P / S / NA | EVALUATOR |
|---|---|---|---|
| 12. | Confirm EPR light OFF | P<br>(F.2.10) | SAT / UNSAT<br><br>**STD:** Observes EPR yellow light off |
| 13. | Adjust MPR setpoint to return Reactor pressure to pre-transfer value<br><br>**Cue:** Another operator will return pressure to the pre-transient value. | P<br>(F.2.11) | SAT / UNSAT<br><br>**STD:** Proper communications used |
| **Evaluator Note:** | **Cue:** Your task is complete. | | |

| | |
|---|---|
| **TASK STANDARD** | MPR has been placed in service in accordance with N1-OP-31 section F.2.0 |

| | |
|---|---|
| **STOP TIME** | |

# JPM Handout

| | |
|---|---|
| **INITIAL CONDITIONS** | Given:<br>• The reactor is operating at approximately 100% power<br>• EPR is in service<br>• Pressure control will be transferred to the MPR<br>• Thermal limits have been verified SAT per N1-OP-31 section H, Operation with One Pressure Regulator Inoperable<br>• An Equipment Operator is standing by at the Front Standard for support |

| | |
|---|---|
| **INITIATING CUE** | **(Operators Name)**, transfer Reactor pressure control from the EPR to the MPR in accordance with N1-OP-31, section F.2. |

Training Id: **2024 NRC NMP1 Simulator JPM S-F** Revision: **0.0**

Title: **Vent Torus through RBEVS**

**Approvals:**

| | Signature / Printed Name | Date |
|---|---|---|
| Developed By | Mike Dougherty | 02/09/24 |
| Validated By | Tim Heagerty | 03/29/24 |
| Facility Reviewer | Dave Mason | |

Approximate Duration: 10 minutes

**Documentation of Performance:**

Performer: _____

Evaluator: _____

Start Time: _____ Stop Time: _____ Completion Time _____

Grade: **Pass / Fail**

Comments: _____

Evaluators Signature: _____ Date: _____

## References

1. N1-EOP-4.1
2. NUREG 1123 K/A 223001 K1.17, (3.2)

Instructor Information

## A. JPM Information

1. Description

   a. This JPM tests the operator's ability to vent the Torus using RBEVS train 11, in accordance with N1-EOP-4.1, section 1.0.

   b. This JPM is NOT considered Alternate Path.

   c. Critical steps are annotated in the Evaluator standard column with a bolded **Pass/Fail**.

2. Task Information:

   a. N1-EOP-4.1-01001

   b. K/A 223001 K1.17, (3.2)

3. Evaluation / Task Criteria

| | |
|---|---|
| Evaluation Method | Perform |
| Evaluation Location | Simulator |
| Time Critical Task | No |
| Alternate Path | No |
| LOD >1.0 | Yes |

4. Recommended Start Location

   a. Unit 1 Simulator

5. Simulator Setup (if required)

   a. The reactor operating at 100% reactor power
   b. Initialize simulator to IC 236
   c. Drywell pressure ~ 1.85 psig as follows:
      i. Take the simulator out of freeze
      ii. Add $N_2$ to the drywell via the $N_2$ Makeup and Bleed System until Drywell pressure is approximately 1.85 psig
      iii. After Shutting off $N_2$ flow, allow Drywell temperature to restore to normal

6. JPM Setup (if required)

    a. Copies of N1-OP-9, Precaution and Limitations and Section H.1.0

## B. Read Before Every JPM Performance

1. For the performance of this JPM, I will act as all those you need to talk to. Prior to providing direction to perform this task, I will provide you with the initial conditions and answer any questions. During task performance, I will identify the steps to be simulated, or discuss and provide cues as necessary. (Note, read the next only if conducting a plant JPM). With the exception of accessing panels, no plant equipment will be physically manipulated. Repositioning of devices will be simulated by discussion and acknowledged by my cues.

## C. Read Before Each Evaluated JPM

1. This evaluated JPM is a measure of your ability to perform this task independently. The US has determined that a verifier is not available and that additional verification will not be provided.

| | |
|---|---|
| **INITIAL CONDITIONS** | Given:<br>• Drywell Pressure is 1.85 psig.<br>• $H_2O_2$ is Out of Service<br>• Chemistry sample has identified Hydrogen in the Drywell<br>• N1-EOP-PC, Primary Containment Control, has been entered<br>• Torus water level is below 27 ft.<br>• Offsite releases are expected to stay below ODCM limits during venting<br><br>**Evaluator:** *Ask trainee if he/she has any questions after presenting initial conditions* |

| | |
|---|---|
| **INITIATING CUE** | **(Operators Name)**, vent the Torus using RBEVS train 11, in accordance with N1-EOP-4.1, section 1.0. |

| | |
|---|---|
| **START TIME** | |

| | **PERFORMANCE** | **ACT. CODE**<br>P / S / NA | **EVALUATOR** |
|---|---|---|---|
| 1. | Provide repeat back of initiating cue<br><br>**Cue:** Acknowledge repeat back providing correction if necessary | P | SAT / UNSAT<br><br>**STD:** Proper communications used |
| 2. | Obtain a copy of the reference procedure and review / utilize the correct section of the procedure | P | SAT / UNSAT<br><br>**STD:** *N1-EOP-4.1, Section 1.0 has been obtained and referenced* |
| 3. | Verify the following valves are closed: | 1.1 | |
| | • 201.2-32, DRYWELL N2 MAKE-UP & BLEED ISOL VALVE 11. | P | SAT / UNSAT<br><br>***STD:*** *Verifies that the green light is ON and the red light is OFF.* |
| | • 201.2-03, DRYWELL N2 MAKE-UP & BLEED ISOL VALVE 12. | P | SAT / UNSAT<br><br>***STD:*** *Verifies that the green light is ON and the red light is OFF.* |

| | PERFORMANCE | ACT. CODE<br>P / S / NA | EVALUATOR |
|---|---|---|---|
| | • 201.2-33, TORUS N2 MAKE-UP & BLEED ISOL VALVE 11. | P | SAT / UNSAT<br>***STD:*** *Verifies that the green light is ON and the red light is OFF.* |
| | • 201.2-06, TORUS N2 MAKE-UP & BLEED ISOL VALVE 12. | P | SAT / UNSAT<br>***STD:*** *Verifies that the green light is ON and the red light is OFF.* |
| | • 201-11, TORUS VENT TO CONDENSER.<br>• **NOTE:** Candidate will likely identify component label in simulator is as "TORUS/DRYWELL…." | P | SAT / UNSAT<br>***STD:*** *Verifies that the green light is ON and the red light is OFF.* |
| | • 201-21, DRYWELL & TOR VENT & PURGE FAN INLET BV. | P | SAT / UNSAT<br>***STD:*** *Verifies that the green light is ON and the red light is OFF.* |
| | • 202-47, EM VENTILATION TIE BV. | P | SAT / UNSAT<br>***STD:*** *Verifies that the green light is ON and the red light is OFF.* |
| | • 201.13-74, HCVS INBOARD PRIMARY CONTAINMENT IV<br>**Cue:** Another operator has verified that valve 201.13-74 indicates closed on the HCVS Primary Operating Panel in the Auxiliary Control Room. | P | SAT / UNSAT<br>***STD:*** *Acknowledges cue using proper communications that IV 210.13-74 has been verified closed* |
| | • 201.13-71, HCVS OUTBOARD PRIMARY CONTAINMENT IV<br>**Cue:** Another operator has verified that valve 201.13-71 indicates closed on the HCVS Primary Operating Panel in the Auxiliary Control Room. | P | SAT / UNSAT<br>***STD:*** *Acknowledges cue using proper communications that IV 210.13-74 has been verified closed* |

| | PERFORMANCE | ACT. CODE<br>P / S / NA | EVALUATOR |
|---|---|---|---|
| 4. | Close 202-36, EM VENTILATION FROM REACTOR BLDG BV | P<br><br>1.2 | ***PASS / FAIL**<br><br>***STD:*** *Unlocks and rotates the keylock selector switch CCW to CLOSE position on L Panel. Verifies Green Light ON, Red Light OFF.* |
| 5. | OPEN 201-18 EM VENTLATION FROM DW & TORUS BV. | P<br><br>1.3 | ***PASS / FAIL**<br><br>***STD:*** *Rotates the control switch CW to the OPEN position and verifies RED light ON and GREEN light OFF.* |
| 6. | IF pressure in indicator 201.2-540A, $N_2$ VENT TO EM VENT PRESS, on L Panel is >0 psig, then perform the following.<br><br>**Note**: The applicant should verify that indicator 201.2-540A is reading 0 psig and should mark step 1.4 as N/A | N/A<br><br>1.4 | SAT / UNSAT<br><br>***STD:*** *Verifies that indicator 201.2-540A is reading 0 psig and should mark this step as N/A.* |
| 7. | • Line up one loop of EVS:<br><br>• For loop 11, open 202-37, EM VENTILATION LOOP 11 INLET BV<br><br>**Note**: based on initial conditions provided the applicant should mark the block for loop 11 and open BV 202-37 | P<br><br>1.5 | ***PASS / FAIL**<br><br>***STD:*** *Rotates the control switch CW to the OPEN position and verifies RED light ON and GREEN light OFF.* |
| 8. | • Start the selected EVS Fan (System 11). | P<br><br>1.6 | ***PASS / FAIL**<br><br>***STD:*** *Rotates the control switch CW to start 202-53 EVS FAN 11. Verifies control switch is RED Flagged, RED light ON and GREEN light OFF.*<br>. |

| | PERFORMANCE | ACT. CODE<br>P / S / NA | EVALUATOR |
|---|---|---|---|
| 9. | Open 201-16, TORUS $N_2$ VENT & PURGE ISOLATION VALVE 11 | P<br><br>1.7 | ***PASS / FAIL**<br><br>***STD:*** *Rotates the control switch CW to the OPEN position and verifies RED light ON and GREEN light OFF.* |
| 10. | Using Pull-To-Stop, throttle open 201-17, TORUS $N_2$ VENT & PURGE ISOLAITON VALVE 12 | P<br><br>1.8 | ***PASS / FAIL**<br><br>***STD:*** *Rotates the control switch CW in a turn and Pull-to-Stop fashion to slowly OPEN 201-31. Verifies both RED light and GREEN light are ON as the valve is throttled OPEN.* |
| 11. | Adjust 201-17, 201-17, TORUS $N_2$ VENT & PURGE ISOLAITON VALVE 12, as necessary to maintain EVS flow <1600 cfm. | P<br><br>1.9 | SAT / UNSAT<br><br>***STD:*** *Verifies flow rate is ≤1600 CFM on EMER VENT FLOW 11 flow meter.* |
| **Note** | Task will be complete when flow rate of ≤ 1600 CFM has been established. | | |
| **CUE** | *Step 1.10 will be completed by another operator when Torus pressure has been lowered to 0 psig.* | | |
| 15. | Report status to US. | P | SAT / UNSAT<br><br>***STD:*** *Proper communications used.* |
| **Evaluator Note:** | **Cue:** *Your task is complete.* | | |

| | |
|---|---|
| **TASK STANDARD** | Torus venting in progress through the Reactor Building Emergency Ventilation system in accordance with N1-EOP-4.1, Section 1.0. |

| | |
|---|---|
| **STOP TIME** | |

# JPM Handout

| | |
|---|---|
| **INITIAL CONDITIONS** | Given:<br>• Drywell Pressure is 1.85 psig.<br>• $H_2O_2$ is Out of Service<br>• Chemistry sample has identified Hydrogen in the Drywell<br>• N1-EOP-PC, Primary Containment Control, has been entered<br>• Torus water level is below 27 ft.<br>• Offsite releases are expected to stay below ODCM limits during venting |

| | |
|---|---|
| **INITIATING CUE** | **(Operators Name)**, vent the Torus using RBEVS train 11, in accordance with N1-EOP-4.1, section 1.0. |

Constellation

Training Id: **2024 NRC NMP1 Simulator JPM S-G** Revision: **0.0**

Title: **Adjust Generator Reactive Output (Alternate Path)**

**Approvals:**

| | Signature / Printed Name | Date |
|---|---|---|
| Developed By | Mike Dougherty | 02/13/24 |
| Validated By | Tim Heagerty | 03/29/24 |
| Facility Reviewer | Dave Mason | |

Approximate Duration: 10 minutes

**Documentation of Performance:**

Performer: _____

Evaluator: _____

Start Time: _____ Stop Time: _____ Completion Time _____

Grade: **Pass / Fail**

Comments: _____

Evaluators Signature: _____ Date: _____

## References

1. N1-OP-32
2. N1-SOP-33B.1
3. NUREG 1123 K/A 700000 AA2.11, (3.5)

Instructor Information

## A. JPM Information

1. Description

   a. This JPM tests the operator's ability to raise Main Generator reactive loading when requested by Power Control in accordance with N1-OP-32, section F.1.0

   b. This JPM is considered alternate path because once the reactive loading has been adjusted a grid disturbance on the 345Kv will occur. The operator will be required to execute N1-SOP-33B.1 to scram the reactor and stabilize the plant.

   c. Critical steps are annotated in the Evaluator standard column with a bolded **Pass/Fail**.

2. Task Information:

   a. N1-SOP-33B.1-01001

   b. K/A 700000 AA2.11, (3.5)

3. Evaluation / Task Criteria

| | |
|---|---|
| Evaluation Method | Perform |
| Evaluation Location | Simulator |
| Time Critical Task | No |
| Alternate Path | Yes |
| LOD >1.0 | Yes |

4. Recommended Start Location

   a. Unit 1 Simulator

5. Simulator Setup (if required)

   a. The reactor operating at 100% reactor power
   b. Initialize simulator to IC 237
   c. Set malfunction **EG11**, 345KV Power Grid Transient, SV =335, RT = 2 min on **TRG 1**
   d. Set up Trigger 1 on an Event Trigger “hzaegvare > 0.612)”
      i. This event trigger will insert Trigger 1 when the MVARs meter is ~ 110 MVAR

6. JPM Setup (if required)

    a. Copies of N1-OP-32, Precaution and Limitations and Section F.1.0

## B. Read Before Every JPM Performance

1. For the performance of this JPM, I will act as all those you need to talk to. Prior to providing direction to perform this task, I will provide you with the initial conditions and answer any questions. During task performance, I will identify the steps to be simulated, or discuss and provide cues as necessary. (Note, read the next only if conducting a plant JPM). With the exception of accessing panels, no plant equipment will be physically manipulated. Repositioning of devices will be simulated by discussion and acknowledged by my cues.

## C. Read Before Each Evaluated JPM

1. This evaluated JPM is a measure of your ability to perform this task independently. The US has determined that a verifier is not available and that additional verification will not be provided.

| | |
|---|---|
| **INITIAL CONDITIONS** | Given:<br>• Unit 1 is at 100% power<br>• Power Control has requested NMP1 to raise reactive loading 50 MVAR<br><br>**Evaluator:** *Ask trainee if he/she has any questions after presenting initial conditions* |

| | |
|---|---|
| **INITIATING CUE** | **(Operators Name)**, raise Main Generator reactive loading 50 MVAR in accordance with N1-OP-32, section F.1.0 |

| | |
|---|---|
| **START TIME** | |

| | **PERFORMANCE** | **ACT. CODE**<br>P / S / NA | **EVALUATOR** |
|---|---|---|---|
| 1. | Provide repeat back of initiating cue<br><br>**Cue:** Acknowledge repeat back providing correction if necessary | P | SAT / UNSAT<br><br>**STD:** Proper communications used |
| 2. | Obtain a copy of the reference procedure and review / utilize the correct section of the procedure | P | SAT / UNSAT<br><br>**STD:** *N1-OP-32, Section F.1.0 has been obtained and referenced* |
| 3. | At E console, ADJUST EXCITATION switch to RAISE MVARS as requested by System Power Control | P<br>(F.1.1) | ***PASS / FAIL**<br><br>***STD:*** *In accordance with N1-OP-32 section F. Observe limitations described in step F.2.1. Rotate EXCITATION switch in the clockwise direction to raise MVARs.* |
| Evaluator Note: | When MVAR loading is raised 50 MVAR above the initial conditions (or at the lead evaluators discretion) insert **TRG1** EG11, 345KV POWER GRID TRANSIENT, SV=335, RT = 2 min. | | |

| | PERFORMANCE | ACT. CODE<br>P / S / NA | EVALUATOR |
|---|---|---|---|
| Alternate Path: | Annunciator A6-3-3 345 KV SYS FREQUENCY HIGH-LOW and/or A6-2-6 345 KV BUS VOLTAGE HIGH-LOW alarm. The operator should recognize these are entry conditions to N1-SOP-33B.1 MAJOR 345 KV GRID DISTRUBANCES. Frequency variation is greater than +/- 1.9 Hz with FIRST STAGE BOWL PRESSURE not in alarm, which requires a reactor SCRAM and Turbine Trip. | | |
| 4. | Observe and report A6-3-3 345 KV SYS FREQUENCY HIGH-LOW and/or A6-2-6 345 KV BUS VOLTAGE HIGH-LOW alarm. Recommend entry into N1-SOP-33B.1.<br>● Instructor cue *"(operator) execute N1-SOP-33B.1"* | P | SAT / UNSAT<br>**STD:** Proper communications used |
| 5. | Enters N1-SOP-33B.1, identifies override conditions are met for scramming the reactor and verifying turbine trip. | P | SAT / UNSAT<br>**STD:** *N1-SOP-33B.1, has been obtained and referenced* |
| 6. | Depress REACTOR TRIP CH 11 / CH 12 pushbuttons | P<br>(N1-SOP-33B.1 override) | ***PASS / FAIL**<br>***STD:*** *At E console, depress REACTOR TRIP CH 11/12 pushbuttons to SCRAM the reactor in accordance with N1-SOP-33B.1 override and N1-SOP-1.* |
| 7. | Place MODE SWITCH in SHUTDOWN | P<br>(N1-SOP-33B.1 override) | ***PASS / FAIL**<br>***STD:*** *At E console, place the Mode Switch in SHUTDOWN in accordance with N1-SOP-33B.1 override and N1-SOP-1.*<br>. |

| | PERFORMANCE | ACT. CODE<br>P / S / NA | EVALUATOR |
|---|---|---|---|
| 8. | Depress UNIT EMERGENCY TRIP pushbutton (or VERIFY Turbine Trip occurred) | P<br>(N1-SOP-33B.1 override) | SAT / UNSAT<br><br>***STD:*** *At E console (or A panel), depress the Unit Emergency Trip pushbutton to trip the Main Turbine (or verify a Turbine Trip has occurred). This action will automatically occur following a reactor SCRAM above 35% reactor power, however per N1-SOP-33B.1 the operator must verify the turbine is tripped.* |
| Evaluator Note: | The operator will recommend entry into N1-SOP-1 Reactor SCRAM and N1-SOP-31.1 Turbine Trip. Another Operator will perform the actions of N1-SOP-1 and N1-SOP-31.1. | | |
| 9. | Notifies Unit Supervisor override actions of N1-SOP-33B.1 are complete. The reactor is scrammed and turbine is tripped.<br><br>**Cue:** *"Understand, reactor is scrammed and turbine is tripped. Another operator will be performing the actions of N1-SOP-1 and N1-SOP-31.1"* | S<br>(N1-SOP-33B.1 override) | SAT / UNSAT<br><br>**STD:** Proper communications used |
| **Evaluator Note:** | **Cue:** *Your task is complete.* | | |

| | |
|---|---|
| **TASK STANDARD** | The initial task is for the operator to raise MVAR loading at the request of Power Control in accordance with N1-OP-32. Once MVAR loading is raised, a 345 kV grid disturbance will be experienced which requires entry into N1-SOP-33B.1 Major 345 kV Grid Disturbances. The magnitude of the grid disturbance is greater than +/- 1.9 Hz which requires a reactor SCRAM and turbine trip in accordance with the override statement. |

| | |
|---|---|
| **STOP TIME** | |

# JPM Handout

| | |
|---|---|
| **INITIAL CONDITIONS** | Given:<br>• Unit 1 is at 100% power<br>• Power Control has requested NMP1 to raise reactive loading 50 MVAR |

| | |
|---|---|
| **INITIATING CUE** | **(Operators Name)**, raise Main Generator reactive loading 50 MVAR in accordance with N1-OP-32, section F.1.0 |

Training Id: **2024 NRC NMP1 Simulator JPM S-H** Revision: **0.0**

Title: **Lineup and Inject Containment Spray Raw Water into Core Spray Loop 11 (RO Only)**

**Approvals:**

| | Signature / Printed Name | Date |
|---|---|---|
| Developed By | Mike Dougherty | 02/09/24 |
| Validated By | Ken Graham | 03/29/24 |
| Facility Reviewer | Dave Mason | |

Approximate Duration: 10 minutes

**Documentation of Performance:**

Performer: _____

Evaluator: _____

Start Time: _____ Stop Time: _____ Completion Time _____

Grade: **Pass / Fail**

Comments: _____

_____

_____

_____

Evaluators Signature: _____ Date: _____

## References

1. N1-EOP-1, Attachment 5, Emergency Cooling System
2. NUREG 1123 K/A 209001 A4.01, (4.3)

Instructor Information

## A. JPM Information

1. Description

   a. This JPM tests the operator's ability to lineup and inject Containment Spray Raw Water to Core Spray Loop 11 per N1-EOP-1, Attachment 5.

   b. This JPM is NOT considered alternate path.

   c. Critical steps are annotated in the Evaluator standard column with a bolded **Pass/Fail**.

2. Task Information:

   a. N1-2226001-01019

   b. K/A 209001 A4.01, (4.3)

3. Evaluation / Task Criteria

| | |
|---|---|
| Evaluation Method | Perform |
| Evaluation Location | Simulator |
| Time Critical Task | No |
| Alternate Path | No |
| LOD >1.0 | Yes |

4. Recommended Start Location

   a. Unit 1 Simulator

5. Simulator Setup (if required)

    a. The reactor is in a shutdown condition (Post-scram)
    b. Initialize simulator to IC 238
    c. Verify inserted malfunctions RR29 and CU01 at 100%
    d. Verify inserted malfunctions CS01A and CS01B
    e. Verify Core Spray and Core Spray Topping Pumps 112 are in PTL with a Caution/Info tag hanging
    f. Take appropriate EOP actions (bypass ADS, maximize injection with LP and CRD, Control Containment Spray, initiate ECs, open 4 ERVs, install N1-EOP-1 Attachment 4 Core Spray jumpers)
    g. Verify inserted remote CT03A at 10% *(This remote will not show when the simulator is reloaded)*
    h. Verify Core Spray and Core Spray Topping pumps 111 are red flagged and not running (tripped)
    i. Verify valves 40-10 and 40-11 are OPEN

6. JPM Setup (if required)

    a. No steps need to be marked up.

## B. Read Before Every JPM Performance

1. For the performance of this JPM, I will act as all those you need to talk to. Prior to providing direction to perform this task, I will provide you with the initial conditions and answer any questions. During task performance, I will identify the steps to be simulated, or discuss and provide cues as necessary. (Note, read the next only if conducting a plant JPM). With the exception of accessing panels, no plant equipment will be physically manipulated. Repositioning of devices will be simulated by discussion and acknowledged by my cues.

## C. Read Before Each Evaluated JPM

1. This evaluated JPM is a measure of your ability to perform this task independently. The US has determined that a verifier is not available and that additional verification will not be provided.

| | |
|---|---|
| **INITIAL CONDITIONS** | Given:<br>• The plant was operating at 100% power<br>• A LOCA has occurred<br>• Core Spray Loop 112 is OOS for maintenance<br>• Core Spray loop 111 pumps have tripped<br>• Core Spray jumpers have been installed per N1-EOP-1, Att. 4.<br>• RPV level cannot be maintained above -126”<br>• Core Spray flow for both loops cannot be established ≥ 180 x $10^6$ lbm/hr<br><br>**Evaluator:** *Ask trainee if he/she has any questions after presenting initial conditions* |

| | |
|---|---|
| **INITIATING CUE** | **(Operators Name)**, Lineup and inject Containment Spray Raw Water to Core Spray Loop 11 per N1-EOP-1, Attachment 5. |

| | |
|---|---|
| **START TIME** | |

| | **PERFORMANCE** | **ACT. CODE** P / S / NA | **EVALUATOR** |
|---|---|---|---|
| 1. | Provide repeat back of initiating cue<br><br>**Cue:** Acknowledge repeat back providing correction if necessary | P | SAT / UNSAT<br><br>**STD:** Proper communications used |
| 2. | Obtain a copy of the reference procedure and review / utilize the correct section of the procedure | P | SAT / UNSAT<br><br>**STD:** N1-EOP-1, Attachment 5 has been obtained and referenced. |
| 3. | Verify each loop Core Spray flow is greater than or equal to 180 x $10^6$ lbm/hr<br><br>**Note**: Applicant should mark this step as N/A based on information provided in the initial conditions | P<br>(3.1) | SAT / UNSAT<br><br>**STD:** Applicant should mark this step as N/A based on initial conditions |

| | PERFORMANCE | ACT. CODE<br>P / S / NA | EVALUATOR |
|---|---|---|---|
| 4. | If the conditions described in Step 3.1 can NOT be established or as directed by SAPs, then inject Raw Water into the RPV as follow:<br><br>**Note**: Based on information provided in the initial conditions, the applicant should determine they need to perform step 3.3 to inject Containment Spray Raw Water to Core Spray Loop 11 per the Initiating Cue. | P<br>(3.2) | SAT / UNSAT<br><br>**STD:** Marks step 3.3 as the steps required to be performed for this attachment based in the direction provided in the Initiating Cue. |
| 5. | Verify at least one Core Spray Pump set injecting, if available, into the RPV through Core Spray Loop 12. | P<br>(3.3.1) | SAT / UNSAT<br><br>**STD:** Determines both Core Spray 12 pump sets injecting into the RPV based on referencing initial conditions or by observing Core Spray 12 pump Red Lights ON, Green Lights OFF, and Core Spray Isolation Valve Red Lights ON, and Green Lights OFF |
| 6. | Verify Core Spray Jumpers installed per N1-EOP-1, Attachment 4.<br><br>**Note**: These jumpers were installed as part of the simulator setup. | P<br>(3.3.2) | SAT / UNSAT<br><br>**STD:** Applicant should verify Core Spray Jumpers installed by verifying installed jumpers inside the N-Panel. |
| 7. | Place AND hold in STOP positon CORE SPRAY TOPPING PUMP 111 | P<br>(3.3.3) | SAT / UNSAT<br><br>**STD:** Rotates Core Spray Topping Pump 111 control switch CCW to the STOP position |

| | PERFORMANCE | ACT. CODE<br>P / S / NA | EVALUATOR |
|---|---|---|---|
| 8. | Place in PULL-TO-LOCK CORE SPRAY PUMP 111<br><br>**Note**: This pump has malfunction CS01A inserted to simulate the pump tripping on initial start. | P<br>(3.3.4) | SAT / UNSAT<br><br>**STD:** Rotates Core Spray Pump 111 control switch CCW and pulls to PTL |
| 9. | Release CORE SPRAY TOPPING PUMP 111 to Neutral position | P<br>(3.3.5) | SAT / UNSAT<br><br>**STD:** Releases Core Spray Topping Pump 111 control switch to the Neutral position |
| 10. | Place AND hold in STOP positon CORE SPRAY TOPPIING PUMP 112<br><br>**Role Play**: Core Spray and Core Spray topping pumps have been tagged out for maintenance. If the applicant informs you this step cannot be accomplished, acknowledge the report and direct the applicant to continue with the next step of the procedure. | P<br>(3.3.6) | SAT / UNSAT<br><br>**STD:** The operator should mark this step as N/A due the pump being tagged out for maintenance per initial conditions. |
| 11. | Place in PULL-TO-LOCK CORE SPRAY PUMP 112<br><br>**Role Play**: Core Spray and Core Spray topping pumps have been tagged out for maintenance. If the applicant informs you this step cannot be accomplished, acknowledge the report and direct the applicant to continue with the next step of the procedure. | P<br>(3.3.7) | SAT / UNSAT<br><br>**STD:** Core Spray Pump 112 is tagged out for maintenance and is in PTL per the initial conditions. They should mark this step as N/A. |

| | PERFORMANCE | ACT. CODE<br>P / S / NA | EVALUATOR |
|---|---|---|---|
| 12. | Release CORE SPRAY TOPPING PUMP 112 to Neutral position<br><br>**Role Play**: Core Spray and Core Spray topping pumps have been tagged out for maintenance. If the applicant informs you this step cannot be accomplished, acknowledge the report and direct the applicant to continue with the next step of the procedure. | P<br>(3.3.8) | SAT / UNSAT<br><br>**STD:** Core Spray Pump 112 is tagged out for maintenance and is in PTL per the initial conditions. They should mark this step as N/A. |
| 13. | Open 93-71, CONT SPRAY RAW WATER 111 INTERTIE to Core Spray Loop 11 | P<br>(3.3.9) | ***PASS / FAIL**<br><br>**STD:** Rotates 93-71 control switch CCW, observes Red Light ON, Green Light OFF |
| 14. | Verifies 93-25, BV-111 Cont Spray HX Raw Water Outlet closed | P<br>(3.3.10) | SAT / UNSAT<br><br>**STD:** Observes 93-25 Green Light ON, Red Light OFF |
| 15. | Verify Core Spray Loop 11 Inside IV's are closed | P<br>(3.3.11) | |
| 15a | • Closes 40-10 | P<br>(3.3.11) | ***PASS / FAIL**<br><br>**STD:** Rotates the control switch CCW to CLOSE and observes Red Light OFF, Green Light ON for 40-10 |
| 15b | • Closes 40-11 | P<br>(3.3.11) | ***PASS / FAIL**<br><br>**STD:** Rotates the control switch CCW to CLOSE and observes Red Light OFF, Green Light ON for 40-11 |

| | PERFORMANCE | ACT. CODE<br>P / S / NA | EVALUATOR |
|---|---|---|---|
| 16. | Start CONTAINMENT SPRAY RAW WATER PUMP 111 (CSRW) | P<br>(3.3.12) | ***PASS / FAIL**<br><br>**STD:** Rotates Containment Spray Raw Water Pump 111 control switch CW and observes Red Light ON and Green Light OFF. Verifies proper system response. |
| 17. | While monitoring CSRW Pump 111 motor amps, throttle 40-10 and 40-11 as follows<br><br>A. Maximize flow rate, maintaining motor current <76 amps<br>B. IF current exceeds 76amps, then throttle 40-10 and 40-11 closed as necessary to reduce current below 76 amps<br><br>**Note**: The evaluator does not have to wait until valves 40-10 and 40-11 are fully open. Once operator demonstrates proficiency at throttling the valves the evaluator can stop the JPM. See Evaluator Note below. | P<br>(3.3.13) | ***PASS / FAIL**<br><br>**STD:** applicant will monitor CSRW pump 111 amps and throttle 40-10 and 40-11 by rotating their respective control switches CW, and then pull out on the control switch to stop the valve at its current position. The applicant will repeat this process until maximum flow has been established from the Containment Spray Raw Water pump. |
| **Evaluator Note:** | **Cue:** *Your task is complete.* | | |

| | |
|---|---|
| **TASK STANDARD** | Containment Spray Raw Water Pump 111 is lined up to Core Spray Loop 11 and is injecting to the RPV in accordance with N1-EOP-1, Attachment 5. |

| | |
|---|---|
| **STOP TIME** | |

# JPM Handout

| | |
|---|---|
| **INITIAL CONDITIONS** | Given:<br>• The plant was operating at 100% power<br>• A LOCA has occurred<br>• Core Spray Loop 112 is OOS for maintenance<br>• Core Spray loop 111 pumps have tripped<br>• Core Spray jumpers have been installed per N1-EOP-1, Att. 4.<br>• RPV level cannot be maintained above -126”<br>• Core Spray flow for both loops cannot be established ≥ 180 x $10^6$ lbm/hr |

| | |
|---|---|
| **INITIATING CUE** | **(Operators Name)**, Lineup and inject Containment Spray Raw Water to Core Spray Loop 11 per N1-EOP-1, Attachment 5. |

Training Id: **2024 NRC NMP1 Plant JPM P-I** Revision: **0.0**

Title: **Transfer Reactor Trip Bus 141 to I&C Bus 130A**

**Approvals:**

| | Signature / Printed Name | Date |
|---|---|---|
| Developed By | Mike Dougherty | 02/09/2024 |
| Validated By | Ken Graham | 04/05/24 |
| Facility Reviewer | Dave Mason | |

Approximate Duration: 15 minutes

**Documentation of Performance:**

Performer: _____

Evaluator: _____

Start Time: _____ Stop Time: _____ Completion Time _____

Grade: **Pass / Fail**

Comments: _____

Evaluators Signature: _____ Date: _____

## References

1. SA-NM-129-1000 Electrical Safety
2. N1-OP-30.2 Attachment 3
3. NUREG 1123 K/A 295003, AA1.01 (4.0)

Instructor Information

## A. JPM Information

1. Description

   a. This JPM tests the operator's ability to transfer Reactor Trip Bus 141 to I&C Bus 130A per N1-SOP-30.2 Attachment 3.

   b. This JPM is NOT considered alternate path.

   c. Critical steps are annotated in the Evaluator standard column with a bolded **Pass/Fail**.

2. Task Information:

   a. N1-SOP-30.2-04001

   b. K/A 264000, A4.04 3.7/3.7

3. Evaluation / Task Criteria

| | |
|---|---|
| Evaluation Method | Simulate |
| Evaluation Location | Plant |
| Time Critical Task | No |
| Alternate Path | No |
| LOD >1.0 | Yes |

4. Recommended Start Location

   a. Unit 1 Turbine Building elevation 261'

5. JPM Setup (if required)

   a. Prepare a copy of N1-SOP-30.2, Attachment

   b. Print out a copy of SA-AA-129, Electrical Safety **(provide if requested)**

   c. Print out a copy of SA-NM-129-1000, Electrical Safety **(provide if requested)**

## B. Read Before Every JPM Performance

1. For the performance of this JPM, I will act as all those you need to talk to. Prior to providing direction to perform this task, I will provide you with the initial conditions and answer any questions. During task performance, I will identify the steps to be simulated, or discuss and provide cues as necessary. (Note, read the next only if conducting a plant JPM). With the exception of accessing panels, no plant equipment will be physically manipulated. Repositioning of devices will be simulated by discussion and acknowledged by my cues.

## C. Read Before Each Evaluated JPM

1. This evaluated JPM is a measure of your ability to perform this task independently. The US has determined that a verifier is not available and that additional verification will not be provided.

| | |
|---|---|
| **INITIAL CONDITIONS** | Given:<br>• The plant is at approximately 50% power<br>• A loss of Power Board 12 has led to the loss of Reactor Trip Bus 141<br><br>**Note:** The Electrical Safety PPE requirements for this task have been assessed and are as follows:<br>• Leather gloves<br>• 100% cotton long sleeve shirt and pants OR 100% cotton short sleeve shirt and pants under fire retardant lab coat<br>• Safety glasses or goggles<br><br>**Evaluator:** *Ask trainee if he/she has any questions after presenting initial conditions* |

| | |
|---|---|
| **INITIATING CUE** | **(Operators Name)**, restore power to Reactor Trip Bus 141 from I&C Bus 130A in accordance with N1-SOP-30.2, Attachment 3. |

| | |
|---|---|
| **START TIME** | |

| | **PERFORMANCE** | **ACT. CODE** P / S / NA | **EVALUATOR** |
|---|---|---|---|
| 1. | Provide repeat back of initiating cue<br><br>**Cue:** Acknowledge repeat back providing correction if necessary. | P | SAT / UNSAT<br><br>**STD:** Proper communications used |
| 2. | Obtain a copy of the reference procedure and review / utilize the correct section of the procedure | P | SAT / UNSAT<br><br>**STD:** N1-SOP-30.2, Attachment 3 has been obtained |

| | PERFORMANCE | ACT. CODE<br>P / S / NA | EVALUATOR |
|---|---|---|---|
| 3. | Confirm I&C Bus 130A power supply is available (PB 13B energized)<br><br>**Cue**: If the applicant looks at any Power Board voltage indication, indicate that voltage is 600VAC<br><br>Cue: If the applicant looks at Power Board 13 breakers, indicate breaker position as follows:<br>• PB 13A Feeder – Closed<br>• PB 13 A-B Tie – Closed<br>• PB 13 B-C Tie – Closed<br>• PB 13C Feeder - Open | P<br>(1.1) | SAT / UNSAT<br><br>**STD:** Determines Power Board 13B is energized by observing voltage present and/or POWERBOARD 13 A-B SECTION TIE BREAKER – closed (TB 277 South) |
| 4. | Open REACTOR TRIP BUS #141 NORMAL SUPPL FROM M.G. SET #141 (TB 277' East)<br><br>**Cue:** Breaker is in the position you described | S<br>(1.2) | ***PASS / FAIL**<br><br>**STD:** Opens breaker REACTOR TRIP BUS #141 NORMAL SUPPLY FROM M.G. SET #141 by positioning breaker lever down |
| 5. | Close REACTOR TRIP BUS #141 MAINTENANCE SUPPLY from I&C Bus 130A (TB 277' East)<br><br>**Cue:** Breaker is in the position you described | S<br>(1.3) | ***PASS / FAIL**<br><br>**STD:** Closes breaker REACTOR TRIP BUS #141 MAINTENANCE SUPPLY by positioning breaker lever up |
| 6. | Notify CRO that Rx Trip bus #141 transfer is completed and half scram and Feedwater Heaters may be reset<br><br>**Role Play**: Acknowledge report, Inform the applicant that the ½ scram has been reset.<br><br>**Cue:** Another operator will reset Feedwater Heaters 121 - 125 | P<br>(1.4) | SAT / UNSAT<br><br>**STD:** Proper communications used |

| | PERFORMANCE | ACT. CODE<br>P / S / NA | EVALUATOR |
|---|---|---|---|
| **Evaluator Note:** | **Cue:** *Your task is complete.* | | |

| | |
|---|---|
| **TASK STANDARD** | Reactor Trip Bus 141 is energized from I&C Bus 130A in accordance with N1-SOP-30.2, Attachment 3. |

| | |
|---|---|
| **STOP TIME** | |

# JPM Handout

| | |
|---|---|
| **INITIAL CONDITIONS** | Given:<br>• The plant is at approximately 50% power<br>• A loss of Power Board 12 has led to the loss of Reactor Trip Bus 141<br><br>**Note:** The Electrical Safety PPE requirements for this task have been assessed and are as follows:<br>• Leather gloves<br>• 100% cotton long sleeve shirt and pants OR 100% cotton short sleeve shirt and pants under fire retardant lab coat<br>Safety glasses or goggles<br><br>**Evaluator:** *Ask trainee if he/she has any questions after presenting initial conditions* |

| | |
|---|---|
| **INITIATING CUE** | **(Operators Name)**, restore power to Reactor Trip Bus 141 from I&C Bus 130A in accordance with N1-SOP-30.2, Attachment 3. |

Training Id: **2024 NRC NMP1 Plant JPM P-J** Revision: **0.0**

Title: **Lineup Liquid Poison to inject to RPV from the Demin Tank**

**Approvals:**

| | Signature / Printed Name | Date |
|---|---|---|
| Developed By | Mike Dougherty | 02/15/24 |
| Validated By | Ken Graham | 04/05/24 |
| Facility Reviewer | Dave Mason | |

Approximate Duration: 15 minutes

**Documentation of Performance:**

Performer: _____

Evaluator: _____

Start Time: _____ Stop Time: _____ Completion Time _____

Grade: **Pass / Fail**

Comments: _____

Evaluators Signature: _____ Date: _____

## References

1. N1-EOP-1, Attachment 12
2. NUREG 1123 K/A 295031, EA1.08 (3.9)

Instructor Information

## A. JPM Information

1. Description

   a. This JPM tests the operator's ability to lineup the Liquid Poison System for RPV injection from the Demin Tank in accordance with N1-EOP-1, Attachment 12.

   b. This JPM is NOT considered alternate path.

   c. Critical steps are annotated in the Evaluator standard column with a bolded **Pass/Fail**.

2. Task Information:

   a. N1-EOP-1-04002

   b. K/A 295031, EA1.08 (3.9)

3. Evaluation / Task Criteria

| | |
|---|---|
| Evaluation Method | Simulate |
| Evaluation Location | Plant |
| Time Critical Task | No |
| Alternate Path | No |
| LOD >1.0 | Yes |

4. Recommended Start Location

   a. Unit 1 Turbine Building

5. JPM Setup (if required)

   a. Prepare a copy of N1-EOP-1, Attachment 12

## B. Read Before Every JPM Performance

1. For the performance of this JPM, I will act as all those you need to talk to. Prior to providing direction to perform this task, I will provide you with the initial conditions and answer any questions. During task performance, I will identify the steps to be simulated, or discuss and provide cues as necessary. (Note, read the next only if conducting a plant JPM). With the exception of accessing panels, no plant equipment will be physically manipulated. Repositioning of devices will be simulated by discussion and acknowledged by my cues.

## C. Read Before Each Evaluated JPM

1. This evaluated JPM is a measure of your ability to perform this task independently. The US has determined that a verifier is not available and that additional verification will not be provided.

| | |
|---|---|
| **INITIAL CONDITIONS** | Given:<br>• The plant has been scrammed.<br>• N1-EOP-RPV (1-2) has been entered<br><br>**Evaluator:** *Ask trainee if he/she has any questions after presenting initial conditions* |

| | |
|---|---|
| **INITIATING CUE** | **(Operators Name)**, lineup the Liquid Poison System for RPV injection from the Demin Tank in accordance with N1-EOP-1, Attachment 12. |

| | |
|---|---|
| **START TIME** | |

| | **PERFORMANCE** | **ACT. CODE**<br>P / S / NA | **EVALUATOR** |
|---|---|---|---|
| 1. | Provide repeat back of initiating cue<br><br>**Cue:** Acknowledge repeat back providing correction if necessary. | P | SAT / UNSAT<br><br>**STD:** Proper communications used |
| 2. | Obtain a copy of the reference procedure and review / utilize the correct section of the procedure<br><br>**Note:** Valve operations are performed on RB El. 298' | P | SAT / UNSAT<br><br>**STD:** N1-EOP-1, Attachment 12 is obtained and referenced |
| 3. | Unlock and close 41-05, BV-LIQUID POISON TANK OUTLET<br><br>**Cue**: the valve is in the position you described<br><br>**Note**: A VA-1 key is required to unlock this valve. | S<br>(3.1) | ***PASS / FAIL**<br><br>**STD:** 41-05 unlocked and shut (fully clockwise). |
| 4. | Unlock and open the following valves: | | |

| | PERFORMANCE | ACT. CODE<br>P / S / NA | EVALUATOR |
|---|---|---|---|
| 4a. | • 41-13, BV-LP PUMP 11 SUCTION FROM DEMIN WATER<br><br>**Cue**: the valve is in the position you described<br><br>**Note**: A VA-1 key is required to unlock this valve. | S<br>(3.2, first bullet) | ***PASS / FAIL**<br><br>**STD:** 41-13 unlocked and open (fully counter clockwise). |
| 4b. | • 41-12, BV-LP PUMP 12 SUCTION FROM DEMIN WATER<br><br>**Cue**: the valve is in the position you described<br><br>**Note**: A VA-1 key is required to unlock this valve. | S<br>(3.2, second bullet) | ***PASS / FAIL**<br><br>**STD:** 41-12 unlocked and open (fully counter clockwise). |
| 5. | Inform the Control Room, start a Liquid Poison Pump per step 3.3<br><br>**Role Play**: Acknowledge the report. | P<br>(3.3) | SAT / UNSAT<br><br>**STD:** Proper communications used |
| **Evaluator Note:** | **Cue:** *Your task is complete.* | | |

| | |
|---|---|
| **TASK STANDARD** | Liquid Poison is aligned to inject water from the Demin Tank into the Reactor Vessel in accordance with N1-EOP-1, Attachment 12. |

| | |
|---|---|
| **STOP TIME** | |

# JPM Handout

| | |
|---|---|
| **INITIAL CONDITIONS** | Given:<br>• The plant has been scrammed.<br>• N1-EOP-RPV (1-2) has been entered<br><br>**Evaluator:** *Ask trainee if he/she has any questions after presenting initial conditions* |

| | |
|---|---|
| **INITIATING CUE** | **(Operators Name)**, lineup the Liquid Poison System for RPV injection from the Demin Tank in accordance with N1-EOP-1, Attachment 12. |

Training Id: **2024 NRC NMP1 Plant JPM P-K** Revision: **0.0**

Title: **Lineup Lake Water to Supply the EC Makeup Tanks Using the Electric Fire Pump (Alternate Path)**

**Approvals:**

| | Signature / Printed Name | Date |
|---|---|---|
| Developed By | Mike Dougherty | 02/09/24 |
| Validated By | Ken Graham | 04/05/24 |
| Facility Reviewer | Dave Mason | |

Approximate Duration: 20 minutes

**Documentation of Performance:**

Performer: _____

Evaluator: _____

Start Time: _____ Stop Time: _____ Completion Time _____

Grade: **Pass / Fail**

Comments: _____

Evaluators Signature: _____ Date: _____

## References

1. N1-SOP-21.2, Control Room Evacuation
2. NUREG 1123 K/A 295016 AA1.09 (4.0)

Instructor Information

## A. JPM Information

1. Description

   a. This JPM simulates a control room evacuation where Emergency Condenser are required for RPV pressure control. This JPM tests the operator's ability to locally operate components associated with maintaining the Emergency Condensers available.

   b. This JPM is considered alternate path because the Diesel Fire Pump will fail upon start, requiring use of the Electric Fire Pump.

   c. Critical steps are annotated in the Evaluator standard column with a bolded ***PASS/FAIL**.

2. Task Information:

   a. N1-SOP-21.2-01001-05

   b. K/A 295016 AA1.09 (4.0)

3. Evaluation / Task Criteria

| | |
|---|---|
| Evaluation Method | Simulate |
| Evaluation Location | Plant |
| Time Critical Task | No |
| Alternate Path | Yes |
| LOD >1.0 | Yes |

4. Recommended Start Location

   a. Unit 1 Screenhouse

5. JPM Setup (if required)

   a. Prepare a copy of N1-SOP-21.2, Section 4.3

   b. Have available, copies of N1-OP-21A, sections H.1.0 and H.2.0.

   c. The applicant will need to simulate the use of a VA-1 key to unlock the valves in this procedure.

## B. Read Before Every JPM Performance

1. For the performance of this JPM, I will act as all those you need to talk to. Prior to providing direction to perform this task, I will provide you with the initial conditions and answer any questions. During task performance, I will identify the steps to be simulated, or discuss and provide cues as necessary. (Note, read the next only if conducting a plant JPM). With the exception of accessing panels, no plant equipment will be physically manipulated. Repositioning of devices will be simulated by discussion and acknowledged by my cues.

## C. Read Before Each Evaluated JPM

1. This evaluated JPM is a measure of your ability to perform this task independently. The US has determined that a verifier is not available and that additional verification will not be provided.

| | |
|---|---|
| **INITIAL CONDITIONS** | Given:<br>• The plant was operating at 100% power.<br>• A control room evacuation has occurred.<br>• Emergency Condenser Makeup Tank levels are low at 6 feet.<br>• Emergency Condenser Makeup cross-connect valve, 60-13, has failed closed.<br><br>**Evaluator:** *Ask trainee if he/she has any questions after presenting initial conditions* |

| | |
|---|---|
| **INITIATING CUE** | **(Operators Name)**, raise Emergency Condenser Makeup Tank levels to 10 feet using the Diesel Fire Pump, in accordance with N1-SOP-21.2, Section 4.3. |

| | |
|---|---|
| **START TIME** | |

| | **PERFORMANCE** | **ACT. CODE**<br>P / S / NA | **EVALUATOR** |
|---|---|---|---|
| 1. | Provide repeat back of initiating cue<br><br>**Cue:** Acknowledge repeat back providing correction if necessary. | P | SAT / UNSAT<br><br>**STD:** Proper communications used |
| 2. | Obtain a copy of the reference procedure and review / utilize the correct section of the procedure | P | SAT / UNSAT<br><br>**STD:** Obtains copy of N1-SOP-21.2 |

| | PERFORMANCE | ACT. CODE<br>P / S / NA | EVALUATOR |
|---|---|---|---|
| 3. | Start at least one of the following Fire Pumps (Screenhouse):<br>• Diesel Fire Pump (Preferred)<br>• Electric Fire Pump<br>**Note:** If the candidate requests an additional procedure to perform the Fire Pump start, give them N1-OP-21A sections H.1.0 and H.2.0. The candidate will then perform additional pre- and post-start actions beyond those listed in the standard, but listed below:<br>• Verify Diesel Fire Pump lubricant level is at Full mark.<br>• Verify Diesel Fire Pump gearcase lubricant level full.<br>• Verify the following:<br>• Engine coolant full<br>• Day tank level 150 gallons or greater<br>• Air tank pressures 11 and 12 > 80 psig<br>• Engine water temperature > 100°F<br>**-OR-**<br>• Verify lubricant levels of Electric Fire Pump is at STANDSTILL LEVEL in top and bottom Bull's Eyes<br>• Verify Electric Fire Pump oil reservoir full<br>**Cue:** When the applicant looks at the indications listed above inform them that the indications are as indicated. | S<br>(SOP-21.2, 4.3.1) | SAT / UNSAT<br>**STD:** Attempts to starts the Diesel Fire Pump by one of the following means:<br>• Rotates control switch counterclockwise to TEST position<br>• Rotates control switch clockwise to the MAN 1 position and depresses the start pushbutton<br>• Rotates control switch clockwise to the MAN 2 position and depresses the start pushbutton |

| | |
|---|---|
| **Evaluator Cue:** | Report that the Diesel Fire Pump failed to start and the engine is now damaged. |
| **Alternate Path:** | The alternate path begins when the operator attempts to start the Diesel Fire Pump. The pump will fail, requiring use of the Electric Fire Pump. |

| | PERFORMANCE | ACT. CODE P / S / NA | EVALUATOR |
|---|---|---|---|
| **Role Play:** | If candidate asks for direction how to proceed, direct them to take appropriate action to raise Emergency Condenser Makeup Tank levels to 10 feet. | | |
| 4. | Starts the Electric Fire Pump<br><br>**Cue:** When Electric Fire Pump control switch is taken to START, inform the candidate that the Electric Fire Pump started.<br><br>**Cue:** If requested, Electric Fire Pump indications are:<br>• Oiler Flow rate = 5 drops per minute<br>• Motor current = 40 amps<br>• Discharge pressure = 140 psig<br>• Strainer D/P = 3.0 psid | S (OP-21A H.1.3 thru H.1.5) | ***PASS / FAIL**<br><br>**STD:** Starts the Electric Fire Pump by rotating control switch clockwise to START |
| **Evaluator Note:** | The candidate will proceed to TB 369' for the remaining steps. | | |
| 5. | Unlock and throttle open the following valves to maintain level in EC Makeup Tanks: | | |
| 5a | 100-68, BV – FIRE WATER TO EMERG COND M/U TANK 12<br>**Cue:** The component is in the position you described | S (SOP-21.2, 4.3.2) | ***PASS / FAIL**<br><br>**STD:** Unlocks 100-68 and removes chain from valve handwheel |
| | **Cue:** The valve is in the position you described | S (SOP-21.2, 4.3.2) | ***PASS / FAIL**<br><br>**STD:** Rotates 100-68 handwheel CCW to open valve |
| | **Cue:** Once candidate has opened 100-68, report that EC M/U Tank 12 level is 10 feet | S | SAT / UNSAT<br><br>**STD:** Observes Makeup tank level |

| | PERFORMANCE | ACT. CODE<br>P / S / NA | EVALUATOR |
|---|---|---|---|
| | **Cue:** The valve is in the position you described | S | SAT / UNSAT<br>**STD:** Rotates 100-68 handwheel CW to close the valve |
| 5b | 100-69, BV – FIRE WATER TO EMERG COND M/U TANK 11<br>**Cue:** The component is in the position you described | S<br>(SOP-21.2, 4.3.2) | ***PASS / FAIL**<br>**STD:** Unlocks 100-69 and removes chain from valve handwheel |
| | **Cue:** The valve is in the position you described | S<br>(SOP-21.2, 4.3.1) | ***PASS / FAIL**<br>**STD:** Rotates 100-69 handwheel CCW to open valve |
| | **Cue:** Once candidate has opened 100-69, report that EC M/U Tank 11 level is 10 feet | S | SAT / UNSAT<br>**STD:** Observes Makeup tank level |
| | **Cue:** The valve is in the position you described | S | SAT / UNSAT<br>**STD:** Rotates 100-69 handwheel CW to close the valve |
| **Evaluators Note:** | **Cue:** Your task is complete. | | |

| | |
|---|---|
| **TASK STANDARD** | The candidates attempted to fill Emergency Condenser Makeup Tanks to 10 feet using the Diesel Fire Pump (DFP). Once the candidates were unable to start DFP, the candidates successfully started the Electric Fire Pump and filled the Emergency Condenser Makeup Tanks to 10 feet in accordance with N1-SOP-21.2. |

| | |
|---|---|
| **STOP TIME** | |

# JPM Handout

| | |
|---|---|
| **INITIAL CONDITIONS** | Given:<br>• The plant was operating at 100% power.<br>• A control room evacuation has occurred.<br>• Emergency Condenser Makeup Tank levels are low at 6 feet.<br>• Emergency Condenser Makeup cross-connect valve, 60-13, has failed closed.<br><br>**Evaluator:** *Ask trainee if he/she has any questions after presenting initial conditions* |

| | |
|---|---|
| **INITIATING CUE** | **(Operators Name)**, raise Emergency Condenser Makeup Tank levels to 10 feet using the Diesel Fire Pump, in accordance with N1-SOP-21.2, Section 4.3. |

**Copy ___ of ___**

Training Id: **NMP1 NRC 2024 Scenario #1** Revision: **0.0**

Title: **Simulator Scenario #1**

| | Signature / Printed Name | Date |
|---|---|---|
| Developed By | / Mike Dougherty | 1/31/2024 |
| Validated By | Elmer Warner | 4/4/2024 |
| | Tim Heagerty | |
| | Ken Graham | |
| Facility Reviewer | / Dave Mason | |

## References

1. N1-OP-30, 4.16 KV, 600V, 480V, House Service
2. N1-OP-43B, Normal Power Operations
3. N1-ARP-A2, Control Room Panel A2
4. N1-SOP-1.4, Stuck Open ERV
5. N1-SOP-25.2, Fuel Failure or High Activity in Rx Coolant or Off Gas
6. N1-SOP-30.2, Loss of Power Board 12
7. N1-SOP-31.1, Turbine Trip
8. N1-SOP-40.1, Loss of RPS
9. N1-SOP-1, Reactor Scram
10. N1-EOP-RPV, RPV Control for Modes 1 and 2
11. N1-EOP-C2, RPV Blowdown for Modes 1 and 2
12. N1-EOP-RR (1-2), Radioactivity Release Control for Modes 1 and 2
13. N1-EOP-1, NMP1 EOP Support Procedure
14. Unit 1 Technical Specifications
15. EP-AA-1013, Addendum 3, Appendix 1, NMP Unit 1 EAL Wallboard

# Instructor Information

## A. Scenario Description

Sequence of Events / Expected Crew Response:

The scenario begins at approximately 100% power. Containment Spray Pump 111 is out of service for maintenance. The crew is directed to cross-tie PB 16A to PB 16B. Following the transfer of the Power Boards the crew will then lower reactor power by approximately 5% using Reactor Recirculation Pumps (RRPs) per the provided instructions.

Following the transfer of PB 16A to PB 16B and reactor power being lowered by 5% using RRPs, Reactor Recirculation Pump (RRP) #13 controller will begin to oscillate. The crew will enter N1-SOP-1.5 due to the Reactor Power $MW_{th}$ oscillations. The crew will attempt to take manual control of RRP #13 controller. Once manual control of RRP #13 is unsuccessful the crew will trip RRP #13 and enter N1-SOP-1.3. The SRO will review Technical Specifications and determine that a 1-hour shutdown statement is in effect until the Reactor Recirculation Pump discharge valve is shut.

Next, an ERV will inadvertently open. The crew will respond per N1-SOP-1.4 and close the open ERV **(Critical Task)**. The ERV inoperability will require Technical Specification entry.

Next, a depleted Generator Stator Water Cooling water demineralizer will result in rising conductivity levels in the cooling water. The crew will respond per the Alarm Response procedures. This includes performing an emergency power reduction per N1-SOP-1.1, and eventually scramming the Reactor per N1-SOP-1. When the Generator trips after the scram, Power Board 12 will fail to transfer to reserve power. The crew will execute N1-SOP-30.2, Loss of Power Board 12, to re-energize the power board. Feedwater Pump 11 will trip shortly after the reactor scram.

Following the scram, a Main Steam line break will occur with a Fuel Element Failure. The MSIVs will fail to close both automatically and manually, leading to an un-isolable leak into the Turbine Building. The running Turbine Building ventilation exhaust fan will trip. The crew will start the standby Turbine Building ventilation exhaust fan; however, it will trip after a short time delay. This will allow an un-monitored, ground level release from the Turbine Building. The crew will enter N1-EOP-RR, Radioactivity Release Control. Field reports will indicate off-site release rates approaching the General Emergency level. The crew will perform an RPV Blowdown per N1-EOP-C2 **(Critical Task)**.

1. Termination Criteria
    a. RPV water level controlled in assigned band
    b. RPV Blowdown in progress or complete.

2. Critical Tasks

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| ***Critical Task (#1)*** | **Given an inadvertently open ERV at power, close the ERV or insert a manual scram prior to Torus temperature exceeding 110°F, in accordance with N1-SOP-1.4.** |
| *CT-1.0 Justification:* | |
| *Safety Significance* | *A manual Reactor scram is required before Torus temperature exceeds 110°F. This reduces the rate of energy production and thus heat input to the Torus. Additionally, this allows evaluating the success of the Reactor scram before boron injection would be required due to Torus temperature in the event of a failure to scram. Closing the ERV prior to the need for the scram avoids the need for these more substantial actions, prevents challenging the plant with a scram, and stops heat input to the Torus.* |
| *Initiating Cue:* | *ERV position, ERV acoustic monitors, ERV tailpipe temperature, Torus temperature, Reactor pressure, and steam flow indicate an open ERV. N1-SOP-1.4 provides direction to close the ERV or scram the Reactor.* |
| *Success Path:* | *The crew will perform the actions of N1-SOP-1.4 to successfully close the affected ERV.* |
| *Performance Feedback:* | *ERV position, ERV acoustic monitors, ERV tailpipe temperature, Torus temperature, Reactor pressure, and steam flow will provide performance feedback regarding success of crew actions to close the ERV. Control rod position and Reactor power will provide performance feedback regarding success of crew actions to scram the Reactor.* |
| *Measurable Performance Standard:* | *Pulling ERV fuses, directing an operator to pull ERV fuses in the field, rotating the depressing the manual scram pushbuttons, and/or Mode Switch to SHUTDOWN will provide observable actions for the evaluation team.* ***Bounding Criteria:*** *Based on procedure requirements of N1-SOP-1.4 to scram the reactor prior to Torus temperature exceeding 110 °F.* |

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| ***Critical Task (#2)*** | **Given the plant with a primary system discharging outside the primary and secondary containments, which cannot be isolated, and the offsite release rate is projected to exceed the E-Plan "General Emergency" level, the crew will perform an RPV blowdown before the offsite release rate reaches the Emergency Plan "General Emergency" level in accordance with N1-EOP-RR (1-2), and N1-EOP-C2 (1-2).** |
| *CT-2.0 Justification:* | |
| *Safety Significance* | *With an unisolable main steam isolation valve and a steam leak into the Turbine Building, a primary system is discharging outside both the primary and secondary containment. This discharge is approaching E-Plan general emergency levels which is an indication that all 3 fission product barriers have been breached and that dose to the public is in progress. Blowing down the RPV will both take away the driving head and send the radionuclides through the Torus to be "scrubbed", thus reducing the release to the public.* |
| *Initiating Cue:* | *The Offsite release rate has already exceeded the "Alert" level of the Emergency Plan and release rate calculations are projected to exceed the "General Emergency" Emergency Plan level.* |
| *Success Path:* | *The crew will perform a RPV blowdown in accordance with N1-EOP-C2 (1-2) to lower pressure within the capacity of all available injection sources.* |
| *Performance Feedback:* | *Emergency Condenser and ERV instrumentation will provide indication that these systems are functioning properly once placed in service. Multiple Reactor pressure indicators and annunciators will provide performance feedback regarding the success of the blowdown.* |
| *Measurable Performance Standard:* | *Manipulation of Emergency Condenser valve and ERV control switches will provide observable actions for the evaluation team.* ***Bounding Criteria:*** *Based on procedural requirements to perform an RPV Blowdown prior to the offsite release rate reaches the Emergency Plan "General Emergency" level in accordance with N1-EOP-RR (1-2)* |

3. Length
   a. ~60 minutes
4. Mitigation Strategy Code
   a. DMS-RR-02, primary system leak outside of primary and secondary containment; fuel failure; offsite release approaching Emergency Plan General Emergency level; RPV Blowdown required.
5. Technical Specifications
   a. TS 3.3.7.b, 15 day LCO
6. EAL Classification
   a. Site Area Emergency per EAL FS1 – Loss or Potential Loss of ANY two barriers
7. Special Orders
   a. None

## B. Initial Conditions

1. IC Number

   a. IC-241

2. Presets / With Triggers

   a. Malfunctions

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| 1) | **CT01A,** CT Pump 111 Trip, FINAL = True | **Inserted** |
| 2) | **RR73C**, RRP13 M/A Station Output Signal Oscillation Fail Trigger, FINAL = True | **TRG 1** |
| 3) | **RR73C_A,** _RRP13 M/A Station Output Signal Oscillation Fail Amplitude, FINAL = 1ma, RT = 1 min | **TRG 1** |
| 4) | **RR73C_P,** _RRP13 M/A Station Output Signal Oscillation Fail Amplitude, FINAL = 60 sec, RT = 1 min | **TRG 1** |
| 5) | **AD05A,** ERV 111 Failure – Opens Electrically, FINAL = +c/w F-PNL FUSE | **TRG 2** |
| 6) | **EG13,** Stator Water Cooling Demin Resin Depletion (0-100%), FINAL = 75% | **TRG 3** |
| 7) | **RX01**, Fuel Cladding Failure, FINAL = 34, INITIAL = 10, RT = 5 min, DT = 3 min | **TRG 3** |
| 8) | **ED27,** PB 12 Auto Transfer Failure, FINAL = True | **Inserted** |
| 9) | **FW03A**, Feedwater Pump Trip 11, FINAL = True | **Inserted** |
| 10) | **MS01**, Steam Line Break Outside PC Area, FV=12, IV=5, RT=4:00 | **TRG 4** |
| 11) | **HV07E**, TB Exhaust Fan Trip 11, FINAL = True, DT = 2 min | **TRG 4** |
| 12) | **HV07F**, TB Exhaust Fan Trip 12, FINAL = True, DT = 1 min | **TRG 5** |
| 13) | **MS13A**, MSIV FAILS OPEN MS01-01(111), FINAL = 10% | **Inserted** |
| 14) | **MS13C**, MSIV FAILS OPEN MS01-03(112), FV=10% | **Inserted** |
| 15) | **FW44**, FWP11, 12 Excessive Motor Restart Trip Algorithm Bypass, FINAL = True | **Inserted** |

   b. Remotes

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| 1) | **AD01A,** ERV 111 IN PLANT Fuses, FINAL = pulled | **TRG 27** |
| 2) | **AD07,** Acoustic Monitor Alarm Reset, FINAL = reset | **TRG 28** |
| 3) | **RP04**, RPS BUS 12 POWER SOURCE, FINAL = MAINT | **TRG 29** |
| 4) | **ED40A,** I&C Bus 130 NC Supply Breaker from IC XFMR(PB13B), FINAL = Open | **TRG 30** |

5) **ED40B**, I&C Bus 130 NO Supply Breaker from PB 167A, FINAL= Closed **TRG 30**

c. Overrides

1) **DI-3078**, POS_2 2F29/NG03A POS A, FINAL = Off **Inserted**

d. Annunciators

1) None

e. Event Triggers

| Event # | Event Action | Command |
|---|---|---|
| **TRG 5,** Initiates when Turbine Building exhaust fan 12 is started | zdhvtbft(6)==0 | **Blank** |

f. Equipment Out of Service

1) Containment Spray Pump #111 is out of service with an info/caution tag

g. Support Documentation

1) N1-OP-30 section H.21, completed up to step H.21.1.c

2) ReMA power reduction with Reactor Recirculation Pumps

h. Miscellaneous

1) Verify Power Board 11 has been transferred to Reserve Power per N1-OP-30, Section H.26 prior to crew taking the shift.

## SHIFT TURNOVER INFORMATION

ON COMING SHIFT: □ N ■ D DATE: Today

**PART I: To be performed by the oncoming Operator before assuming the shift.**

- Control Panel Walkdown (all panels) (SRO, ROs)

**PART II: To be reviewed by the oncoming Operator before assuming the shift.**

- LCO Status (SRO)
- Shift Turnover Information Sheet

Evolutions/General Information/Equipment Status:

- Reactor power is approximately 100%.
- Containment Spray Pump #111 is out of service for maintenance.
- 15 day LCO 3.3.7.b is in effect for Containment Spray

**PART III: Remarks/Planned Evolutions:**

- Cross Tie PB 16A to PB 16B per N1-OP-30. The procedure is in progress up to step H.21.1.c. PB 11 will remain aligned to reserve power. JAF has been notified that PB 11 is on Reserve Power. Attachment 4, OP-NM-108-107-1002, has been completed.
- Then, reduce reactor power approximately 5% using Reactor Recirculation pumps per the provided instructions.

## Event #1: Cross-tie PB16A to PB16B

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| **Event Information** | • PB 11 is aligned to reserve power<br>• The crew will cross-tie Power Board 16A with Power Board 16B |

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| | **SRO**<br>• Directs cross-tie of PB 16A to PB 16B with PB 16 B supplying per N1-OP-30<br>• Provides oversight for evolution |
| **Note:** The crew will be informed that loads have been minimized during the pre-job brief.<br><br>**Note:** Crew will be informed during turnover JAF notification and S-ODP-OPS-0112, Attachment 4 have been completed | **BOP**<br>• Acknowledges direction from SRO<br>• References N1-OP-30, Section H.21 |
| | **ATC**<br>• Monitors plant parameters |
| *-A4-4-4, POWER BD. 16 BKR BY-PASS SW clears*<br><br>**Note:** As soon as A4-4-4 clears proceed to the next event<br><br>**Role Play:** If contacted to mark amps on PB-16 supply transformer, report amps as indicated by the simulator. | **BOP**<br>• Updates the crew that annunciator A4-4-4 will alarm<br>• Places PB16 SUPPLY-BUS TIE BREAKER INTERLOCK BYPASS SWITCH in BYPASS at panel B5<br>• Close R1042, 16 PB-Bus Tie Sect A to Sect B<br>• Open R1041, Supply Bkr to 16 PB (A Section)<br>• Updates the crew that annunciator A4-4-4 will clear<br>• Places PB16 SUPPLY-BUS TIE BREAKER INTERLOCK BYPASS SWITCH in NORMAL at panel B5<br>• Determines PB 11 will remain aligned to reserve per the turnover sheet<br>• Verifies <962 amps on PB-16 Supply transformer |

## Events #2: Lower Reactor Power with Recirculation Flow

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| **Event Information** | • Reactor power is approximately 100%<br>• The crew lowers Reactor power approximately 5% with Recirc |

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| | **SRO**<br>• Directs power reduction with recirculation flow in accordance with N1-OP-43B and the Reactivity Maneuver Instruction (ReMA)<br>• Provides oversight of reactivity maneuver |
| **Note:** Lead examiner may move to the next event once sufficient action is observed on the reactivity manipulation. | **ATC**<br>• Acknowledges direction from SRO<br>• Lowers recirculation flow with master recirculation flow controller<br>• Monitors recirculation flow<br>• Monitors Feedwater flow and RPV water level<br>• Monitors APRMs |
| | **BOP**<br>• Monitors individual RRPs for response<br>  • Individual M/A-Speed Control stations trending uniformly<br>  • Individual RRP indications trending normally for speed decrease<br>• Monitors Feedwater controls for proper response<br>  • FWP 13 FCV responding<br>• RPV water level remains within program band (65” - 83”) |

## Events #3: RRMG Flow Oscillations, Failure of RRP #13 Discharge Valve to Close

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| **Event Information** | • The crew will respond to flow oscillations on RRMG #13<br>• The crew will be unable to take manual control of RRMG #13 and will trip the Recirculation pump<br>• The discharge valve on Reactor Recirculation pump #13 will fail to shut forcing the crew to shut the Suction Isolation valve and bypass valves<br>• SRO will evaluate Tech Specs |

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| Verify the following **override** is **inserted**:<br><br>DI-3078, POS_2 2F29/NG03A-A POS A, FINAL = Off<br><br>When directed by lead examiner, **insert malfunction**:<br><br>**RR73C,** RRP13 M/A Station Output Singal Oscillation Fail Trigger, FINAL = True<br><br>**RR73C_A**, _RRP13 M/A Station Output Signal Oscillation Fail Amplitude, FINAL = 3ma, RT = 1 min<br><br>**RR73C_P**, _RRP13 M/A Station Output Signal Oscillation Fail Amplitude, FINAL = 60 sec, RT = 1 min<br><br>**TRG 1**<br><br>*APRMs, MWth, MWe, RPV water level and RPV pressure will oscillate* | **CREW**<br>• Diagnose/report Recirc pump 13 oscillating |

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| | **SRO**<br>• Acknowledges report of oscillations<br>• May direct entry into SOP-1.5<br>• May direct entry into OP-1, Sect H for Recirculation System Failures<br>• Directs taking manual control of RR Pump 13 M/A station<br>• Acknowledges report that Reactor Power is exceeding 2% power oscillations<br>• Acknowledges report that Reactor Power is exceeding 2% power oscillations<br>• Directs tripping RRP #13<br>• Directs entering into N1-SOP-1.3. |
| **Event 3 continued** | **RO**<br>• Monitors plant parameters |
| **Role Play**:<br>If asked to inspect RRMG #13 **wait** 3 minutes, or 5 minutes after the malfunction is inserted, and **report** that you are hearing loud noises coming from RRMG #13. It sounds as though equipment damage is imminent. (*If the crew directs the operator to take local manual control of RRMG #13, use the role play provided above.)*<br><br>*At the Lead Evaluators discretion, if the crew has not taken any actions to secure RRMG #13 for more than 5 minutes, then manually insert the following malfunction:*<br><br>**RR01C**, RR Pump 13 27x Short Drive Bkr Trip, FINAL = True | **BOP**<br>• Recognizes and reports RR Pump #13 oscillating<br>• Nulls deviation meter on RR Pump #13 (may reference OP-1, Sect. F.1.0)<br>• Places RR Pump #13 M/A station in MAN<br>• Verifies oscillations stop<br>• Notifies SRO/Crew that RR Pump #13 is in manual and that the oscillations continue<br>• Reports RRMG #13 oscillations are causing pump limits to be exceeded<br>• Acknowledges report from the field of impending damage to RRMG #13 and updates the crew |

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| *N1-SOP-1.3 Actions*<br><br>**T.S. 3.6.2a**<br>(1) Instrumentation that initiates scram – control rods shall be inserted, unless there is no fuel in the reactor vessel.<br><br>**T.S. 3.1.7.e:**<br>When operating in four recirculation loops in operation and one loop isolated the reactor may operate at 100% of full licensed power IAW the power/flow limits specified in the LOLR and APLHGR not to exceed the applicable limiting values proved in the COLR for the fuel type, provided the following conditions are met for the isolated loop:<br>• Suction valve, discharge valve and discharge bypass valve in the isolated loop shall be in the closed position and the associated motor breakers shall be locked in the open position.<br>• Associated pump motor circuit breaker shall be opened and the breaker removed.<br>If these conditions are not met, core power shall be restricted to 90.5% of full licensed power. | **SRO**<br>• Acknowledges report that RR Pump 13 is in manual and oscillations continue<br>• Acknowledges report that Reactor Power is exceeding 2% power oscillations<br>• Acknowledges report of impending damage to RRMG #13<br>• Directs tripping RRP #13<br>• Directs entering into N1-SOP-1.3.<br>• Acknowledges report that APRMs are inoperable and an 8 hour ENS report is required.<br>• Evaluates Technical Specifications for INOP APRMs and determines T.S 3.6.2 is in affect while the affected loops discharge valve is open.<br>• Acknowledges report that RRP #13 Discharge valve has failed to close<br>• Acknowledges report to monitor and maintain Reactor power <90% power<br>• Evaluates Technical Specifications for RRP #13 discharge valve failing to close and determines T.S. 3.1.7.e is in affect<br>• Acknowledges the report that APRMs are now operable with the RRP #13 Suction Isolation Valve closed |
| **Event 3 continued** | **RO**<br>• Monitors plant parameters<br>• Verifies P/F map is updated for 4 loop operation.<br>• Will Monitor and maintain Reactor power <90% while closing the RRP #13 Suction Isolation valve<br>• Monitors plant limits for 4 loop operation |

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| | **BOP**<br>• Acknowledges direction to trip RRP #13 and enter N1-SOP-1.3<br>• Trips RRP #13 and enters N1-SOP-1.3<br>• Notifies the SRO that APRMs are inoperable and an 8 hour ENS report is required<br>• Attempts to close RECIRC PUMP 13 DISCHARGE VALVE<br>• Reports that RECIRC PUMP 13 DISCHARGE VALVE fails to close<br>• Reports taking the following actions per N1-SOP-1.3 for reactor recirculation discharge valve failing to close:<br>o Notifies the SRO that Reactor power must be maintained <90%<br>o Close RECIRC PUMP 13 SUCTION VALVE<br>o Opens RECIRC PUMP 13 SUCTION VALVE for 2-3 seconds for valve stem warmup<br>o Reports 30-minute stem warmup has commenced<br>• Notify the SRO that APRMs are operable<br>• Places RRP #13 M/A station in MAN |

## Event #4: Inadvertent ERV Opening

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| **Event Information** | • Reactor Power at approximately 85%<br>• ERV 111 inadvertently opens<br>• Operators will respond per SOP-1.4 to attempt to shut the ERV and SOP-1.1 to lower reactor power<br>• The ERV will close when fuses are pulled |

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| When directed by lead examiner, **insert malfunction**:<br><br>**AD05A,** ERV 111 Failure – Opens Electrically, FINAL = +c/w F-PNL FUSE<br><br>**TRG 2**<br><br>*ERV 111 opens*<br>*Reactor pressure lowers slightly*<br>*Reactor power lowers and then rises slightly*<br>*Torus temperature rises*<br>*Torus level rises*<br>*Expected annunciators:*<br>*F1-4-8, STEAM LINE DETECTION SYS FLOW OFF NORM*<br>*F2-4-1, MAIN STM LINE ELECTROMATIC RELIEF VALVE OPEN*<br>*H3-4-5, PRESS SAFETY/RELIEF VALVES FLOW* | **CREW**<br>• Acknowledges/reports annunciators<br>• Diagnoses ERV 111 has inadvertently opened |

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| **Event 4 continued**<br><br>**T.S. 3.1.5:**<br><br>a. During power operating condition whenever the reactor coolant, pressure is greater than 110 psig and the reactor coolant temperature is greater than saturation temperature, all six solenoid-actuated pressure relief valve shall be operable.<br><br>b. If specification 3.1.5a above is not met, the reactor coolant pressure and the reactor coolant temperature shall be reduced to 110 psig or less and saturation temperature or less, respectively, within ten hours.<br><br>**Note:**<br><br>Reactor power should already approximately ≤ 85% due to the actions of the crew from previous events | **SRO**<br><br>• Acknowledges reports<br>• Directs entry into N1-SOP-1.4, Stuck Open ERV<br>• Directs emergency power reduction to approximately 85% power per N1-SOP-1.1<br>• Provides oversight for reactivity manipulation<br>• Determines that ERV 111 is inoperable per TS 3.1.5.b, requiring a 10-hour shutdown<br>• Directs taking action to close ERV 111 or directs a manual scram prior to Torus temperature exceeding 110°F, in accordance with N1-SOP-1.4<br>• Acknowledges that ERV 111 has closed<br>• Enters N1-EOP-PC if Torus temp rises above 85°F or Torus level rises above 11.25'<br>- Directs Containment Spray to PTL<br>• May direct initiation of Torus Cooling per N1-EOP-1 att 16 or N1-SOP-1.4<br>• Determines loss of Containment Spray system is inoperable per TS 3.3.7d and N1-OP-14, P&L# D.11, Loss of Containment Spray loop sea.l |
| | **ATC**<br><br>• Performs emergency power reduction per N1-SOP-1.1<br>• Lowers power to approximately 85% using recirc flow and/or cram rods<br>• Monitors APRMs<br>• Monitors Recirculation flow<br>• Monitors Feedwater flow and RPV water level<br>• Monitors position on power to flow map<br>• **If Torus temperature approaches 110°F, inserts manual Reactor scram**<br><br>**CT-1** |

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| **Event 4 continued**<br><br>**Role Play:** When directed as Operator to go to Auxiliary Control Room and verify ERV 111 is open, wait one minute and report ERV 111 is open. Subsequent reports on acoustic monitor status may be given immediately, with close attention paid to the actual status of the ERV.<br><br>**Note:** Safety glasses, gloves, no metal, and long sleeves or a lab coat are required for pulling fuses in F panel.<br><br>**Role Play:** If the Operator is directed to pull fuses or place the ERV switches to PWR-OFF in the RB wait 3 minutes and **insert remote**:<br>**AD01A,** ERV 111 IN PLANT Fuses, FINAL = pulled<br>**TRG 27**<br>Report fuses are pulled -or- switches are in PWR-OFF position.<br><br>**Role Play:** When directed as operator to reset the acoustic monitor, wait 1 minute and **insert remote**:<br>**AD07,** Acoustic Monitor Alarm Reset, FINAL = reset<br>**TRG 28**<br>Report acoustic monitor is reset. | **BOP**<br>• Enters N1-SOP-1.4<br>• Determines ERV 111 is open using:<br>- Valve indicating lights on F panel<br>- Red ERV flow indicating light on F panel<br>• Sends an operator to the Aux Control Room to verify ERV open using Acoustic Monitor<br>• May send an operator to RB 237' to standby for pulling local ERV fuses<br>• Attempts to close ERV 111 by performing one or all of the following:<br>- Depresses ADS Timer Reset pushbuttons<br>- Cycles Control Switch for ERV 111<br>- **Pulls control power fuses F15 and F30 in F panel (ERV 111)**<br>o **OR**<br>- **Directs operator to pull ERV 111 fuses or place ERV switches to PWR-OFF on RB 237'**<br>**CT-1**<br>• Checks with operator in Aux Control Room to see if ERV is still open<br>• Determines/verifies ERV closes<br>• Directs reset of acoustic monitor<br>• Notifies crew that ERV 111 has closed<br>• Monitors Torus temperature<br>• Reports if/when Torus temperature exceeds 85°F or Torus level exceeds 11.25'<br>• Places Containment Spray pumps in PTL if directed |

| **Event 4 continued** | **BOP continued** |
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| **T.S. 3.3.7.d** *(Containment Spray in Torus Cooling)*<br><br>If a containment spray system or its associated raw water system becomes inoperable and all the components are operable in the other systems, the reactor may remain in operation for a period not to exceed 7 days or in accordance with the Risk Informed Completion Time program. (This LCO is in affect due to the loss of the loop seal per N1-OP-14, P&L# D.11.)<br><br>**Role Play**:<br><br>If asked to perform the RICT for T.S. 3.3.7.d, acknowledge the request. This action will not completed before the end of the scenario. | • Places Torus cooling in service when directed, per N1-SOP-1.4, att 2 or N1-EOP-1 att 16:<br>- Close CONT SPRAY BYPASS BV(s) for selected loop<br>- Verifies closed 80-115<br>- Verifies closed 80-114<br>- Verifies closed Cont Spray Discharge IV for selected loop<br>- Verifies open CONT SPRAY BYPASS BV for selected loop<br>- Fully opens 80-118<br>- Starts Containment Spray Raw Water pump in selected loop<br>- Starts Containment Spray pump in selected loop |

## Events #5, & #6: Rising Generator Stator Wtr Cond., Failure of PB 12 Fast Transfer

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| **Event Information** | • Crew will respond to rising Generator Stator Water Cooling Conductivity<br>• The crew will be required to Trip the Main Turbine per N1-SOP-31.1<br>• The crew will scram the reactor followed by tripping the Main Turbine<br>• Following the reactor scram, Power Board 12 will fail to fast transfer but will be available |

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| **<u>Role Play</u>**<br>When directed by the lead instructor / evaluator, **insert** the following **malfunction**:<br><br>**EG13**, Stator Water Cooling Demin Resin Depletion (0-100%), FINAL = 75%<br>**RX01**, Fuel Cladding Failure, FINAL = 34, INITIAL = 10, RT = 5 min, DT = 2 min<br>**TRG 3**<br><br>**Severity level may be modified, if required to raise conductivity.**<br><br>**Note:**<br>**If** the crew decides to transfer House Service Loads to Reserve power before Scramming the reactor, **then manually** insert the following **malfunction before** the crew transfers the power boards:<br><br>**RP03**, Reactor Scram, FINAL = True<br><br>*The following plant response for Stator Conductivity occurs:*<br>• *Conductivity indications on Stator Water Conductivity XR-(2B) on B2 Panel will begin to rise*<br>*When conductivity reaches 0.5micromhos*<br>• *A2-2-1 GENERATOR STAT WATER HIGH CONDUCTIVITY, alarms*<br>• *Computer point D101, STATOR WTR CONDUC 0.5 YES alarms* | **CREW**<br>• Responds to annunciator A2-2-1 for system high conductivity<br>• Recognizes rising Generator Stator Water conductivity<br><br>• Acknowledges/reports annunciator H1-1-7, OFF GAS HIGH RADIATION<br>• Recognizes rising offgas and main steam line radiation levels<br>• Diagnoses fuel failure |

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| • *A2-1-3, STATOR COOLING alarms*<br><br>*The following plant response for Fuel Cladding Failure Occurs:*<br>• *Rising off-gas radiation levels*<br>• *Rising main steam line radiation levels*<br>• *Expected Annunciators:*<br>• *H1-1-7, OFF GAS HIGH RADIATION*<br>• *H1-2-7, OFF GAS RAD MON 11-12 FILTER ΔP SAMPLE FLOW*<br>• *H1-3-7, MAIN CNDSR OG TIMER STARTED ISOL TD 15 M*<br>• *H1-4-8, AREA RADIATION MONITORS*<br>• *F1-2-7, MAIN STEAM RAD MONITOR CH 11 HI/LO (later)*<br>• *F4-2-2, MAIN STEAM RAD MONITOR CH 1*<br>• *2 HI/LO (later)*<br><br>Verify the following **malfunctions** are **preset**:<br><br>**ED27**, PB 12 Auto Transfer Failure, FINAL = True<br><br>**FW03A,** Feedwater Pump Trip 11, FINAL = True<br><br>*Power Board 12 voltage drops to zero after Generator trip*<br>*Feedwater pump 11 starts and then immediately trips* | |

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| **Role Play**<br>If dispatched to Stator Water Cooling skid to investigate wait 1 minute and report storage tank level is normal and steady. Demineralizer flow is normal.<br>*Conductivity continues to rise and reaches 9.9 micromhos/cm*<br>• Computer point D106 STATOR WTR CONDUC 9.9 YES alarms<br><br>**Role Play:**<br>*If contacted as Chemistry to validate reading, report that readings are valid.* | **BOP**<br>• Confirm alarm on computer printout.<br>• Monitor conductivity recorder for trends.<br>• Informs crew of current conductivity readings and trends<br>• May dispatch operator to Stator Water Cooing skid to monitor storage tank level and determine proper demineralizer flow<br>• IF conductivity continues to rise, THEN reduce generator load. Informs crew load reduction is required.<br>• IF cause of conductivity cannot be determined OR conductivity reaches 9.9 micromhos/cm, THEN trip turbine AND enter N1-SOP-31.1. Informs crew of turbine trip requirement. |
| **Note:**<br>High pressure Feedwater will be unavailable until Power Board 12 is re-energized.<br><br>**Note:**<br>Recommend initiating the next event once the crew has stabilized the plant post-scram and PB 12 is re-energized. | **US**<br>• Acknowledge reports from crew<br>• Direct power reduction due to rising conductivity<br>• Establishes critical parameter for conductivity and directs reactor scram/turbine trip prior to reaching 9.9 micromhos/cm<br>• Directs entry into N1-SOP-25.2, Fuel Failure/High Activity<br>• Acknowledges the scram report<br>• Enters N1-EOP-RPV, RPV Control, on low Reactor water level<br>• Following reactor scram directs entry into N1-SOP-1, Reactor Scram<br>• Directs Reactor water level control 53-95” using Feedwater/Condensate and CRD<br>• *Directs Reactor pressure control 800-1000 psig on Turbine Bypass Valves or Emergency Condenser 11*<br>• *Enters N1-EOP-SC, Secondary Containment Control on high Radiation levels.*<br>• *Acknowledges failure of Power Board 12 to transfer to reserve power*<br>• *Directs entry in N1-SOP-30.2, Loss of Power Board 12* |

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| | **RO**<br>• When directed depresses the REACTOR TRIP 11 AND 12 pushbuttons<br>• Places the Modes Switch in SHUTDOWN<br>• Provide scram report<br>• Notifies the crew that Power Board 12 is deenergized |
| **Event 5 & 6 continued**<br><br>**Role Play:** When dispatched as Chemistry to take samples, acknowledge request.<br><br>**Role Play:** When directed to close Turbine Building Roof Vents, Sidewall Vents and Roll Doors, wait 2 minutes then report the Turbine Building Roof Vents, Sidewall Vents and Roll Doors are CLOSED. Also report that RP is with you, and they have detected higher than normal rad levels near the offgas piping.<br><br>**Note:** Normal Full Power Background (NFPB) Main Steam Line (MSL) radiation levels are approximately 410 mR/hr at this power level. Any value between 400-500 mR/hr is reasonable. This makes a reasonable range of 1500-1875 mR/hr for the 3.75 times NFPB benchmark.<br><br>**Note:**<br>High pressure Feedwater will be unavailable until Powerboard 12 is re-energized. | **BOP**<br>• Executes N1-SOP-25.2<br>• Notifies Chemistry to sample offgas and reactor coolant for gross activity<br>• Directs operator to close Turbine Building Roof Vents, Sidewall Vents and Roll Doors<br>• Monitors ARMs and rad monitors to determine plant radiation levels<br>• Notifies US/Crew of rising Main Steam Line Radiation Monitor levels<br>• Notifies US if MSL Rad Monitors approach/exceed 3.75 X Normal Full Power Background (NFPB)<br><br>• Performs Reactor water level control actions of N1-SOP-1:<br>  • Attempts to restore RPV water level to 53-95” by controlling injection and rejecting through RWCU, as necessary<br>  • Determines both electric Feedwater pumps NOT running<br>• May close MSIVs<br>• May verify Vessel Isolation per N1-SOP-40.2 |

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| **Event 5 & 6 continued**<br><br>**Role Play:** If contacted to check PB 12, wait two minutes and report that you see nothing abnormal at the powerboard.<br><br>**Role Play:**<br>If directed to place IC bus 130 on the maintenance supply **wait** 2 minutes and insert the **following remote**:<br><br>**ED40A,** I&C Bus 130 NC Supply Breaker from IC XFMR(PB13B), FINAL = Open<br><br>**ED40B**, I&C Bus 130 NO Supply Breaker from PB 167A, FINAL= Closed<br><br>**TRG 30**<br><br>**Role Play:**<br>If directed to place Reactor Trip Bus 141 on its maintenance supply **wait** 2 minutes and insert the **following remote**:<br><br>**RP04**, RPS BUS 12 POWER SOURCE, FINAL = MAINT<br><br>**TRG 29**<br><br>**Note:**<br>At the examiner's discretion, move on to next event when PB 12 has been re-energized. | **ATC/BOP**<br>• Executes N1-SOP-30.2<br><br>N1-SOP-30.2 Actions:<br>• Verifies Service Water pump 11 running<br>• Continuously monitors for thermal hydraulic instability<br>• Place I&C Bus 130 on maintenance per attachment 2<br>• Verifies:<br>  • TBCLC pump 11 running<br>  • IAC 11 running<br>  • RBCLC system running<br>• Re-energizes Powerboard 12:<br>  • Answers "Are A5-4-1, A5-4-2, A5-1-8, OR A5-2-8 in alarm?" No<br>  • Obtains SRO permission to re-energize Powerboard 12<br>  • Verifies open Breaker R122<br>  • Inserts Sync. Key into R123 AND rotates clockwise to ON<br>  • Closes Breaker R123<br>  • Rotates Sync. Key counterclockwise to OFF AND removes<br>  • Closes Breakers R124 and R121<br>• Restores Reactor Trip bus 141 AND Feedwater Heating per Attachment 3 |

## Events #7, #8, & #9: Main Steam Line Break in Turbine Building, MSIVs Fail to Close, Turbine Building Ventilation Exhaust Fan Trips

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| **Event Information** | • Fuel failure is complicated by a MSL break in the TB<br>• Two MSIVs fail to isolate<br>• Lose TB Exhaust ventilation leading to unmonitored ground level release. |

When directed by the lead examiner, **insert malfunctions**:

**MS01,** Steam Line Break Outside PC Area, FV=12, IV=5, RT=4:00

**TRG 4**

*Main Steam tunnel temperatures rise*
*Turbine Building radiation levels rise*

*Expected Annunciators:*
*F1-2-2, RPS CH 11 MAIN STM LINE BREAK*
*F2-3-2, MAIN STM LINE BREAK AREA TEMP HIGH*
*F4-2-7, RPS CH 12 MAIN STM LINE BREAK*
*H1-4-8, AREA RADIATION MONITORS*

The following **malfunction** also **inserts** on **TRG 4:**

**HV07E,** TB Exhaust Fan Trip 11, FINAL = True, DT = 2 min

Verify the following **malfunction** is **automatically inserted** when Turbine Building exhaust fan 12 is started:

**HV07F**, TB Exhaust Fan Trip 12, FINAL = True, DT = 1 min

**TRG 5**

Verify the following **malfunctions** are **preset**:

**MS13A,** MSIV FAILS OPEN MS01-01(111), FINAL = 10%

**MS13C,** MSIV FAILS OPEN MS01-03(112), FINAL = 10%
*Two MSIVs fail mid-position on closure*

**CREW**

- Acknowledges/reports annunciators H1-1-7, OFF GAS HIGH RADIATION

**Diagnoses Main Steam Line break**

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| **Events 7, 8, & 9 continued**<br>**Role Plays:**<br>When the crew has determined both Turbine Building exhaust fans are inoperable, **-OR-** as directed by the lead examiner, **-OR-** when requested as RP to assess dose, wait 2 minutes then report as Off-Site Dose Assessment Manager that dose assessment indicates 50 mr/hr TEDE and rising at the site boundary.<br><br>Three minutes later, report that dose is 500 mrem TEDE and rising at the site boundary.<br>(Criteria for SAE RS1 is met)<br><br>Three minutes after previous field report, report that dose assessment indicates 800 mrem TEDE and rising at the site boundary.<br><br>Three minutes later, report that dose assessment indicates 950 mrem TEDE and rising at the site boundary. We are approximately 3 minutes from exceeding the GE TEDE dose limit<br><br>Three minutes later, report that dose assessment indicates 1100 mrem TEDE and rising at the site boundary. (Criteria for GE RG1 is met)<br><br>If asked for an update between reports, give an extrapolated value for dose at the site boundary.<br><br>If dispatched into the plant for any other task, wait 2 minutes and then report that there is excessive steam in the Turbine Building and you have backed out. | **SRO**<br>• Acknowledges reports<br>• Directs MSIVs closed<br>• Directs Turbine Building evacuation<br>• Acknowledges failure of two MSIVs to close<br>• May direct cooldown using Turbine Bypass Valves or Emergency Condensers <100°F/hr<br>• Acknowledges trip of Turbine Building exhaust fan 11<br>• Directs start of Turbine Building exhaust fan 12<br>• Acknowledges trip of Turbine Building exhaust fan 12<br>• Dispatches personnel to monitor on-site/off-site radiation levels<br>• Acknowledges off-site release rate reports<br>• Enters N1-EOP-RR due to off-site release rate above Alert level (RN10A/B >1.5x10E5 cps)<br>• May direct rapid depressurization with Emergency Condensers >100°F/hr<br>• Determines off-site release rate is approaching the General Emergency level<br>• Enters N1-EOP-C2, RPV Blowdown<br>• Directs initiation of ECs<br>• **Directs open 4 ERVs**<br><br>**CT-2** |

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| **Events 7, 8, & 9 continued**<br><br>**Role Play:**<br>If directed to manually close the MSIV's, wait 5 minutes and report that RP won't allow access to the area. | **ATC/BOP**<br>• Reports failure of MSIVs to automatically close on high steam tunnel temperature<br>• Attempts to manually close MSIVs<br>• Reports failure of two MSIVs to close<br>• Makes Turbine Building evacuation announcement<br>• May open Turbine Bypass Valves or initiate Emergency Condensers to cooldown <100°F/hr<br>• Reports trip of Turbine Building exhaust fan 11<br>• Starts Turbine Building exhaust fan 12<br>• Reports trip of Turbine Building exhaust fan 12<br>• Dispatches personnel to monitor on-site/off-site radiation levels<br>• Acknowledges/reports off-site release rate reports<br>• May perform rapid depressurization with Emergency Condenser 11 >100°F/hr<br>• Initiates EC 11/12<br>• **Open 4 ERVs** **CT-2**<br>• Controls Reactor water level 53-95" using Condensate/Feedwater and CRD |

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| **Event Termination Criteria** | • RPV water level controlled in assigned band<br>• RPV Blowdown in progress or complete |

**Copy ___ of ___**

Training Id: **NMP1 NRC 2024 Scenario #2** Revision: **0.0**

Title: **Simulator Scenario #2**

| | Signature / Printed Name | Date |
|---|---|---|
| Developed By | / Mike Dougherty | 2/1/2024 |
| Validated By | Elmer Warner | 4/4/2024 |
| | Tim Heagerty | |
| | Ken Graham | |
| Facility Reviewer | / Dave Mason | |

## References

1. N1-OP-11, Reactor Building Closed Loop Cooling System
2. N1-OP-43B, Normal Power Operations
3. N1-ARP-F1, Control Room Panel F1
4. N1-ARP-H1, Control Room Panel H1
5. N1-SOP-1.3, Recirc Pump Trip at Power
6. N1-SOP-1.5, Unplanned Reactor Power Change
7. N1-SOP-5.1, Loss of Control Rod Drive
8. N1-SOP-1, Reactor Scram
9. N1-EOP-RPV, RPV Control for Modes 1 and 2
10. N1-EOP-C2, RPV Blowdown for Modes 1 and 2
11. N1-EOP-SC (1-2), Secondary Containment Control for Modes 1 and 2
12. N1-EOP-1, NMP1 EOP Support Procedure
13. Unit 1 Technical Specifications

# Instructor Information

## A. Scenario Description

Sequence of Events / Expected Crew Response:

The scenario begins at approximately 90% power. Containment Spray Pump 111 is out of service for maintenance. The crew is directed to Place RBCLC pump 13 in service and place RBCLC 12 in standby per N1-OP-11. Shortly after the pump swap RBCLC pump 13 will trip requiring the crew to place RBCLC pump 12 back into service. After the pump swap and subsequent trip of RBCLC pump 13 , the crew will raise reactor power to 100% utilizing Reactor Recirculation pumps.

Then a Main Steam Line Rad monitor will fail inoperable. The SRO will address Technical Specifications.

Next, the crew will experience a fault on PB 101 which will result in a loss of Reactor Recirculation Pump 13 and entry into N1-SOP-1.3, RECIRC PUMP TRIP at POWER. The Unit Supervisor will evaluate Technical Specifications and determine that reactor power will be limited to 90.5% until until the recirc loop has been isolated.

Next, the on service CRD Flow Control Valve (FCV) will fail closed. The crew will enter N1-SOP-5.1, LOSS of CONTROL ROD DRIVE, to place the standby FCV into service. The Unit Supervisor will evaluate Technical Specifications and enter LCO 3.1.1.

Next the RWCU system will develop a leak into the Secondary Containment. The crew will enter N1-EOP-SC due to the cleanup leak into the Reactor Building. All efforts to isolate the leak will be unsuccessful. The crew will be required to scram the reactor prior to any general area temperature exceeding a max safe temperature **(Critical Task)**. When the crew attempts to scram the reactor the manual scram pushbuttons will fail to scram the reactor. The crew will be able to successfully scram the reactor utilizing the Mode Switch or ARI pushbuttons.

Following the reactor scram the crew will monitor and report that all Turbine Valves have failed closed. The crew will be required to shift pressure control to Emergency Condensers and ERVs.

When the crew receives reports from the field that two general areas have exceeded maximum safe temperatures, the crew will perform an RPV Blowdown per N1-EOP-C2 **(Critical Task)**.

1. Termination Criteria
    a. RPV Blowdown in progress

2. Critical Tasks

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| ***Critical Task (#1)*** | **Given an un-isolable RWCU leak outside primary containment and one general area temperature approaching a maximum safe limit, the crew will insert a manual scram prior to any general area he field report indicating one general area temperature exceeding a max safe limit in accordance with N1-EOP-SC (1-2).** |
| *CT-1.0 Justification:* | |
| *Safety Significance* | With an un-isolable primary system discharging outside of Primary Containment resulting in general area temperature approaching the maximum safe limit, the Reactor must be scrammed. This reduces the rate of energy production and thus the heat input, radioactivity release, and break flow into the Secondary Containment. This also ensures the Reactor is shutdown prior to need for a blowdown. |
| *Success Path:* | The crew will be able to scram the reactor utilizing the Mode Switch and ARI pushbuttons per N1-SOP-1 |
| *Initiating Cue:* | *Annunciators, field reports, and RWCU flow and temperature changes, failure of isolation valves to isolate, will provide indication of the leak coming from the RWCU system.* |
| *Measurable Performance Standard:* | *Manipulation of Reactor Scram Pushbuttons, Mode Switch, and ARI pushbuttons, will provide observable actions for the evaluation team.* ***Bounding Criteria****: Before exceeding any maximum safe temperature in accordance with N1-EOP-SC..* |
| *Performance Feedback:* | *Control rod position and Reactor power indications will provide performance feedback regarding the success rod insertion.* |

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| ***Critical Task (#2)*** | **Given an un-isolable RWCU leak outside primary containment and two general area temperatures above the maximum safe limit, within 15 minutes the crew will commence an RPV Blowdown in accordance with N1-EOP-C2.** |
| *CT-2.0 Justification:* | |
| *Safety Significance* | An un-isolable primary system discharging outside of Primary Containment resulting in two general area temperatures above the maximum safe limit indicates a wide-spread problem posing a direct and immediate threat to Secondary Containment. A blowdown minimizes flow through the break, rejects heat to the suppression pool in preference to outside the containment, and places the primary system in the lowest possible energy state. |
| *Success Path:* | The crew will open 4 ERVs in accordance with N1-EOP-SC (1-2), Secondary Containment Control for Modes 1 and 2, and N1-EOP-C2 (1-2), RPV Blowdown for Modes 1 and 2, to rapidly lower the energy released into the Secondary Containment. |
| *Initiating Cue:* | *Multiple annunciators will provide indications of a primary system discharging into Secondary Containment. Field reports will provide indication that two general area temperature limits are above the maximum safe value. N1-EOP-SC (1-2) provides direction to perform an RPV Blowdown per N1-EOP-C2 (1-2).* |
| *Measurable Performance Standard:* | *The crew will manually initiate Emergency Condensers and open ERVs. This will provide observable actions for the evaluation team.* ***Bounding Criteria:*** *Based on procedural requirements in N1-EOP-SC (1-2) to perform an RPV Blowdown once 2 or more General areas have exceeded their Maximum Safe Values. Time requirement based on Operations representative recommendation.* |
| *Performance Feedback:* | *Emergency Condenser and ERV instrumentation will provide indication that these systems are functioning properly once placed in service. Multiple Reactor pressure indicators and annunciators will provide performance feedback regarding the success of the blowdown.* |

3. Length

    a. ~60 minutes

4. Mitigation Strategy Code

    a. DMS-SC-01, Primary Leak in secondary containment, exceed max safe level, radiation and/or temperature n 2 or more areas; blowdown required.

5. Technical Specifications

    a. TS 3.3.7.b, 15 day LCO

6. EAL Classification

    a. Site Area Emergency per EAL FS1 – Loss or Potential Loss of ANY two barriers

7. Special Orders

    a. None

## B. Initial Conditions

1. IC Number

   a. IC-242

2. Presets / With Triggers

   a. Malfunctions

| | | |
|---|---|---|
| 1) | **CT01A,** CT Pump 111 Trip, FINAL = True | **Inserted** |
| 2) | **CW04C,** RBCLC Pump 13 Trip, FINAL = True, DT = 1 min | **TRG 1** |
| 3) | **RM51A**, PRM Steam Line rad Mon 111 – Norm Malfunction, SV = 0 | **TRG 2** |
| 4) | **ED06**, PB 101 Electrical Fault, FINAL = True | **TRG 3** |
| 5) | **RD36A**, CRD FCV 44-151 FAILURE – CLOSED, FINAL = True | **TRG 4** |
| 6) | **CU11**, CU Coolant Leak Outside of Drywell, FINAL = 45 | **TRG 5** |
| 7) | **TC12**, All Bypass Valves Fail – Closed (0-100%), FINAL = 100% | **TRG 5** |
| 8) | **CU14**, CU Isolation Valves Stuck Open, FINAL = True | **Inserted** |

   b. Remotes

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|---|---|---|
| 1) | **RD05,** CRD FLOW CONTROL VALVE ISOL, FINAL = nc30b | **TRG 16** |

   c. Overrides

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|---|---|---|
| 1) | **DI-4113,** Clear 1E46 RPS CH-11 Trip, FINAL = Off | **Inserted** |
| 2) | **DI-4114,** Clear 1E47 RPS CH-12 Trip, FINAL = Off | **Inserted** |

   d. Annunciators

1) None

   e. Event Triggers

| Event # | Event Action | Command |
|---|---|---|
| **TRG 1** – Activates when RBCLC Pump #12 switch is taken to “STOP” | zdcwrbtp(2)==1 | **Blank** |

| Event # | Event Action | Command |
|---|---|---|
| **TRG 16,** Swaps CRD FCV when operator operates switch in Control room | zdrdfcvb==1 | **Blank** |
| | | |

f. Equipment Out of Service

1) Containment Spray Pump #111 is out of service with an info/caution tag

g. Support Documentation

1) ReMA for raising Reactor Power with Reactor Recirculation Pumps

h. Miscellaneous

1) Ensure RBCLC pumps 11 and 12 are in service at the start of the scenario.

## SHIFT TURNOVER INFORMATION

ON COMING SHIFT: □ N ■ D DATE: Today

**PART I: To be performed by the oncoming Operator before assuming the shift.**

- Control Panel Walkdown (all panels) (SRO, ROs)

**PART II: To be reviewed by the oncoming Operator before assuming the shift.**

- LCO Status (SRO)
- Shift Turnover Information Sheet

Evolutions/General Information/Equipment Status:

- Reactor power is approximately 90%.
- Containment Spray Pump #111 is out of service for maintenance.
- 15 day LCO 3.3.7.b is in effect for Containment Spray

**PART III: Remarks/Planned Evolutions:**

- Place RBCLC pump #13 in service and place RBCLC pump #12 in standby per N1-OP-11, Section F.1. Venting of the pump returning to service is NOT required. All RBCLC Heat Exchangers are in service.
- Then, raise reactor power to rated using Reactor Recirculation pumps per the provided instructions.

## Event #1 & #3: Place RBCLC Pump #13 in Service and RBCLC Pump #12 in Standby, RBCLC Pump #13 Trips

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| **Event Information** | • Reactor Power is at approximately 90% power<br>• The crew will place RBCLC Pump #13 in service and RBCLC Pump #12 will be place in STBY. |

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| | **SRO**<br>• Directs placing RBCLC Pump #13 in service and placing RBCLC pump #12 in STBY per N1-OP-11<br>• Provides oversight for evolution<br>• Acknowledges report that RBCLC Pump #13 has tripped<br>• Directs placing RBCLC Pump #12 back into service per the N1-ARP-H1 |
| **Note:** The crew will be informed venting of RBCLC pump #13 prior placing it in service is not required. The crew will also be informed that all RBCLC HX are in service at the start of the evolution. | **BOP**<br>• Acknowledges direction from SRO<br>• References N1-OP-11, Section F.1 |
| | **ATC**<br>• Monitors plant parameters |
| **Note:** The operator should N/A mark the steps for venting the RBCLC Pump #13 based on guidance provided in the Shift Turnover.<br><br>**Role Play:** If contacted to verify proper pump start RBCLC pump #13 and/or verify that RBCLC Pump #12 has stopped, **wait** 2 minutes and **report** that we have a good pump start for RBCLC Pump #13 and RBCLC Pump #12 has stopped. | **BOP**<br>• Makes a plant announcement for starting and securing RBCLC pumps #13 and #12<br>• Starts RBCLC #13<br>• Should determine that venting RBCLC pump #13 is not required based on information provided on the shift turnover page<br>• Confirms system pressure and pump amps indicate system is stable.<br>• Should determine based on the number of RBCLC HXs in service that they will be required to take manual control of TCV-70-137 and mark this step as N/A.<br>• Stops RBCLC Pump #12 |

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| **Verify** the following **malfunction inserts** when RBCLC Pump #12 control switch is taken to STOP:<br><br>**CW04C**, RBCLC Pump 13 Trip, FINAL = True, DT = 1 min<br><br>**TRG 1**<br><br>**Role Play:** If asked to investigate trip of #13 RBCLC pump, **wait** 2 minutes and **report** overcurrent trip on the circuit breaker for #13 RBCLC pump and the motor is hot to the touch. | • Updates the crew that RBCLC Pump #13 is in service and RBCLC pump #12 has been placed in STBY.<br>• After a short time-delay RBCLC pump #13 trips<br>• Updates the crew that RBCLC pump #13 has tripped<br>• Acknowledges direction to place RBCLC Pump #12 in service.<br>• Places RBCLC Pump #12 in service per N1-OP-11, Section F.1.0 |

## Events #2: Raise Reactor Power with Recirculation Flow

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| **Event Information** | • Reactor power is approximately 90%<br>• The crew raised Reactor power with Recirc |

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| | **SRO**<br>• Directs power ascension with Recirculation flow in accordance with N1-OP-43B and the Reactivity Maneuver Instruction (REMA)<br>• Provides oversight of reactivity maneuver |
| | **ATC**<br>• Acknowledges direction from SRO<br>• Raises Recirculation flow with master Recirculation flow controller<br>• Monitors APRMs<br>• Monitors Recirculation flow<br>• Monitors Feedwater flow and RPV water level<br>• Observes power-to-flow map restrictions<br>• |
| **Note:**<br>Recommend initiating the next event once Reactor power has been raised 2-3%. | **BOP**<br>• Monitors individual RRPs for response<br>  • Individual M/A-Speed Control stations trending uniformly<br>  • Individual RRP indications trending normally for speed increase<br>• Monitors Feedwater controls for proper response<br>  • FWP 13 FCV responding to power change<br>• RPV water level remains within program band (65” - 83”) |

## Event #4: Main Steam Line Radiation Monitor Failure

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| **Event Information** | • Reactor Power at approximately 95%.<br>• A MSL Rad monitor will fail inoperable<br>• SRO will make Technical Specification determination for the failed instrument |

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| When directed by the examiner, **insert malfunction**:<br><br>**RM51A,** PRM Steam Line Rad Mon 111, Norm Malfunction, SV = 0 **TRG 2**<br><br>*Expected Annunciator:*<br>*F1-2-7, MAIN STEAM RAD MONITOR CH 11 HI/LO* | **CREW**<br>• Acknowledge/report Annunciator F1-2-7, MAIN STEAM RAD MONITOR CH 11 HI/LO<br>Diagnose failure of MSL Rad Monitor 111 |
| **Role Play:** When requested to investigate the MSL rad monitor failure, report that you will begin an immediate investigation | **SRO**<br>• Acknowledges report of MSL Rad Monitor alarm<br>• Directs execution of ARP for F1-2-7<br>• Acknowledges MSL Rad Monitor has failed<br>• Informs WEC, requests investigation of the failure<br>• Enters TS Table 3.6.2.h (requires placing channel in the tripped condition within 12 hours or verifying Mechanical Vacuum pump isolated)<br>• Briefs crew on failure of MSL Rad Monitor |
| | **ATC**<br>• Monitors plant parameters |
| | **BOP**<br>• Executes ARP for F1-2-7<br>• Goes to back panel or checks PPC and determines the other MSL Rad Monitors indicate normal readings<br>• Observes downscale light on MSL Rad Monitor Ch 11 and diagnoses the Monitor is INOP<br>• Notifies crew MSL Rad Monitor Ch 11 is INOP |

## Event #5: Power Board 101 Fault

| | |
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| **Event Information** | • Reactor Power at approximately 95%<br>• Crew will respond to a fault on PB 101<br>• N1-SOP-1.3 will need to be executed due to the loss of a Recirc Pump |

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| When directed by examiner, **insert malfunction:**<br><br>**ED06,** PB 101 Electrical Fault, FINAL = True<br>**TRG 3**<br><br>*Loss of voltage on PB 101*<br>*Lowering Reactor power*<br>*Lowering Recirc flow*<br>*Expected Annunciators:*<br>*A5-1-1, POWER BOARD 101 R1014 TRIP*<br>*A4-2-7, POWER BOARD 101 LOCKOUT 86BT*<br>*F2-2-3, React Recirc M-G Set 13* | **CREW**<br>• Diagnose/report loss of PB 101<br>• Diagnose/report the trip of RRP 13<br>• |
| | **SRO**<br>• Acknowledges reports<br>• Directs entry into N1-SOP-1.3 for RRP trip<br>• May direct entry into N1-SOP-1.5 for unplanned power change<br>• Acknowledges APRMs are inoperable for scram and rod block functions of TS 3.6.2<br>• Provides oversight of reactivity change during closure of RRP 13 discharge valve<br>• Reviews Powerboard 101 loads<br>• Reviews TS 3.1.7.e and determines Reactor power may be raised to 100%<br>• Determines thermal limits penalty |

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| **Event 5 continued** | **RO**<br>• Monitors plant parameters<br>• Monitors Reactor power and recirculation flow during closure of RRP suction valve<br>• Verifies position on four loop P/F map<br>• May take a critical parameter of number of operating recirc pumps |
| **Role Play:** If dispatched to PB 101, wait 2 minutes to report R1014 tripped on overcurrent and that Electrical Maintenance is at the PB investigating. | **BOP**<br>• May execute N1-SOP-1.5 for unplanned power change<br>• Monitors RPV water level response to transient<br>• Enters N1-SOP-1.3 for RRP trip<br>• Verifies proximity to restricted zone on four loop Power to Flow Map<br>• Notifies SRO that APRMs are inoperable<br>• Closes RRP 13 discharge valve<br>• Opens RRP 13 discharge valve for 2-3 seconds<br>• Notifies SRO that APRMs are operable<br>• Green flags RRP 13 control switch<br>• Places RRP 13 M/A station to MAN |

## Events #6: CRD Flow Control Valve (FCV) fails closed

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| **Event Information** | • Crew will swap to alternate FCV in accordance with N1-SOP-5.1<br>• SRO will make Technical Specification determination for the failed CRD FCV |

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| When actions for the loss of Power Board 101 are complete and when directed by the examiner, **insert malfunction:**<br><br>**RD36A,** CRD FCV 44-151 FAILURE – CLOSED, FINAL = True<br><br>**TRG 4**<br><br>*Expected plant response:*<br>*CRD charging water pressure rises*<br>*CRD drive water pressure lowers*<br>*CRD cooling water pressure and flow lower*<br>*Expected Annunciator:*<br>*F3-1-5, CRD CHARGING WTR PRESSURE HI/LO* | **CREW**<br>• Acknowledge/report annunciator F3-1-5, CRD CHARGING WTR PRESS HI/LO<br>Diagnose CRD FCV 11 is closed |
| | **SRO**<br>• Acknowledge report from the crew<br>• Direct entry into N1-SOP-5.1, LOSS OF CRD<br>• Acknowledge report that CRD FCV has failed<br>• Acknowledge report that CRD FCV has been shifted and is in AUTO<br>• Enter Technical Specifications 3.1.6.c while CRD flow less than 50 gpm. |
| | **ATC**<br>• Monitor plant parameters |

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| **Role Play:** As Equipment Operator, when directed to transfer CRD FCV, wait 1 minute and inform the control room that you are standing by<br><br>**Role Play:** As Equipment Operator, when directed to open 44-148 and 44-152, wait 1 minute and report task completion (Step 1.3.1, both bullets)<br><br>**Role Play:** As Equipment Operator, when directed to place 113-297 SEL-CRD FCV Service Selector in the up position, **wait** 3 seconds and **report** 113-297 SEL-CRD FCV Service Selector is in the up position. (Step 1.3.2, first bullet)<br><br>**Note:** When the control room operator takes the CRD Flow Control Transfer Switch to "VALVE 12" TRG 16 will automatically **insert remote:**<br><br>**RD05,** CRD FLOW CONTROL VALVE ISOL, FINAL = nc30b<br>**TRG 16**<br>This precludes the console operator from having to time the insertion of this remote.<br><br>**Role Play:** As Equipment Operator, when directed to close 44-150 and 44-153, wait 1 minute and report task completion. (Step 1.3.5) | **BOP**<br>• Acknowledge direction from SRO<br>• Execute N1-SOP-5.1<br>• Answers “Is a CRD pump operating” YES<br>• Determines need to switch CRD Flow Control Valves per N1-SOP-5.1, Attachment 1:<br>• Dispatch PO to report to RB 237’ and establish communications<br>• Place CRD Flow Control in MAN and 0 demand<br>• Direct Equipment Operator to open 44-148 and 44-152<br>• Direct Equipment Operator to place 113-297 SEL-CRD FCV Service Selector in the up position.<br>• Places CRD FLOW CONTROL VALVE TRANSFER switch in VALVE 12 position<br>• Adjust FIC 44-146B, CRD FLOW CONTROL, to set CRD Flow between 32 x $10^3$ and 33 x $10^3$ LB/HR<br>• Places FIC 44-146B, CRD FLOW CONTROL M/A station in AUTO<br>• Direct Equipment operator to close 44-150 and 44-153<br>• Monitor flow<br>• Place CRD Flow Control in AUTO |

## Events #7, #8, & #9: Unisolable RWCU leak in RX Bldg, RPS Pushbuttons fail to scram, All TBVs fail closed

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|---|---|
| **Event Information** | • Unisolable RWCU leak into Reactor Building<br>• RPV Blowdown will be required<br>• Crew may anticipate blowdown<br>• Reactor Trip pushbuttons 11 and 12 fail to scram the reactor<br>• All TBVs fail closed |

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| When directed by lead examiner, **insert malfunction**:<br><br>**CU11,** CU Coolant Leak Outside of Drywell, FINAL = 45<br><br>**TC12,** All Bypass Valves Fail – Closed (0-100%), FINAL = 100%<br><br>**TRG 5**<br><br>*Rising temperatures, pressures and radiation levels in the Secondary Containment*<br>*RBEVS auto-starts (if secured from previous event)*<br>*RBVS isolates (if restored from previous event)*<br>*RWCU IVs fail to close on isolation signal and manually*<br>*Expected Annunciators:*<br>*H1-4-8, AREA RADIATION MONITORS*<br>*L1-4-3, REACT BLDG VENT RAD MONITOR OFF NORMAL*<br>*L1-3(4)-6, EMER VENT SYS CHANNEL 11(12) RELAY OPERATE (~1 min)*<br>*K3-3-4, CLEAN-UP SYS LEAK AREA T HI (~5 min)*<br>*L1-3-3, CONTINUOUS AIR RAD MONITOR (~1 min)*<br>*K3-1-6, CU Pump 12 Trip*<br>*Later Reactor Building D/P goes to Zero*<br><br>Verify the following **malfunctions** are **preset**:<br><br>**CU14,** CU Isolation Valves Stuck Open, FINAL = True<br><br>Verify the following **overrides** are **preset:**<br><br>**DI-4113,** Clear 1E46 RPS CH-11 Trip, FINAL = Off<br>**DI-4114,** Clear 1E47 RPS CH-12 Trip, FINAL = Off<br>*(Overrides listed above prevent the Reactor Trip Pushbuttons from operating)* | **CREW**<br>• Diagnoses/reports leak from RWCU into secondary containment<br>• Diagnoses/reports RWCU failed to isolate<br>• Recognize/report failure of the Reactor Trip Pushbuttons<br>• Recognize/report failure of Turbine bypass valves opening |

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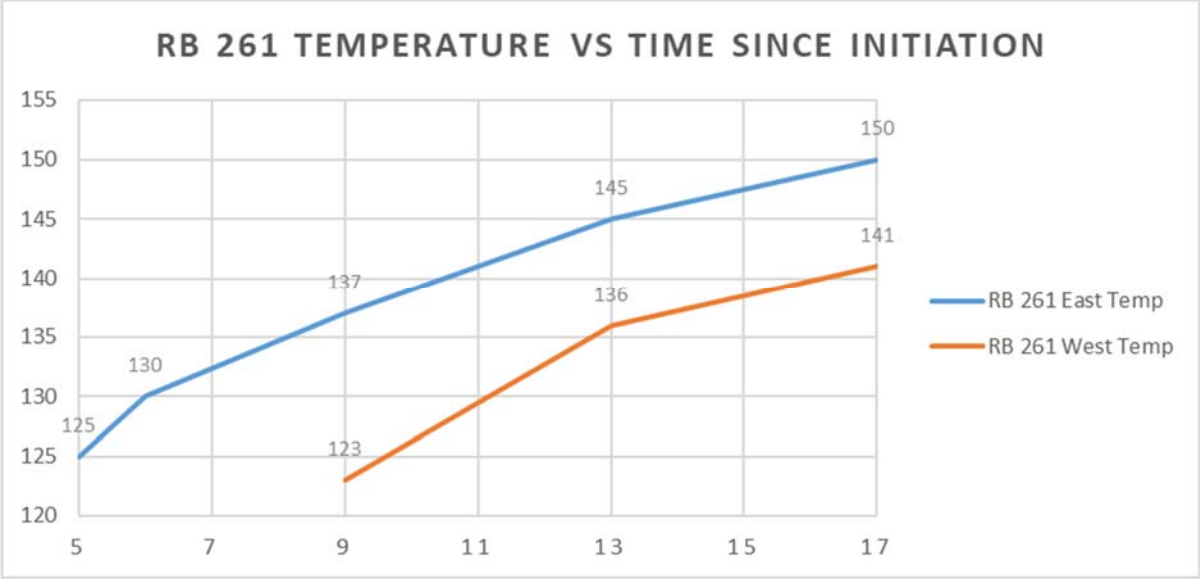
| **Event 7, 8 and 9 continued** | **SRO**<br>• Acknowledges reports<br>• Enters N1-EOP-SC, Secondary Containment Control, on high Reactor Building Vent rad levels, high Reactor Building area temperatures loss of D/P and rad levels<br>• Directs RWCU system isolation<br>• Directs Reactor Building evacuation<br>• Acknowledges report that RWCU failed to isolate both automatically and manually<br>• Acknowledges report that All Turbine Bypass valves have failed closed<br>• Directs Reactor Pressure control utilizing Emergency Condensers and ERVs.<br>• Directs dispatching of an operator and RP tech to obtain general area temperatures and radiation levels in the Reactor Building<br>• When Reactor Building Ventilation exhaust radiation exceeds 5 mR/hr, directs verification of RB Vent isolation and RBEVS initiation<br>• Determines area temperatures and radiation levels are above setpoints in Tables T and R, and transitions to N1-EOP-SC circle 27<br>• Determines a primary system is discharging into the Reactor Building and the discharge cannot be isolated, and transitions to N1-EOP-SC circle 28<br>• Before any area temperature or radiation level reaches 135ºF or 8 R/hr, respectively:<br>• **Directs manual scram (CT-1.0)**<br>• Acknowledges scram report |

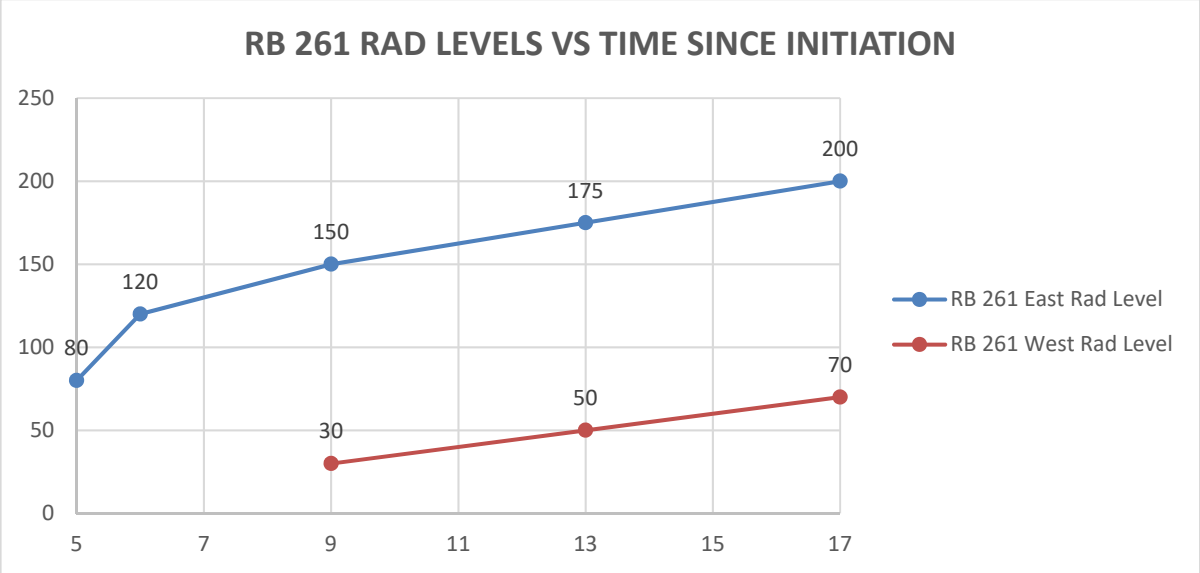
| | |
|---|---|
| **Event 7, 8, and 9 continued**<br><br>**Note:** Anticipatory blowdown is likely to be directed from N1-EOP-RPV once one Reactor Building General Area temperature is above 135°F with a second temperature trending towards 135°F. | **SRO continued**<br>• Enters N1-EOP-RPV, RPV Control on low RPV water level<br>• Answers "Are all control rods inserted to at least position 04?" YES<br>• Directs entry into N1-SOP-1, Reactor Scram<br>• Directs RPV water level control 53-95" using Feedwater/Condensate and CRD<br>• Directs RPV pressure maintained <1080 psig using Emergency Condensers & ERVs<br>• May direct anticipatory blowdown with Emergency Condensers with cooldown in excess of 100°F/hr<br>• Acknowledges reports of Reactor Building temperatures and radiation levels<br>• When report is received that 2 General Areas temperatures are above 135°F, enters N1-EOP-8, RPV Blowdown<br>• Answers "Are all control rods inserted to at least position 04" YES<br>• Answers "Drywell pressure?" <3.5 psig<br>• Directs initiation of Emergency Condensers<br>• Answers "Torus water level?" >8 ft<br>• **Directs open 4 ERVs (CT-2.0)** |

| | |
|---|---|
| **Event 7, 8. and 9 continued** | **RO**<br>• Makes evacuation announcements due to steam leak<br>• Depresses REACTOR TRIP 11 and 12 pushbuttons<br>• Recognizes/reports failure of the REACTOR TRIP 11 and 12 pushbuttons to scram the reactor<br>• Places Reactor Mode Switch to SHUTDOWN<br>• Manually initiates ARI (BOP)<br>• Provides scram report<br>• Performs scram verification actions of N1-SOP-1, Reactor Scram:<br>• Confirms all rods inserted<br>• Observes Reactor power lowering<br>• Places IRMs on range 9<br>• Inserts IRM and SRM detectors<br>• Down-ranges IRMs as necessary<br>• Reduces Recirc Master flow to 25-43 x $10^6$ lbm/hr<br>• Verifies main turbine and generator tripped<br>• Controls RPV pressure as directed using Emergency Condensers or ERVs |

| | |
|---|---|
| **Event 7, 8, and 9 continued**<br><br>**Role Play:** If dispatched to attempt RWCU valve closures, acknowledge direction and delay action.<br><br>**Note:** No General Area temperature information is available until an operator has been dispatched to the Reactor Building to monitor area temperatures. The timeline of field reports should commence when annunciator K3-3-4 begins to alarm. Two minutes after an operator is dispatched, use the attached graphs to report RB temperatures and radiation levels, with the first report being available 5 minutes after event initiation.<br><br>**Role Play:** If dispatched as the Fire Brigade Leader (FBL), **wait** 2 minutes and **report** that you have steam coming from the Reactor Water Clean Up room. | **BOP**<br>• Attempts to isolate RWCU by closing 33-01R, 33-02R and 33-04R<br>• Recognizes/reports failure of RWCU IVs (33-01R, 33-02R and 33-04R) to close<br>• Dispatches operator and RP tech to obtain Reactor Building General Area temperatures and radiation levels<br>• Notifies crew of reports on General Area temperatures and radiation levels<br>• Performs RPV water level control actions of N1-SOP-1:<br>• Restores RPV water level to 53-95" by controlling injection. (unable to reject through RWCU)<br>• Determines #13 FWP was running<br>• Determines RPV water level is recovering<br>• Verifies at least one Electric FW Pump running<br>• Terminates 13 FWP injection as follows:<br>  • Places FWP 13 FCV in manual and closes<br>  • Disengages 13 FWP<br>  • Gives 29-10, FEEDWATER PUMP 13 BLOCKING VALVE a CLOSE signal<br>• Verifies RPV water level above 53"<br>• Verifies 11/12 FWP controllers in MANUAL and set to zero output<br>• Resets HPCI signal<br>• Places 11 or 12 FWP BYPASS Valve in AUTO, sets to 65-70"<br>• If RPV level reaches 85 inches and rising, then:<br>  • Verifies off all Feedwater Pumps<br>  • Secures CRD Pumps not required<br>  • Closes FWIVs if required<br>• Closes MSIVs if required |

| **Event 7, 8, and 9 continued** | **BOP continued**<br>• Maintains RPV water level in assigned band<br>• Notifies crew of reports on General Area temperatures and radiation levels<br>• When EOP-C2, RPV Blowdown, is entered:<br>  • Verifies Emergency Condensers in service<br>  • Verifies open 39-05<br>  • Verifies open 39-06<br>  • **Opens 4 ERVs (CT-2.0)** |
|---|---|



| | |
|---|---|
| **Event Termination Criteria** | • RPV Blowdown in progress |

**Copy ___ of ___**

Training Id: **NMP1 NRC 2024 Scenario #3** Revision: **0.0**

Title: **Simulator Scenario #3**

| | Signature / Printed Name | Date |
|---|---|---|
| Developed By | / Mike Dougherty | 2/2/2024 |
| Validated By | Elmer Warner | 4/4/2024 |
| | Tim Heagerty | |
| | Ken Graham | |
| Facility Reviewer | / Dave Mason | |

## References

N1-OP-5, Control Rod Drive System

N1-OP-10, Reactor Building Heating Cooling and Ventilation System

N1-OP-16, Feedwater System Booster Pump to Reactor

N1-OP-43A, Plant Startup

N1-ARP-K1, Control Room Panel K1

N1-ARP-L1, Control Room Panel L1

N1-SOP-24.1, TBCLC Failure

N1-SOP-1, Reactor Scram

N1-EOP-RPV, RPV Control for Modes 1 and 2

N1-EOP-PC, Primary Containment Control for Modes 1 and 2

N1-EOP-1, NMP1 EOP Support Procedure

N1-EOP-HC, EOP Hard Cards Procedure

Unit 1 Technical Specifications

# Instructor Information

## Scenario Description

Sequence of Events / Expected Crew Response:

The scenario begins at approximately 1% power. Containment Spray Pump 111 is out of service for maintenance. The crew is directed to place Feedwater Booster Pump #13 into service per N1-OP-16. After the pump has been placed into service, the crew will continue with rod withdrawals to establish and maintain a 75°F/hr HUR. Once the crew commences with rod withdrawals to establish a 75°F/hr HUR one of the control rods will double notch. The crew will take actions per N1-OP-5 to restore the rod to its intended position.

Then, the RB Ventilation Rad Monitor will fail upscale. The Reactor Building Ventilation system (RBVS) will fail to automatically isolate in which the crew will take actions to shutdown RBVS and verify the Reactor Building Emergency Ventilation system (RBEVS) has started. The failure of the RBVS system to automatically isolate will require a Technical Specification entry.

Next, IRM #11 will fail downscale. The crew will bypass the failed instrument to clear the rod block in accordance with N1-ARP-F2 and N1-OP-38B. Once IRM #11 has been bypassed the Rod Block will clear and allow the crew to continue pulling rods to establish a 75°F/hr HUR.

Next, an Emergency Condenser (EC) #11 tube leak will occur. The crew will isolate the affected EC per the ARP **(Critical Task**). The tube leak in the isolated EC will require a Technical Specification entry.

Next, Turbine Building Closed Loop Cooling (TBCLC pump 11 will trip. TBCLC pump #12 will fail to auto start requiring the crew wo manually start TBCLC pump #12. Shortly after the crew starts TBCLC #12, the pump will trip requiring the crew to manually scram the reactor per N1-SOP-24.1. Following the scram, Condensate Pump #11 will trip and all remaining Condensate pumps will trip when the crew attempts to start them. This will remove all high pressure injection sources from the Feedwater system.

When the scram occurs a LOCA will also occur. The crew will also monitor and report that all Core Spray pumps failed to auto start. The crew must take action to manually start the Core Spray pumps. The crew will utilize all sources of low pressure injection sources to restore and maintain RPV level > TAF (**Critical Task).**

- Termination Criteria

1) RPV water level being restored to the ordered band
2) Drywell pressure controlled < PSP utilizing Containment Spray Pumps

- Critical Tasks

| ***Critical Task (#1)*** | **Given indications of an Emergency Condenser tube leak, the crew will isolate the Emergency Condenser within 15 minutes of receiving indications of the tube leak per N1-ARP-K1-1-4.** |
|---|---|
| *CT-1.0 Justification:* | |
| *Safety Significance* | *The EC tube leak results in a radiological release to the atmosphere. The crew will isolate the EC steam line to terminate the release before an emergency declaration is required.* |
| *Success Path:* | *The crew will isolate the affected EC in accordance with N1-ARP-K1 to minimize the release or radioactivity to the public.* |
| *Initiating Cue:* | *Multiple annunciators will provide indications of a tube leak in the Emergency Condenser. Rising water temperatures and rad levels on the Emergency Condenser Vents will provide indication on the affected Emergency Condenser. Field reports will provide indication of steam coming from the EC vent on the North side of the Reactor Building. N1-ARP-K1 provides direction to isolate the Emergency Condenser.* |
| *Measurable Performance Standard:* | *The crew will manually isolate the EC's will provide observable actions for the evaluation team.* ***Bounding Criteria:*** *Time based on Operations Representative recommendation to allow the crew time to evaluate the validity of the tube leak and to take the actions required in N1-ARP-K1-1-4.* |
| *Performance Feedback:* | *Emergency Condenser pressure lowering, lowering Rad Levels on the EC vents will provide performance feedback regarding the successful isolation of the Emergency Condenser.* |

| ***Critical Task (#2)*** | **Given a coolant leak inside the Containment, lowering Reactor water level, and failure of injection systems to automatically inject to the Reactor, the crew will inject to the Reactor to restore and maintain Reactor water level above -84", in accordance with N1-EOP-RPV. Reactor water level must not be below -84" for more than 10 minutes.** |
|---|---|
| *CT-2.0 Justification:* | |
| *Safety Significance* | *Maintaining Reactor water level above -84 inches ensures adequate core cooling through the preferred method of core submergence. This protects the integrity of the fuel cladding.* |
| *Success Path:* | *The crew will inject with all available low-pressure preferred and alternate injection sources (Details E1 and E2) in accordance with N1-EOP-RPV (1-2).* |
| *Initiating Cue:* | *Multiple Reactor water level indicators and annunciators will provide indications of lowering Reactor water level. N1-EOP-RPV (1-2) provides multiple procedure steps directing injection with preferred and alternate injection systems.* |
| *Measurable Performance Standard:* | *Manipulation of pumps and/or valves in the preferred or alternate injection system(s) will provide observable actions for the evaluation team. As well as operating ERVs for depressurization.* ***Bounding Criteria:*** *Time requirement based on Operations representative recommendation to allow RPV pressure to lower to within the capacity of the low-pressure injection sources.* |
| *Performance Feedback:* | *Multiple Reactor water level indicators and annunciators will provide performance feedback regarding the success of injection with preferred and alternate injection systems.* |

- Length
  - ~60 minutes
- Mitigation Strategy Code
  - DMS-RL-01, LOCA, RPV Level maintained above TAF using preferred injection systems; containment parameters controlled: no blowdown
- Technical Specifications
  - TS 3.3.7.b, 15 day LCO
- EAL Classification
  - Alert per EAL FA1 – ANY Loss or ANY Potential Loss of either Fuel Clad or RCS
- Special Orders
  - None

- **Initial Conditions**
  - IC Number
    - IC-243

Presets / With Triggers

Malfunctions

| | | |
|---|---|---|
| 1) | **RD42**, DOUBLE NOTCH CONTROL RODS, FINAL = True | **TRG 1** |
| 2) | **RM73A**, PRM Rx Bldg Vent Exh Rad Mon 11 – Normal Malfunction, FINAL = 100% | **TRG 2** |
| 3) | **HV04**, RB Ventilation Fails to Isolate, FINAL = True | **Inserted** |
| 4) | **NM11A,** IRM Channel 11 Failure – Downscale, FINAL = True | **TRG3** |
| 5) | **EC06A**, Emergency Condenser Tube Leak 111, FINAL = 75, RT = 3 min | **TRG 4** |
| 6) | **CW05A**, Turbine Building Closed Loop Cooling Pump Trip 11, FINAL = True | **TRG 5** |
| 7) | **CW21B**, TBCLC Pump #12 Auto-Start Failure, FINAL = True | **Inserted** |
| 8) | **CW05B**, Turbine Building Closed Loop Cooling Pump Trip 12, FINAL = 12, DT = 5 min | **TRG 6** |
| 9) | **RR29**, RR Loop Rupture on Pump 15 Suction Line (LOCA), FINAL = 45%, INITIAL = 15%, DT = 3 min, RT = 5 min | **TRG 7** |
| 10) | **CS06**, CS Pumps Failure to Auto Start, FINAL = True | **TRG 7** |
| 11) | **FW01A**, Condensate Pump Trip 11, FINAL = True | **TRG 7** |
| 12) | **FW01B**, Condensate Pump Trip 12, FINAL = True | **TRG 7** |
| 13) | **FW01C**, Condensate Pump Trip 13, FINAL = True | **TRG 7** |

Remotes

1) None

Overrides

| | | |
|---|---|---|
| 1) | **DI-5265**, INOP 2L6/202-31A POS A, FINAL = Off | **Inserted** |

Annunciators

None

Event Triggers

| Event # | Event Action | Command |
|---|---|---|
| **TRG 1** – Inserts control rod double notching malfunction when control rod 10-23 is at position 02. | rdvposd(82)==256 | **Blank** |
| **TRG 15** – Deletes control rod double notching malfunction when control rod 10-23 is at position 06. | rdvposd(82)==320 | dmf rd42 |
| **TRG 6** – Activates when TBCLC Pump #12 switch is taken to "START" | zdcwtbst(2)==1 | **Blank** |
| **TRG 7,** Inserts LOCA on Recirc L 15 and closes FWIV's when Mode Switch is taken to SHUTDOWN | zdrpstdn==1 | **Blank** |
| | | |

Equipment Out of Service

1) Containment Spray Pump #111 is out of service with an info/caution tag

Support Documentation

1) ReMA for conducting a Reactor Startup marked up to the rod currently selected on the Core Display (Page 4)
2) Copy of N1-OP-43A marked up to Step E.3.4
3) Copy of N1-OP-16 with P&L's and section E.3.0

Miscellaneous

1) Verify the HUR screen is active on R-time and selected to #11 RRP suction
2) Set HUR limit to $75^0$F/hr
3) Verify RWM sequence Editor set to A2UP, Simulator Startup – 2023 to 2025 is set to ACTIVE
4) Set RPV water level is in the low end of the band of 75" to 85" on the Yarways.

## SHIFT TURNOVER INFORMATION

ON COMING SHIFT: □ N ■ D DATE: Today

**PART I: To be performed by the oncoming Operator before assuming the shift.**

- Control Panel Walkdown (all panels) (SRO, ROs)

**PART II: To be reviewed by the oncoming Operator before assuming the shift.**

- LCO Status (SRO)
- Shift Turnover Information Sheet

Evolutions/General Information/Equipment Status:

- Reactor power is approximately 1%.

- Containment Spray Pump #111 is out of service for maintenance.
- 15 day LCO 3.3.7.b is in effect for Containment Spray

**PART III: Remarks/Planned Evolutions:**

- Place Feedwater Booster Pump #13 in service per N1-OP-16, Section E.3.0. An Equipment Operator has already verified that cooling water is available to all Feedwater Booster Pumps and lubrication is at the High level for all Feedwater Booster Pumps.
- Then, establish a $75^0$F/Hr HUR per N1-OP-43A and the ReMA provided. Reactor Engineering has directed single notch withdrawal on page 4 of the S/U sequence..
- RPV water level is being maintained 75” to 85” on the Yarways during reactor startup

## Event #1: Place Feedwater Booster Pump #13 in service

| | |
|---|---|
| **Event Information** | • Reactor Power is at approximately 1% power<br>• The crew will place Feedwater Booster Pump in Service per N1-OP-16. |

| | |
|---|---|
| | **SRO**<br>• Directs placing Feedwater Booster Pump (FWBP) #13 in service per N1-OP-16<br>• Provides oversight for evolution |
| | **BOP**<br>• Acknowledges direction from SRO<br>• References N1-OP-16, Section E.3 |
| | **ATC**<br>• Monitors plant parameters |
| **Note:**<br>The crew should realize that cooling water and lubrication for starting FWBP's has already been completed per the Turnover Sheet.<br><br>**Role Play:** If directed to verify cooling water and lubrication levels **wait** 3 minutes and **report** that cooling water is available for FWBP #13, and lubrication levels are at the High level mark.<br><br>**Role Play:** If directed to stand by FWBP #13 to verify pump start and proper oil levels and oil flow, **wait** until the operator starts FWBP #13 and **report** 5 – 7 seconds later that you have a good pump start on FWBP #13 and oil levels are in normal range and oil flows are normal.<br><br>**Role Play:** If contacted to verify proper pump start for FWBP #13, **wait** 2 minutes and **report** that we have a good pump start for FWBP #13. | **BOP**<br>• Makes a plant announcement for starting FWBP #13<br>• Verifies a Condensate pump is running<br>• Verifies Condensate temperature leaving Hotwell (A369) is $>35^0$F<br>• Verifies the annunciator H3-3-6, REACTOR FW BOOSTER P 13 TRIP OL SUCTION is clear<br>• Verifies computer point A024 indicates NORM for FWBP #13<br>• Verifies the following:<br>o Confirms Cooling water is available<br>o Confirms lubrication is at the High level for the Standby pumps.<br>• Starts FWBP #13<br>o Confirms system pressure and pump amps indicate system is stable.<br>• Confirms pump oil levels in normal range and oil return flow normal.<br>• Maintains vessel level using FWP 11(12) BYPASS VALVE controller |

## Events #2 and #3: Raise Reactor Power with control rods to establish a $75^0$F/hr HUR, Control Rod Double Notch

| | |
|---|---|
| **Event Information** | • Reactor power is approximately 1%<br>• The crew raised Reactor power with control rods to establish a $75^0$F/hr HUR<br>• One rod will double notch |

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| **Note:** Lead examiner may move to the next event once sufficient action is observed on the reactivity manipulation. | **SRO**<br>• Directs power ascension with control rods in accordance with N1-OP-43A and the Reactivity Maneuver Instruction (REMA) to establish a $75^0$F/hr HUR<br>• Provides oversight of reactivity maneuver<br>• Acknowledges control rod double notch<br>• Directs execution of N1-OP-5, Section H.9.0 |
| | **ATC**<br>• Acknowledges direction from SRO<br>o Raises reactor power to re-establish a $75^0$F/hr HUR per N1-OP-43A, and rod sequence sheet provided<br>o Verifies Control Rod Power is On<br>o Depresses Rod Select pushbutton for the control rod<br>o Places 4S1 Control Rod Movement switch to the rod out Notch position<br>• Observes IRM and APRM response<br>o Range IRM switches as follows:<br>- 0-125 scale between 25-75 nominal<br>- 0-40 scale between 8-24 nominal<br>• Observes Reactor Vessel temperature to ensure a $100^0$F/hr HUR is not exceeded<br>o Monitors Recirculation flow<br>• Monitors Feedwater flow and RPV water level<br>• Observes power-to-flow map restrictions |
| | **BOP**<br>• Monitors plant parameters<br>• Peer checks ATC<br>• Peer Checks HUR |

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| **Event 2 & 3 continued**<br><br>**Note:**<br><br>Verify **TRG 1 goes active** when control rod 42-31 is at position 16 and **inserts malfunction**:<br><br>**RD42,** DOUBLE NOTCH CONTROL RODS, FINAL = True<br><br>**TRG 1**<br><br>*Expected annunciator:*<br><br>*F3-4-4, ROD BLOCK*<br><br>Verify **TRG 15 goes active** when control rod 42-31 is at position 20 and **deletes malfunction RD42**.<br><br>**Note:** Initiate the next event shortly after rod withdrawals commence. It is not the intention of the scenario to cause IRMs to be shifted to range 10. | **RO**<br><br>• Selects next control rod<br>• Recognizes/reports control rod 10-23 double notched from position 02 to 06<br>• Executes N1-OP-5 section H.9.0<br>• Inserts control rod 10-23 from position 06 to 04 |

# Event #4: RB Vent Rad Monitor Fails Upscale, RB Vent Fails to Isolate

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| **Event Information** | • Reactor Power at approximately 1%.<br>• Reactor Building Ventilation Radiation Monitor, Channel 11, fails upscale<br>• Crew will determine that Reactor Building Ventilation failed to isolate<br>• Crew will take actions to isolate Reactor Building Ventilation and start RB Emergency Ventilation<br>• SRO will make Technical Specification determination for the failed instrument |

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| When directed by examiner, **insert** the following **malfunction**:<br><br>**RM73A,** PRM Rx Bldg Vent Exh Rad Mon 11 – Normal Malfunction, FINAL = 100%<br><br>**TRG 2**<br><br>Verify the following **malfunction** and **Override** is **preset**:<br><br>**HV04**, RB Ventilation Fails to Isolate, FINAL = True<br><br>**DI-5265**, INOP 2L6/202-31A POS A, FINAL = Off<br><br>*RBVS fans trip*<br><br>*RBEVS initiates*<br><br>*RB isolation valves remain open*<br><br>*Expected Annunciators:*<br><br>*L1-4-3, RB VENT RAD MONITOR OFF NORMAL*<br><br>*L1-1-5, RB VENT EXH FAN 11- 12 TRIP*<br><br>*L1-3-6, EMER VENT CH 11 RELAY OPERATE*<br><br>*L1-4-6, EMER VENT CH 12 RELAY OPERATE* | **CREW**<br>• Recognize/reports annunciator<br>• Diagnose radiation monitor failed<br>• Recognize Failure of RB to isolate |

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| **Event 4 continued**<br><br>**Note:**<br>If the crew quickly determines the event is due to a failed signal on a single rad monitor, entry into N1-EOP-SC may be avoided.<br><br>**T.S.: 3.4.2**, Reactor Building Integrity – Isolation Valves<br><br>a. The Normal Ventilation System Isolation valves shall be operable at all times when secondary containment integrity is required.<br><br>b. If Specification 3.4.2a is not met, the actions listed below shall be taken:<br><br>1. The reactor shall be in the cold shutdown condition with ten hours<br><br>**Note**: The following Technical Specification is on in effect if the crew secures one train of RBEVS following initial start of both trains.<br><br>**T.S. 3.4.4**, Emergency Ventilation System<br><br>e. During reactor operation, including when the reactor coolant system temperature is above $215^0$F, from and after the date that one circuit of the emergency ventilation system is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such circuit is sooner made operable., provided that during such seven days all active components of the other emergency ventilation circuit shall be operable. | **SRO**<br>• Acknowledges reports<br>• Directs execution of ARPs<br>• May enter N1-EOP-SC, Secondary Containment Control<br>• Directs manual isolation of the Reactor Building<br>• Determines that the Reactor Building failed to isolate and entry into Tech Spec 3.4.2 is required<br>• Enters LCO 3.4.4.e (7 day plant shutdown LCO when one train of EVS is inoperable)<br>• Exits N1-EOP-SC, if entered, once confirmation is received that radiation levels are normal |
| | **RO**<br>• Monitors plant parameters |

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| **Event 4 continued**<br><br>**Role Play:**<br><br>If contacted as RP to determine Reactor Building radiation levels, wait 5 minutes and report that Reactor Building radiation levels are all normal.<br><br>**Note:**<br><br>The crew may close the Reactor Building Ventilation Isolation Dampers prior to referencing procedure, as a backup to a failed automatic action.<br><br>**Role Play:** If dispatched to check the status of the RB Ventilation Isolation Dampers, **wait** 3 minutes and **report** that the Isolation Dampers open. | **BOP**<br><br>• Verifies Reactor Building Radiation Monitor 11 is UPSCALE<br>• Confirms alarm on computer printout<br>• Closes RB Ventilation Isolation dampers (202-15, 202-16, 202-31, 202-32)<br>• Verifies shutdown of RB Vent Sys and startup of RBEVS per N1-EOP-HC attachment 6 or N1-OP-10 section H.2.0 |

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| **Event 4 continued** | **BOP continued**<br>Possible actions per N1-EOP-HC attachment 6:<br>• Verify the following fans started:<br>  • 202-53, EVS FAN 11<br>  • 202-33, EVS FAN 12<br>• Verify the following fan control switches in OFF:<br>  • REACTOR BLDG SUPPLY FAN 11<br>  • REACTOR BLDG SUPPLY FAN 12<br>  • REACTOR BLDG EXHAUST FAN 11<br>  • REACTOR BLDG EXHAUST FAN 12<br>• Verify open the following valves:<br>  • EM VENT EXHAUST FAN 11 OUTLET BV<br>  • 202-37, EM VENTILATION LOOP 11 INLET BV<br>  • EM VENT EXHAUST FAN 12 OUTLET BV<br>  • 202-38, EM VENTILATION LOOP 12 INLET BV<br>• Verify closed the following valves:<br>  • EM VENTILATION LOOP 11 COOLING BV<br>  • EM VENTILATION LOOP 12 COOLING BV<br>  • 202-15, REACTOR BLDG SUPPLY ISOLATION VALVE 11<br>  • 202-16, REACTOR BLDG SUPPLY ISOLATION VALVE 12<br>  • 202-32, REACTOR BUILDING EXHAUST ISOLATION VALVE 11<br>  • 202-31, REACTOR BUILDING EXHAUST ISOLATION VALVE 12 |

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| **Event 4 continued**<br><br>**Note:**<br><br>Examiner may decide to proceed with next event before one train of RBEVS is secured. | **BOP continued**<br>• Notify Rad Protection, the normal Reactor Building Emergency Ventilation system has isolated AND Reactor Building Emergency Ventilation system is in service<br>• BEFORE 30 minutes has elapsed perform ONE of the following in accordance with N1-EOP-HC, Attachment 7:<br>  • Secure the train of RBEVS associated with an inoperable EDG<br>  OR<br>  • Secure a train of RBEVS as directed by the CRS or SM<br>• WHEN RBEVS operation is NO longer required, secure RBEVS AND establish normal Reactor Building Ventilation per N1-OP-10<br><br>Possible actions per N1-EOP-HC attachment 7:<br>• Place EVS FAN 11(12) control switch in Normal-After-START (red flagged)<br>• Place EM VENTILATION LOOP 11(12) INLET BV, control switch in OPEN<br>• Place EVS FAN 12(11) control switch in PULL-TO-LOCK<br>• Notify SM to enter LCO 3.4.4.e (7 day plant shutdown LCO when one train of RBEVS in inoperable)<br>• Verify closed EM VENTILATION LOOP 12(11) INLET BV |

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| **Event 4 continued**<br><br>**Note:**<br><br>Examiner may decide to proceed with next event before one train of RBEVS is secured. | **BOP continued**<br>Possible actions per N1-OP-10 section H.2.0:<br>• Confirm the following fans started:<br>  • 202-53, EVS FAN 11<br>  • 202-33, EVS FAN 12<br>• Confirm the following fans stopped:<br>  • 202-01, REACTOR BLDG SUPPLY FAN 11<br>  • 202-02, REACTOR BLDG SUPPLY FAN 12<br>  • 202-05, REACTOR BLDG EXHAUST FAN 11<br>  • 202-06, REACTOR BLDG EXHAUST FAN 12<br>• Confirm open the following valves:<br>  • 202-34, EM VENT EXHAUST FAN 11 OUTLET BV<br>  • 202-37, EM VENTILATION LOOP 11 INLET BV<br>  • 202-35, EM VENT EXHAUST FAN 12 OUTLET BV<br>  • 202-38, EM VENTILATION LOOP 12 INLET BV<br>• Confirm closed the following valves:<br>  • 202-74, EM VENTILATION LOOP 11 COOLING BV<br>  • 202-75, EM VENTILATION LOOP 12 COOLING BV<br>  • 202-03, REACTOR BLDG SUPPLY FAN 11 INLET DAMPER<br>  • 202-04, REACTOR BLDG SUPPLY FAN 12 INLET DAMPER |

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| **Event 4 continued**<br><br>**Note:**<br><br>Examiner may decide to proceed with next event before one train of RBEVS is secured. | **BOP continued**<br>• 202-80, REACTOR BLDG EXHAUST FAN 11 OUTLET DAMPER<br>• 202-81, REACTOR BLDG EXHAUST FAN 12 OUTLET DAMPER<br>• 202-15, REACTOR BLDG SUPPLY ISOLATION VALVE 11<br>• 202-16, REACTOR BLDG SUPPLY ISOLATION VALVE 12<br>• 202-32, REACTOR BUILDING EXHAUST ISOLATION VALVE 11<br>• 202-31, REACTOR BUILDING EXHAUST ISOLATION VALVE 12<br>• Notify Rad Protection, the normal Reactor Building Emergency Ventilation system has isolated AND Reactor Building Emergency Ventilation system is in service<br>• If desired to secure one train of RBEVS then perform the following:<br>• Place EVS FAN 11(12) control switch to RED FLAG<br>• Place EM VENTILATION LOOP 11(12) INLET BV control switch to OPEN<br>• Verify flow approximately 1600 CFM<br>• Place EVS FAN 12(11) control switch to PULL TO LOCK<br>• Enter LCO 3.4.4.e (7 day plant shutdown LCO when one train of EVS is inoperable)<br>• Verify closed EM VENTILATION LOOP 12(11) INLET BV<br>• Verify in OFF position the following control switches:<br>• REACTOR BLDG SUPPLY FAN 11<br>• REACTOR BLDG SUPPLY FAN 12<br>• REACTOR BLDG EXHAUST FAN 11<br>• REACTOR BLDG EXHAUST FAN 12 |

## Event #5: IRM Downscale Failure

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| **Event Information** | • A Plant startup is in progress<br>• IRM 11 fails downscale<br>• Crew will bypass the IRM<br>• SRO will evaluate Tech Specs |

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| When directed by lead examiner, **insert malfunction**:<br>**NM11A,** IRM Channel 11 Failure – Downscale, FINAL = True<br>**TRG 3**<br>*IRM 11 downscale/INOP light lit*<br>*IRM 11 indicates downscale*<br>*Expected annunciators:*<br>*F2-3-6, IRM 11-14*<br>*F3-4-4, Rod Block* | **CREW**<br>• Acknowledges/reports annunciators<br>• Diagnoses IRM 11 failed downscale |
| | **SRO**<br>• Acknowledges reports<br>• Reviews Tech Specs<br>• Determines Tech Spec 3.6.2 tables (a) and (g) are satisfied with only one failed IRM<br>• Directs bypassing IRM 11 |
| | **ATC**<br>• Monitors plant parameters<br>• Verifies range switch proper setting<br>• Monitors IRM recorder for flux levels<br>• Verifies correct position of detector in core<br>• Enters OP-38B to bypass IRM 11<br>• Places IRM BYPASS switch in IRM 11 position<br>• Confirms IRM BYPASS light is lit on panel E |

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| **Event 5 continued** | **BOP**<br>• Executes ARP F2-3-6<br>• Confirms alarm on the computer printout<br>• Observes E panel to determine IRM 11 DOWNSCALE OR INOP light lit<br>• Monitors IRM for malfunction on G Panel<br>• Confirms IRM BYPASS light is lit on the IRM auxiliaries drawer<br>• Confirms computer printout "IRM BYPASS YES" |

# Event #6: Emergency Condenser Tube Leak

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| **Event Information** | • A tube leak in Emergency Condenser 11 causes rising rad levels |

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| When directed by lead examiner, **insert malfunction**:<br><br>**EC06A**, Emergency Condenser Tube Leak 111, FINAL = 75, RT = 3 min<br><br>**TRG 4**<br><br>*EC vent rad levels rise*<br>*Expected Annunciators:*<br>*K1-3-2, EMER CONDENSER 111-112 INLET STEAM TEMP HIGH*<br>*K1-1-2, EMER COND VENT 11 RAD MONITOR*<br>*K1-3-3, EMER COND 111-112 SHELL TEMP HIGH* | **CREW**<br>• Recognize/report rising EC 11 temperatures and vent rad monitor levels<br>• Diagnose EC 11 tube leak |
| **Role Play:**<br>If called to determine the Risk Informed Completion Time **acknowledge** the **direction**. This task will not be completed by the end of the scenario.<br><br>**T.S 3.1.3.b:** Emergency Cooling System<br>b. If one emergency cooling system becomes inoperable, Specification 3.1.3.b shall be considered fulfilled, provided that the inoperable system is returned to an operable condition within 7 days or in accordance with the Risk Informed Completion Time Program and the additional Surveillance required in 4.1.3.f is performed. | **SRO**<br>• Acknowledge reports<br>• Direct execution of ARP<br>• Direct isolation of EC 11<br>• Determine Tech Spec 3.1.3.b requires restoring the EC to service within 7 days |

| Event 6 continued | ATC<br>• Monitors plant parameters |
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| **Role Play:** IF asked as RP to evaluate dose rates on RB 340' , acknowledge the request.<br><br>**Role Play:** If asked as Chemistry to perform sampling and/or EC effluent dose assessment, acknowledge the request. | **BOP**<br>• Executes ARP K1-1-2:<br>• Confirms alarm on computer printout<br>• Confirms radiation level on Panel J<br>• Contacts RP to check dose rates at EC piping above elevation 340'<br>• Directs Shift Chem Tech to obtain an evaluate an EC shell side sample<br>• With SRO concurrence, isolates EC 11, as follows:<br>• **Closes 39-07R, EC STM ISOLATION VALVE 112 (CT-1)**<br>• **Closes 39-09R, EC STM ISOLATION VALVE 111 (CT-1)**<br>• **Verifies closed 39-05, EMERG CNDSR COND RET ISOLATION VALVE 11 (CT-1)**<br>• Closes 39-11R, EMERG CNDSR STM SUPPLY DRAIN IV 111<br>• Closes 39-12R, EMERG CNDSR STM SUPPLY DRAIN IV 112<br>• Closes 05-01R, EMERG COND VENT ISOLATION VALVE 111<br>• Closes 05-11, EMERG COND VENT ISOLATION VALVE 112<br>• References N1-OP-13 section H.10, as time permits |

# Event #7 and #8: Running TBCLC Pump Trips, TBCLC Pump Fails to Auto Start, Loss of All TBCLC

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| **Event Information** | • Crew responds to trip of TBCLC pump #11<br>• TBCLC pump 12 fails to auto start<br>• Crew will manually start TBCLC pump 12<br>• TBCLC pump 12 will trip following the manual start<br>• The crew will respond to a loss of all TBCLC<br>• Crew will scram the reactor |

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| Verify the following **malfunction** is **preset**:<br><br>**CW21B**, TBCLC Pump #12 Auto-Start Failure, FINAL = True<br><br>When directed by examiner, **insert malfunction:**<br><br>**CW05A,** Turbine Building Closed Loop Cooling Pump Trip 11, FINAL = True<br><br>**TRG 5**<br><br>*Lowering TBCLC discharge pressure to 0 psig*<br>*TBCLC Pump 11 Amps at 0*<br>*TBCLC temperature starting to rise*<br>*TBCLC Pump 12 indications of not running*<br>*Expected Annunciators:*<br>*H1-1-4, TB Cooling WTR Pump 11 Trip/Trouble*<br>*H1-3-4, Turbine BLDG. Cooling Water PRESS-TEMP Makeup Flow* | **CREW**<br>• Acknowledges/reports annunciators<br>• Reports loss of TBCLC pump #11 |
| | **SRO**<br>• Acknowledges reports<br>• Directs starting TBCLC pump 12 manually<br>• Acknowledges report of starting TBCLC pump 12<br>• Updates the crew and directs entry into N1-SOP-24.1 following trip of remaining TBCLC pump.<br>• Acknowledges report from the BOP<br>• Directs RO to scram the reactor<br>• Directs the crew to enter N1-SOP-1 |

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| **Event 7 & 8 continued** | **RO**<br>• Monitors plant parameters<br>• Dispatches an Equipment Operator to investigate trip of TBCLC 11 |
| **Role Play:** If dispatched to investigate trip of TBCLC pump 11 and/or to verify a good start on TBCLC 12, **wait** 3 minutes and **report** that there is an acrid odor coming from TBCLC 11 and the breaker on PB 11 shows overcurrent flags. If asked report good start on TBCLC 12.<br><br>**Verify** the following **malfunction inserts** when TBCLC Pump 12 is started:<br>**CW05B**, Turbine Building Closed Loop Cooling Pump Trip 12, FINAL = 12, DT = 5 min<br><br>**TRG 6**<br><br>*Lowering TBCLC discharge pressure to 0 psig*<br>*TBCLC Pump 12 Amps at 0*<br>*TBCLC temperature starting to rise*<br>*Expected Annunciators:*<br>*H1-2-4, TB Cooling WTR Pump 12 Trip/Trouble*<br>*H1-3-4, Turbine BLDG. Cooling Water PRESS-TEMP*<br>*Makeup Flow* | **BOP**<br>• Follows actions in N1-ARP-H1 for trip of TBCLC 11<br>• Recognizes and reports TBCLC pump 12 has failed to auto start<br>• Updates the crew they are manually starting TBCLC 12<br>• Will recognize and report that TBCLC 12 has tripped<br>• Acknowledges direction to enter N1-SOP-24.1 |

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| **Event 7 & 8 continued**<br>*N1-SOP-24.1 Actions* | **BOP**<br>• Follows actions in N1-ARP-H1 for trip of TBCLC 11<br>• Monitors all operating equipment coolded by TBCLC<br>• Informs the CRO that the Override directs:<br>o Scram the Reactor and concurrently enter N1-SOP-1<br>o Initiate Emergency Condensers<br>o Close MSIV's<br>o Trip Reactor Recirc Pump MG sets<br>o Start AC or DC Bearing Oil pump<br>o Place in Pull-T0-Lock:<br>- Turb Aux Oil Pump 11<br>- Turb Aux Oil Pump 12<br>- House Service Air compressor 11 |
| | **RO**<br>• Acknowledges direction from the SRO to scram the reactor and enter N1-SOP-1<br>• Depresses the REACTOR TRIP 11 and 12 pushbuttons<br>• Places Mode Switch to SHUTDOWN<br>• Provides Scram Report |

## Events #9 & #10: LOCA into Drywell, Condensate Pumps trip, Core Spray Pumps Fail to Auto-Start

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| **Event Information** | • Crew has experienced a complete loss of Turbine Building Closed Loop Cooling<br>• Reactor has been scrammed<br>• Crew is taking actions for N1-SOP-24.1<br>• Crew has entered N1-SOP-1<br>• Crew will identify Feedwater Isolation Valves failed closed<br>• Crew will identify that Core Spray Pumps have failed to auto start and will manually start the Core Spray Pumps<br>• Crew will identify a Loss of Coolant Accident into the Primary Containment and enter N1-EOP-RPV and N1-EOP-PC when Drywell pressure exceeds 3.5 psig |

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| **Verify** the following **malfunctions** have **inserted** following the Mode Switch taken to SHUTDOWN:<br><br>**FW01A**, Condensate Pump Trip 11, FINAL = True<br><br>**FW01B**, Condensate Pump Trip 12, FINAL = True<br><br>**FW01C**, Condensate Pump Trip 13, FINAL = True<br><br>**RR29**, RR Loop Rupture on Pump 15 Suction Line (LOCA), FINAL = 45%, INITIAL = 15%, DT = 3 min, RT = 5 min<br><br>**CS06**, CS Pumps Failure to Auto Start, FINAL = True<br><br>**TRG 7**<br><br>*Expected plant response:*<br>*CRD charging water pressure rises*<br>*CRD drive water pressure lowers*<br>*CRD cooling water pressure and flow lower*<br>*Expected Annunciator:*<br>*F3-1-5, CRD CHARGING WTR PRESSURE HI/LO* | **CREW**<br>• Identify and report that all condensate pumps have tripped and are unavailable<br>• Identify and report Core Spray Pumps have failed to auto start<br>• Identify and report degrading conditions in the Primary Containment due to a Loss of Coolant Accident |

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| **Event 9 & 10 continued**<br><br>**Note:** Emergency Condenser was isolated earlier in the scenario due to a tube leak. MSIVs were closed or will be closed due to actions required in N1-SOP-24.1. | **SRO**<br>• Acknowledges scram report<br>• Direct entry into N1-SOP-1, Reactor Scram<br>• Acknowledge report that all condensate pumps have tripped and are unav0ailable<br>• Directs BOP to maintain RPV water level with CRD pumps<br>• Acknowledge report that Core Spray Pumps have failed to auto start and are now running<br>• Acknowledges report that Primary Containment conditions are degrading<br>• Directs entry into N1-EOP-RPV and N1-EOP-PC when Drywell pressure exceeds 3.5 psig<br>• Directs RO to insert Core Spray jumpers per N1-EOP-1, Attachment 4.<br>• Directs execution of N1-SOP-1<br>• Directs a level band of 53" to 95" using CRD and Core Spray<br>• Directs RPV pressure controlled < 120 psig to facilitate placing Shutdown Cooling in service using Emergency Condenser 12<br>• Directs MSIVs closed for inventory control and complete loss of TBCLC<br>• Directs RO to place Containment Spray pumps in PTL<br>• Directs RO to spray the Drywell when Torus Pressure exceeds 13 psig or Drywell Average Temperature exceeds $225^0$F |

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| **Event 9 & 10 continued**<br><br>**<u>Role Play:</u>** If dispatched to investigate the trip of all three condensate pumps wait 10 minutes and report that all condensate pump breakers are tripped and all motors are hot to the touch. | **RO**<br>• Provides scram report<br>o Mode Switch in SHUTDOWN<br>o All rods are in<br>o Power is down scale<br>o RPV pressure, trend and control<br>o RPV water level, trend and control<br>• Reports Feedwater is unavailable for RPV level control due to all condensate pumps have tripped.<br>• Acknowledges direction to enter N1-SOP-1<br>• Monitor plant parameters<br>• Reports degrading conditions in Primary Containment<br>• Will take critical parameter to perform a Containment Isolation before Drywell pressure exceeds 3.5 psig<br>• Acknowledges direction to place Containment Spray Pumps in PTL<br>• Places Containment Spray pumps in PTL and notifies SRO<br>• Acknowledges direction to install Core Spray Jumpers per N1-EOP-1, Attachment 4<br>• Starts Containment Spray pumps when directed following Torus Pressure exceeding 13 psig or Drywell average temperature > $225^0$F |

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| **Event 9 & 10 continued** | **BOP**<br>• Acknowledge direction from SRO to enter N1-SOP-1<br>• Verifies RPV level is greater the 53”<br>• Verifies FWP 11 and 12 Yokogawas’ in MANUAL with zero output<br>• Resets HPCI<br>• Place desired BYPASS valve in AUTO, set at 65” to 70”<br>• Recognizes and reports that all condensate pumps have tripped and FW system is unavailable for RPV level control<br>• Acknowledges entry into N1-EOP-RPV and N1-EOP-PC due to degrading Drywell conditions<br>• Acknowledges RPV water level band of 53” to 95”<br>• Acknowledges RPV pressure controlled < 120 psig to facilitate placing Shutdown Cooling in service using Emergency Condenser 12<br>• Reports level is NOT recovering with CRD pumps<br>• Reports that RPV water level is slowly lowering due to the LOCA in the Primary Containment and will need to use Core Spray pumps to maintain RPV water level<br>• Reports Core Spray Pumps have failed to auto start and will be manually starting Core Spray pumps for RPV water level control<br>• **Starts Core Spray pumps and Core Spray Topping Pumps and restores RPV water level to 53” to 95” level band (CT-2)**<br>• Throttles Core Spray flow using Core Spray Discharge Isolation valves once Core Spray Jumpers have been installed to recover RPV Water level to 53” to 95” |

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| **Event Termination Criteria** | • RPV water level being restored to the ordered band<br>• Drywell pressure controlled < PSP utilizing Containment Spray Pumps |

**Form 4.1-BWR Boiling-Water Reactor Examination Outline**

| Facility: | Nine Mile Point 1 | K/A Catalog Rev. 3 | Rev. | 07/16/2023 | Date of Exam: | 8/12/2024 |
|---|---|---|---|---|---|---|

| Tier | Group | RO K/A Category Points | | | | | | | | | | | | SRO-Only Points | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | K1 | K2 | K3 | K4 | K5 | K6 | A1 | A2 | A3 | A4 | G* | Total | A2 | G* | Total |
| 1. Emergency and Abnormal Plant Evolutions | 1 | 3 | 3 | 4 | | | | 3 | 4 | | | 3 | 20 | 4 | 3 | 7 |
| | 2 | 1 | 1 | 1 | | | | 1 | 1 | | | 1 | 6 | 1 | 2 | 3 |
| | Tier Totals | 4 | 4 | 5 | | | | 4 | 5 | | | 4 | 26 | 5 | 5 | 10 |
| 2. Plant Systems | 1 | 2 | 2 | 3 | 2 | 1 | 2 | 3 | 3 | 3 | 3 | 2 | 26 | 2 | 3 | 5 |
| | 2 | 1 | 1 | 0 | 1 | 2 | 1 | 1 | 1 | 1 | 1 | 1 | 11 | 0 / 1 | 2 | 3 |
| | Tier Totals | 3 | 3 | 3 | 3 | 3 | 3 | 4 | 4 | 4 | 4 | 3 | 37 | 3 | 5 | 8 |

| 3. Generic Knowledge and Abilities Categories | CO | EC | RC | EM | 6 | CO | EC | RC | EM | 7 |
|---|---|---|---|---|---|---|---|---|---|---|
| | 2 | 2 | 1 | 1 | | 2 | 2 | 1 | 2 | |

| 4. Theory | Reactor Theory | Thermodynamics | 6 | |
|---|---|---|---|---|
| | 3 | 3 | | |

Notes: CO = Conduct of Operations; EC = Equipment Control; RC = Radiation Control;
EM = Emergency Procedures/Plan

* These systems/evolutions may be eliminated from the sample when Revision 2 of the K/A catalog is used to develop the sample plan.

** These systems/evolutions are only included as part of the sample (as applicable to the facility) when Revision 2 of the K/A catalog is used to develop the sample plan.

ES-4.1-BWR BWR Examination Outline (Nine Mile Point 1)

Emergency and Abnormal Plant Evolutions—Tier 1/Group 1 (RO/**SRO**)

| Item # | E/APE # / Name / Safety Function | K1 | K2 | K3 | A1 | A2 | G* | K/A Topic(s) | IR | Q# |
|---|---|---|---|---|---|---|---|---|---|---|
| 1 | (295001) (APE 1) PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION | **X** | | | | | | (295001AK1.01) Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to the PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Natural circulation | 3.7 | 1 |
| 2 | (295003) (APE 3) PARTIAL OR COMPLETE LOSS OF AC POWER | | | | | | **X** | (295003) PARTIAL OR COMPLETE LOSS OF AC POWER (G2.4.16) Knowledge of emergency and abnormal operating procedures implementation hierarchy and coordination with other support procedures or guidelines such as, operating procedures, abnormal operating procedures, or severe accident management guidelines | 3.5 | 2 |
| 3 | (295004) (APE 4) PARTIAL OR COMPLETE LOSS OF DC POWER | | | **X** | | | | (295004AK3.01) Knowledge of the reasons for the following responses or actions as they apply to PARTIAL OR COMPLETE LOSS OF DC POWER: Load shedding | 3.9 | 3 |
| 4 | (295005) (APE 5) MAIN TURBINE GENERATOR TRIP | **X** | | | | | | (295005AK1.03) Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to the MAIN TURBINE GENERATOR TRIP: Reactor level control | 3.8 | 4 |
| 5 | (295006) (APE 6) SCRAM | | **X** | | | | | (295006AK2.03) Knowledge of the relationship between the SCRAM and the following systems or components: CRD hydraulic system | 3.9 | 5 |
| 6 | (295016) (APE 16) CONTROL ROOM ABANDONMENT | | | | | | **X** | (295016) CONTROL ROOM ABANDONMENT (G2.4.3) Ability to identify post-accident instrumentation | 3.7 | 6 |
| 7 | (295018) (APE 18) PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER (CCW) | | | | **X** | | | (295018AA1.04) Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER (CCW): Reactor water cleanup system | 3.4 | 7 |
| 8 | (295019) (APE 19) PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR | | | | | **X** | | (295019AA2.01) Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: Instrument air pressure | 4.1 | 8 |
| 9 | (295021) (APE 21) LOSS OF SHUTDOWN COOLING | | | | | **X** | | (295021AA2.02) Ability to determine and/or interpret the following as they apply to LOSS OF SHUTDOWN COOLING: RHR/shutdown cooling system flow | 4.1 | 9 |
| 10 | (295023) (APE 23) REFUELING ACCIDENTS | | | **X** | | | | (295023AK3.03) Knowledge of the reasons for the following responses or actions as they apply to REFUELING ACCIDENTS: Ventilation isolation | 3.7 | 10 |
| 11 | (295024) (EPE 1) HIGH DRYWELL PRESSURE | | | | **X** | | | (295024EA1.12) Ability to operate and/or monitor the following as they apply to HIGH DRYWELL PRESSURE: Suppression pool spray (Mark I, II) | 4.2 | 11 |
| 12 | (295025) (EPE 2) HIGH REACTOR PRESSURE | | **X** | | | | | (295025EK2.01) Knowledge of the relationship between the HIGH REACTOR PRESSURE and the following systems or components: RPS | 4.2 | 12 |
| 13 | (295026) (EPE 3) SUPPRESSION POOL HIGH WATER TEMPERATURE | | | | **X** | | | (295026EA1.09) Ability to operate and/or monitor the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: Safety/relief valves | 4.1 | 13 |

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| 14 | (295028) (EPE 5) HIGH DRYWELL TEMPERATURE (MARK I AND MARK II ONLY) | | | | | | X | (295028) HIGH DRYWELL TEMPERATURE (MARK I AND MARK II ONLY) (G2.1.2) Knowledge of operator responsibilities during any mode of plant operation | 4.1 | 14 |
| 15 | (295030) (EPE 7) LOW SUPPRESSION POOL WATER LEVEL | | | X | | | | (295030EK3.05) Knowledge of the reasons for the following responses or actions as they apply to LOW SUPPRESSION POOL WATER LEVEL: Suppression pool makeup system(s) operation | 3.6 | 15 |
| 16 | (295031) (EPE 8) REACTOR LOW WATER LEVEL | X | | | | | | (295031EK1.01) Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to the REACTOR LOW WATER LEVEL: Adequate core cooling | 4.7 | 16 |
| 17 | (295037) (EPE 14) SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN | | X | | | | | (295037EK2.14) Knowledge of the relationship between the SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN and the following systems or components: RPIS (BWR 2, 3, 4, 5) | 4 | 17 |
| 18 | (295038) (EPE 15) HIGH OFFSITE RADIOACTIVITY RELEASE RATE | | | | | X | | (295038EA2.05) Ability to determine and/or interpret the following as they apply to HIGH OFFSITE RADIOACTIVITY RELEASE RATE: Emergency plan implementation | 3.6 | 18 |
| 19 | (600000) (APE 24) PLANT FIRE ON SITE | | | | | X | | (600000AA2.03) Ability to determine and/or interpret the following as they apply to PLANT FIRE ON SITE: Fire alarm | 3.8 | 19 |
| 20 | (700000) (APE 25) GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES | | | X | | | | (700000AK3.01) Knowledge of the reasons for the following responses or actions as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: Reactor and turbine trip criteria | 3.8 | 20 |
| **21** | **(295001) (APE 1) PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION** | | | | | X | | **(295001AA2.13) Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Thermal limits** | **3.9** | **76** |
| **22** | **(295003) (APE 3) PARTIAL OR COMPLETE LOSS OF AC POWER** | | | | | | X | **(295003) PARTIAL OR COMPLETE LOSS OF AC POWER (G2.4.47) Ability to diagnose and recognize trends in an accurate and timely manner using the appropriate control room reference material (reference potential)** | **4.2** | **77** |
| **23** | **(295005) (APE 5) MAIN TURBINE GENERATOR TRIP** | | | | | X | | **(295005AA2.06) Ability to determine and/or interpret the following as they apply to MAIN TURBINE GENERATOR TRIP: Feedwater temperature** | **2.9** | **78** |
| **24** | **(295024) (EPE 1) HIGH DRYWELL PRESSURE** | | | | | | X | **(295024) HIGH DRYWELL PRESSURE (G2.4.22) Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations** | **4.4** | **79** |
| **25** | **(295028) (EPE 5) HIGH DRYWELL TEMPERATURE (MARK I AND MARK II ONLY)** | | | | | | X | **(295028) HIGH DRYWELL TEMPERATURE (MARK I AND MARK II ONLY) (G2.4.14) Knowledge of general guidelines for emergency and abnormal operating procedures usage** | **4.5** | **80** |

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| **26** | **(295030) (EPE 7) LOW SUPPRESSION POOL WATER LEVEL** | | | | | **X** | | **(295030EA2.02) Ability to determine and/or interpret the following as they apply to LOW SUPPRESSION POOL WATER LEVEL: Suppression pool temperature** | **3.8** | **81** |
| **27** | **(295037) (EPE 14) SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN** | | | | | **X** | | **(295037EA2.06) Ability to determine and/or interpret the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: Reactor pressure** | **4.3** | **82** |
| K/A Category Totals: | | 3 | 3 | 4 | 3 | 8 | 6 | Group Point Total: | | 27 |

ES-4.1-BWR

BWR Examination Outline (Nine Mile Point 1)

Emergency and Abnormal Plant Evolutions—Tier 1/Group 2 (RO/**SRO**)

| Item # | E/APE # / Name / Safety Function | K1 | K2 | K3 | A1 | A2 | G* | K/A Topic(s) | IR | Q# |
|---|---|---|---|---|---|---|---|---|---|---|
| 28 | (295009) (APE 9) LOW REACTOR WATER LEVEL | | | X | | | | (295009AK3.01) Knowledge of the reasons for the following responses or actions as they apply to LOW REACTOR WATER LEVEL: Recirculation runback | 3.7 | 21 |
| 29 | (295010) (APE 10) HIGH DRYWELL PRESSURE | | X | | | | | (295010AK2.07) Knowledge of the relationship between the HIGH DRYWELL PRESSURE and the following systems or components: (Component cooling water | 3.5 | 22 |
| 30 | (295014) (APE 14) INADVERTENT REACTIVITY ADDITION | X | | | | | | (295014AK1.09) Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to the INADVERTENT REACTIVITY ADDITION: Reactor power | 4.4 | 23 |
| 31 | (295020) (APE 20) INADVERTENT CONTAINMENT ISOLATION | | | | X | | | (295020AA1.04) Ability to operate and/or monitor the following as they apply to INADVERTENT CONTAINMENT ISOLATION: SGTS/FRVS | 3.6 | 24 |
| 32 | (295032) (EPE 9) HIGH SECONDARY CONTAINMENT AREA TEMPERATURE | | | | | | X | (295032) HIGH SECONDARY CONTAINMENT AREA TEMPERATURE (G2.1.14) Knowledge of criteria or conditions that require plantwide announcements, such as pump starts, reactor trips, and mode changes | 3.1 | 25 |
| 33 | (295035) (EPE 12) SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE | | | | | X | | (295035EA2.03) Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE: Lights and alarms | 3.7 | 26 |
| **34** | **(295007) (APE 7) HIGH REACTOR PRESSURE** | | | | | | **X** | **(295007) (APE 7) HIGH REACTOR PRESSURE (G2.4.6) Knowledge of emergency and abnormal operating procedures major action categories** | **4.7** | **83** |
| **35** | **(295022) (APE 22) LOSS OF CONTROL ROD DRIVE PUMPS** | | | | | | **X** | **(295022) LOSS OF CONTROL ROD DRIVE PUMPS (G2.2.22) Knowledge of limiting conditions for operation and safety limits** | **4.7** | **84** |
| **36** | **(295029) (EPE 6) HIGH SUPPRESSION POOL WATER LEVEL** | | | | | **X** | | **(295029EA2.03) Ability to determine and/or interpret the following as they apply to HIGH SUPPRESSION POOL WATER LEVEL: Drywell/containment water level** | **3.6** | **85** |
| K/A Category Totals: | | 1 | 1 | 1 | 1 | 2 | 3 | Group Point Total: | | 9 |

ES-4.1-BWR BWR Examination Outline (Nine Mile Point 1)

Plant Systems—Tier 2/Group 1 (RO/**SRO**)

| Item # | System / Name | K1 | K2 | K3 | K4 | K5 | K6 | A1 | A2 | A3 | A4 | G* | K/A Topic(s) | IR | Q# |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 37 | (205000) (SF4 SCS) SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) | | | | | | | | **X** | | | | (205000A2.01) Ability to (a) predict the impacts of the following on the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Recirculation loop high temperature | 3.6 | 27 |
| 38 | (206000) (SF2, SF4 HPCI) HIGH PRESSURE COOLANT INJECTION SYSTEM | **X** | | | | | | | | | | | (206000K1.17) Knowledge of the physical connections and/or cause and effect relationships between the HIGH PRESSURE COOLANT INJECTION SYSTEM and the following systems: Reactor protection system | 2.8 | 28 |
| 39 | (207000) (SF4 IC) ISOLATION (EMERGENCY) CONDENSER | | | | | | **X** | | | | | | (207000K6.07) Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the ISOLATION (EMERGENCY) CONDENSER: AC electrical distribution | 3.5 | 29 |
| 40 | (209001) (SF2, SF4 LPCS) LOW PRESSURE CORE SPRAY SYSTEM | | | **X** | | | | | | | | | (209001K3.01) Knowledge of the effect that a loss or malfunction of the LOW PRESSURE CORE SPRAY SYSTEM will have on the following systems or system parameters: Reactor water level | 4.3 | 30 |
| 41 | (209001) (SF2, SF4 LPCS) LOW PRESSURE CORE SPRAY SYSTEM | | | | | **X** | | | | | | | (209001K5.02) Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the LOW PRESSURE CORE SPRAY SYSTEM: Abnormal differential pressure indication (leak detection) | 3.4 | 31 |
| 42 | (211000) (SF1 SLCS) STANDBY LIQUID CONTROL SYSTEM | | | | | | | **X** | | | | | (211000A1.05) Ability to predict and/or monitor changes in parameters associated with operation of the STANDBY LIQUID CONTROL SYSTEM including: Pump amps | 2.9 | 32 |

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|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 43 | (212000) (SF7 RPS) REACTOR PROTECTION SYSTEM | | | | | | | | | X | | | (212000A3.09) Ability to monitor automatic operation of the REACTOR PROTECTION SYSTEM including: System actuation | 4.3 | 33 |
| 44 | (215003) (SF7 IRM) INTERMEDIATE RANGE MONITOR SYSTEM | | | X | | | | | | | | | (215003K3.01) Knowledge of the effect that a loss or malfunction of the INTERMEDIATE RANGE MONITOR SYSTEM will have on the following systems or system parameters: RPS | 4.2 | 34 |
| 45 | (215004) (SF7 SRMS) SOURCE RANGE MONITOR SYSTEM | | | | | | | X | | | | | (215004A1.01) Ability to predict and/or monitor changes in parameters associated with operation of the SOURCE RANGE MONITOR SYSTEM including: Detector position | 3.3 | 35 |
| 46 | (263000) (SF6 DC) DC Electrical Distribution | | | | | | | X | | | | | (263000A1.01) Ability to predict and/or monitor changes in parameters associated with operation of the DC Electrical Distribution, including: Battery charging/discharging rate | 3.3 | 36 |
| 47 | (215005) (SF7 PRMS) AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR | | | X | | | | | | | | | (215005K3.07) Knowledge of the effect that a loss or malfunction of the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR will have on the following systems or system parameters: Rod block monitor system | 3.7 | 37 |
| 48 | (218000) (SF3 ADS) AUTOMATIC DEPRESSURIZATION SYSTEM | | | | | | | | | | X | | (218000A4.01) Ability to manually operate and/or monitor the AUTOMATIC DEPRESSURIZATION SYSTEM in the control room: ADS valves | 4.5 | 38 |
| 49 | (223002) (SF5 PCIS) PRIMARY CONTAINMENT ISOLATION SYSTEM / NUCLEAR STEAM SUPPLY SHUTOFF | | | | | | | | X | | | | (223002A2.13) Ability to (a) predict the impacts of the following on the PRIMARY CONTAINMENT ISOLATION SYSTEM / NUCLEAR STEAM SUPPLY SHUTOFF and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Leak detection system failures | 3.3 | 39 |
| 50 | (223002) (SF5 PCIS) PRIMARY CONTAINMENT ISOLATION SYSTEM / NUCLEAR STEAM SUPPLY SHUTOFF | | | | | | X | | | | | | (223002K6.09) Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the PRIMARY CONTAINMENT ISOLATION SYSTEM / NUCLEAR STEAM SUPPLY SHUTOFF: Plant pneumatic systems | 3.2 | 40 |

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|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 51 | (239002) (SF3 SRV) SAFETY RELIEF VALVES | | | | | | | | | X | | | (239002A3.01) Ability to monitor automatic operation of the SAFETY RELIEF VALVES including: SRV operation after ADS actuation | 4.1 | 41 |
| 52 | (239002) (SF3 SRV) SAFETY RELIEF VALVES | | | | | | | | | | | X | (239002) SAFETY RELIEF VALVES (G2.1.45) Ability to identify and interpret diverse indications to validate the response of another indication | 4.3 | 42 |
| 53 | (259002) (SF2 RWLCS) REACTOR WATER LEVEL CONTROL SYSTEM | | X | | | | | | | | | | (259002K2.03) REACTOR WATER LEVEL CONTROL SYSTEM Knowledge of electrical power supplies to the following: Feedwater controllers | 3.1 | 43 |
| 54 | (510000) (SF4 SWS*) SERVICE WATER SYSTEM | X | | | | | | | | | | | (510000K1.13) SERVICE WATER SYSTEM Knowledge of the physical connections and/or cause and effect relationships between the Service Water System and the following systems: Feedwater system | 2.7 | 44 |
| 55 | (261000) (SF9 SGTS) STANDBY GAS TREATMENT SYSTEM | | X | | | | | | | | | | (261000K2.04) STANDBY GAS TREATMENT SYSTEM Knowledge of electrical power supplies to the following: Standby gas treatment system heaters | 2.9 | 45 |
| 56 | (262001) (SF6 AC) AC ELECTRICAL DISTRIBUTION | | | | | | | | | X | | | (262001A3.03) Ability to monitor automatic operation of the AC ELECTRICAL DISTRIBUTION including: Load shedding | 3.8 | 46 |
| 57 | (262001) (SF6 AC) AC ELECTRICAL DISTRIBUTION | | | | | | | | | | X | | (262001A4.01) Ability to manually operate and/or monitor the AC ELECTRICAL DISTRIBUTION in the control room: Breakers and disconnects | 3.7 | 47 |
| 58 | (262002) (SF6 UPS) UNINTERRUPTABLE POWER SUPPLY (AC/DC) | | | | X | | | | | | | | (262002K4.01) Knowledge of UNINTERRUPTABLE POWER SUPPLY (AC/DC) design features and/or interlocks that provide for the following: Transfer of power supplies | 3.5 | 48 |
| 59 | (263000) (SF6 DC) DC ELECTRICAL DISTRIBUTION | | | | X | | | | | | | | (263000K4.03) Knowledge o) DC ELECTRICAL DISTRIBUTION design features and/or interlocks that provide for the following: Ground detection | 2.9 | 49 |

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|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 60 | (264000) (SF6 EGE) EMERGENCY GENERATORS (DIESEL/JET) | | | | | | | | | | X | | (264000A4.07) Ability to manually operate and/or monitor the EMERGENCY GENERATORS (DIESEL/JET) in the control room: Transfer of control to local/remote panels | 3.4 | 50 |
| 61 | (300000) (SF8 IA) INSTRUMENT AIR SYSTEM | | | | | | | | | | | X | (300000) INSTRUMENT AIR SYSTEM (291001K1.04) Valves - The failed-valve positions for different operators (open, closed, and as-is positions; spring-loaded valves; hydraulically/ pneumatically controlled valves; electric motor-driven valves) | 2.8 | 51 |
| 62 | (400000) (SF8 CCS) COMPONENT COOLING WATER SYSTEM | | | | | | | | X | | | | (400000A2.08) Ability to (a) predict the impacts of the following on the COMPONENT COOLING WATER SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Residual heat removal system heat exchanger tube leak | 3.5 | 52 |
| **63** | **(205000) (SF4 SCS) SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE)** | | | | | | | | **X** | | | | **(205000A2.05) Ability to (a) predict the impacts of the following on the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: System isolation** | **4** | **86** |
| **64** | **(212000) (SF7 RPS) REACTOR PROTECTION SYSTEM** | | | | | | | | **X** | | | | **(212000A2.18) Ability to (a) predict the impacts of the following on the REACTOR PROTECTION SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: SCRAM air header low pressure** | **3.9** | **87** |
| **65** | **(215003) (SF7 IRM) INTERMEDIATE RANGE MONITOR SYSTEM** | | | | | | | | | | | **X** | **(215003) INTERMEDIATE RANGE MONITOR SYSTEM (G2.2.21) Knowledge of pre- and post-maintenance operability requirements** | **4.1** | **88** |

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|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| **66** | **(218000) (SF3 ADS) AUTOMATIC DEPRESSURIZATION SYSTEM** | | | | | | | | | | | **X** | **(218000) AUTOMATIC DEPRESSURIZATION SYSTEM (G2.2.19) Knowledge of maintenance work order requirements** | **3.4** | **89** |
| **67** | **(262002) (SF6 UPS) UNINTERRUPTABLE POWER SUPPLY (AC/DC)** | | | | | | | | | | | **X** | **(262002) UNINTERRUPTABLE POWER SUPPLY (AC/DC) (G2.2.36) Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operation** | **4.2** | **90** |
| K/A Category Totals: | | 2 | 2 | 3 | 2 | 1 | 2 | 3 | 5 | 3 | 3 | 5 | Group Point Total: | | 31 |

ES-4.1-BWR BWR Examination Outline (Nine Mile Point 1)

Plant Systems—Tier 2/Group 2 (RO/**SRO**)

| Item # | System / Name | K1 | K2 | K3 | K4 | K5 | K6 | A1 | A2 | A3 | A4 | G* | K/A Topic(s) | IR | Q# |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 68 | (202001) (SF1, SF4 RS) RECIRCULATION SYSTEM | | **X** | | | | | | | | | | (202001K2.01) RECIRCULATION SYSTEM Knowledge of electrical power supplies to the following: Recirculation pumps | 3.5 | 53 |
| 69 | (216000) (SF7 NBI) NUCLEAR BOILER INSTRUMENTATION | | | | | | | **X** | | | | | (216000A1.05) Ability to predict and/or monitor changes in parameters associated with operation of the NUCLEAR BOILER INSTRUMENTATION including: Lights and alarms | 3.3 | 54 |
| 70 | (223001) (SF5 PCS) PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES | **X** | | | | | | | | | | | (223001K1.16) Knowledge of the physical connections and/or cause and effect relationships between the PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES and the following systems: Containment and drywell atmosphere monitoring system | 3.5 | 55 |
| 71 | (226001) (SF5 RHR CSS) RHR/LPCI: CONTAINMENT SPRAY MODE SYSTEM MODE | | | | | | | | | **X** | | | (226001A3.07) Ability to monitor automatic operation of the RHR/LPCI: CONTAINMENT SPRAY MODE SYSTEM MODE including: Pump start | 3.9 | 56 |
| 72 | (233000) (SF9 FPCCU) FUEL POOL COOLING/CLEANUP | | | | | **X** | | | | | | | (233000K5.03) Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the FUEL POOL OOLING/CLEANUP: Spent fuel decay heat generation | 3.3 | 57 |
| 73 | (239001) (SF3, SF4 MRSS) MAIN AND REHEAT STEAM SYSTEM | | | | | | **X** | | | | | | (239001K6.08) Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the MAIN AND REHEAT STEAM SYSTEM: Main condenser vacuum | 3.5 | 58 |
| 74 | (245000) (SF4 MTGEN) MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS | | | | | | | | **X** | | | | (245000A2.02) Ability to (a) predict the impacts of the following on the MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Loss of lube oil | 3.8 | 59 |

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|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 75 | (259001) (SF2 FWS) FEEDWATER SYSTEM | | | | X | | | | | | | | (259001K4.06) Knowledge of FEEDWATER SYSTEM design features and/or interlocks that provide for the following: RFP lubrication | 3 | 60 |
| 76 | (268000) (SF9 RW) RADWASTE SYSTEM | | | | | | | | | | | X | (268000) RADWASTE SYSTEM (G2.2.43) Knowledge of the process used to track inoperable alarms | 3 | 61 |
| 77 | (271000) (SF9 OG) OFFGAS SYSTEM | | | | | | | | | | X | | (271000A4.09) Ability to manually operate and/or monitor the OFFGAS SYSTEM in the control room: Offgas system controls /components | 3 | 62 |
| 78 | (290003) (SF9 CRV) CONTROL ROOM VENTILATION | | | | | X | | | | | | | (290003K5.04) Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the CONTROL ROOM VENTILATION: Control room habitability | 3.7 | 63 |
| **79** | **(272000) (SF7 RMS) Radiation Monitoring System** | | | | | | | | | | | **X** | **(272000) (SF7 RMS) Radiation Monitoring System (G2.3.12) Knowledge of radiological safety principles and procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, or alignment of filters** | **3.7** | **91** |
| **80** | **(256000) (SF2 CDS) CONDENSATE SYSTEM** | | | | | | | | **X** | | | | **(256000A2.01) Ability to (a) predict the impacts of the following on the CONDENSATE SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Pump trips** | **3.9** | **92** |
| **81** | **(290002) (SF4 RVI) REACTOR VESSEL INTERNALS** | | | | | | | | | | | **X** | **(290002) (SF4 RVI) REACTOR VESSEL INTERNALS (G2.4.21) Knowledge of the parameters and logic used to assess the status of emergency operating procedures critical safety functions or shutdown critical safety functions** | **4.6** | **93** |
| K/A Category Totals: | | 1 | 1 | 0 | 1 | 2 | 1 | 1 | 2 | 1 | 1 | 3 | Group Point Total: | | 14 |

**Form 4.1-COMMON Common Examination Outline**

| ES-4.1-COMMON | COMMON Examination Outline (Nine Mile Point 1) | | |
|---|---|---|---|
| Facility: | Nine Mile Point 1 | Date of Exam: | 8/12/2024 |

**Generic Knowledge and Abilities Outline (Tier 3) (RO/SRO)**

| Category | K/A # | Topic | Item # | RO IR | RO Q# | SRO-Only IR | SRO-Only Q# |
|---|---|---|---|---|---|---|---|
| 1. Conduct of Operations | G2.1.21 | (G2.1.21) Ability to verify that a copy of a controlled procedure is the proper revision | 82 | 3.5 | 64 | | |
| | G2.1.32 | (G2.1.32) Ability to explain and apply system precautions, limitations, notes, or cautions | 83 | 3.8 | 65 | | |
| | **G2.1.23** | **(G2.1.23) Ability to perform general or normal operating procedures during any plant condition** | **84** | | | **4.4** | **94** |
| | **G2.1.42** | **(G2.1.42) Knowledge of new and spent fuel movement procedures (SRO Only)** | **85** | | | **3.4** | **95** |
| | Subtotal | | | N/A | 2 | N/A | **2** |
| 2. Equipment Control | G2.2.2 | (G2.2.2) Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels | 86 | 4.6 | 66 | | |
| | G2.2.38 | (G2.2.38) Knowledge of conditions and limitations in the facility license | 87 | 3.6 | 67 | | |
| | **G2.2.6** | **(G2.2.6) Knowledge of the process for making changes to procedures** | **88** | | | **3.6** | **96** |
| | **G2.2.37** | **(G2.2.37) Ability to determine operability or availability of safety-related equipment (SRO Only)** | **89** | | | **4.6** | **97** |
| | Subtotal | | | N/A | 2 | N/A | **2** |
| 3. Radiation Control | G2.3.11 | (G2.3.11) Ability to control radiation releases | 90 | 3.8 | 68 | | |
| | **G2.3.6** | **(G2.3.6) Ability to approve liquid or gaseous release permits** | **91** | | | **3.8** | **98** |
| | | | | | | | |
| | Subtotal | | | N/A | 1 | N/A | **1** |
| 4. Emergency Procedures / Plan | G2.4.45 | (G2.4.45) Ability to prioritize and interpret the significance of each annunciator or alarm | 92 | 4.1 | 69 | | |
| | **G2.4.29** | **(G2.4.29) Knowledge of the emergency plan implementing procedures** | **93** | | | **4.4** | **99** |
| | **G2.4.51** | **(G2.4.51) Knowledge of emergency operating procedure exit conditions (e.g., emergency condition no longer exists or severe accident guideline entry is required)** | **94** | | | **4.0** | **100** |
| | Subtotal | | | N/A | 1 | N/A | **2** |
| **Tier 3 Point Total** | | | | N/A | 6 | N/A | **7** |

## Form 4.1-COMMON Common Examination Outline

| ES-4.1-COMMON | COMMON Examination Outline (Nine Mile Point 1) | |
|---|---|---|
| Facility: Nine Mile Point 1 | | Date of Exam: 8/12/2024 |

**Theory (Tier 4) (RO)**

| Category | K/A # | Topic | Item # | RO IR | RO Q# |
|---|---|---|---|---|---|
| Reactor Theory | 292005 | (292005K1.10) Control Rods - State the purpose of flux shaping | 95 | 3.3 | 70 |
| | 292004 | (292004K1.10) REACTIVITY COEFFICIENTS) Define the void coefficient of reactivity | 96 | 3.2 | 71 |
| | 292007 | (292007K1.03) FUEL DEPLETION AND BURNABLE POISONS Given a curve of K-effective versus core age, state the reasons for maximum, minimum, and inflection points | 97 | 2.7 | 72 |
| | Subtotal | | | | 3 |
| Thermodynamics | 293003 | (293003K1.16) STEAM Define the following term: Subcooling | 98 | 2.8 | 73 |
| | 293006 | (293006K1.05) FLUID STATICS AND DYNAMICS Explain operational implications of fluid/water hammer | 99 | 3.3 | 74 |
| | 293010 | (293010K1.06) BRITTLE FRACTURE AND VESSEL THERMAL STRESS State the operational concerns of uncontrolled cooldown | 100 | 4.1 | 75 |
| | Subtotal | | | | 3 |
| **Tier 4 Point Total** | | | | N/A | 6 |

**Form 4.1-1 Record of Rejected Knowledge and Abilities**

Refer to Examination Standard (ES)-4.2, “Developing Written Examinations,” Section B.3, for deviations from the approved written examination outline.

| Tier/Group | Randomly Selected K/A | Reason for Rejection |
|---|---|---|
| 1 / 1 | Question #2<br>295003 PARTIAL OR COMPLETE LOSS OF AC POWER<br>2.4.23 - Knowledge of the bases for prioritizing emergency operating procedures implementation | An acceptable question could not be developed for the randomly sampled generic K/A at the RO license level due limited tie to the given topic.<br>Randomly reselected K/A 295003 PARTIAL OR COMPLETE LOSS OF AC POWER 2.4.16 - Knowledge of emergency and abnormal operating procedures implementation hierarchy and coordination with other support procedures or guidelines such as, operating procedures, abnormal operating procedures, or severe accident management guidelines. |
| 1 / 1 | Question #10<br>295023 REFUELING ACCIDENTS<br>AK3.01 - Knowledge of the reasons for the following responses or actions as they apply to REFUELING ACCIDENTS: Personnel evacuation | An acceptable question could not be developed for the randomly sampled K/A due to low level of difficulty of the topic.<br>Randomly reselected K/A 295023 REFUELING ACCIDENTS AK3.03 - Knowledge of the reasons for the following responses or actions as they apply to REFUELING ACCIDENTS: Ventilation isolation. |

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| 1 / 1 | Question #13<br><br>295026 SUPPRESSION POOL HIGH WATER TEMPERATURE<br><br>EA1.04 - Ability to operate and/or monitor the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: HPCI | An acceptable question could not be developed for the randomly sampled K/A due to lack of a tie between HPCI and the Suppression Pool at this facility.<br><br>Randomly reselected K/A 295026 SUPPRESSION POOL HIGH WATER TEMPERATURE EA1.09 - Ability to operate and/or monitor the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: Safety/relief valves. |
| 2 / 1 | Question #30<br><br>209001 LOW PRESSURE CORE SPRAY SYSTEM<br><br>K3.05 - Knowledge of the effect that a loss or malfunction of the LOW PRESSURE CORE SPRAY SYSTEM will have on the following systems or system parameters: Drywell cooling | An acceptable question could not be developed for the randomly sampled K/A due to lack of a tie between Core Spray and Drywell Cooling at this facility.<br><br>Randomly reselected K/A 209001 LOW PRESSURE CORE SPRAY SYSTEM K3.01 - Knowledge of the effect that a loss or malfunction of the LOW PRESSURE CORE SPRAY SYSTEM will have on the following systems or system parameters: Reactor water level. |

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| 2 / 1 | Question #31<br><br>209001 LOW PRESSURE CORE SPRAY SYSTEM<br><br>K5.07 - Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the LOW PRESSURE CORE SPRAY SYSTEM: Adequate core cooling | An acceptable question could not be developed for the randomly sampled K/A without oversampling adequate core cooling topics across the exam.<br><br>Randomly reselected K/A 209001 LOW PRESSURE CORE SPRAY SYSTEM K5.02 - Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the LOW PRESSURE CORE SPRAY SYSTEM: Abnormal differential pressure indication (leak detection). |
| 2 / 1 | Question #36<br><br>215004 SOURCE RANGE MONITOR SYSTEM<br><br>A1.07 - Ability to predict and/or monitor changes in parameters associated with operation of the SOURCE RANGE MONITOR SYSTEM including: SRMS count rate and period | An acceptable question could not be developed for the randomly sampled K/A without oversampling SRM topics.<br><br>Randomly reselected K/A 263000 DC Electrical Distribution A1.01 - Ability to predict and/or monitor changes in parameters associated with operation of the DC Electrical Distribution, including: Battery charging/discharging rate. |

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| 2 / 1 | Question #37<br><br>215005 AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR<br><br>K3.09 - Knowledge of the effect that a loss or malfunction of the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR will have on the following systems or system parameters: Recirculation flow control system | An acceptable question could not be developed for the randomly sampled K/A due to lack of an effect of a loss/malfunction of APRMs/LPRMs on Recirc flow control.<br><br>Randomly reselected K/A 215005 AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR K3.07 - Knowledge of the effect that a loss or malfunction of the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR will have on the following systems or system parameters: Rod block monitor system. |
| 2 / 1 | Question #42<br><br>239002 SAFETY RELIEF VALVES<br><br>2.1.37 - Knowledge of procedures, guidelines, or limitations associated with reactivity management | An acceptable question could not be developed for the randomly sampled K/A due to lack of a tie between the system and reactivity management.<br><br>Randomly reselected K/A 239002 SAFETY RELIEF VALVES 2.1.45 - Ability to identify and interpret diverse indications to validate the response of another indication. |

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| 2 / 1 | Question #45<br>261000 STANDBY GAS TREATMENT SYSTEM<br>K2.02 - Knowledge of electrical power supplies to the following: Standby gas treatment system motor-operated valves/dampers | An acceptable question could not be developed for the randomly sampled K/A due to lack of motor-operated valves/dampers in this system at this facility.<br>Randomly reselected K/A 261000 STANDBY GAS TREATMENT SYSTEM K2.04 - Knowledge of electrical power supplies to the following: Standby gas treatment system heaters. |
| 2 / 1 | Question #46<br>262001 AC ELECTRICAL DISTRIBUTION<br>A3.05 - Ability to monitor automatic operation of the AC ELECTRICAL DISTRIBUTION including: Synchronization of AC sources | An acceptable question could not be developed for the randomly sampled K/A due to lack of automatic features related to synchronization of AC sources at this facility.<br>Randomly reselected K/A 262001 AC ELECTRICAL DISTRIBUTION A3.03 - Ability to monitor automatic operation of the AC ELECTRICAL DISTRIBUTION including: Load shedding. |
| 2 / 1 | Question #51<br>300000 INSTRUMENT AIR SYSTEM<br>291004 PUMPS (POSITIVE DISPLACEMENT)<br>K1.25 - Safety procedures and precautions associated with positive displacement pumps | An acceptable question could not be developed for the randomly sampled K/A due to lack of pumps in the Instrument Air system.<br>Randomly reselected K/A 300000 INSTRUMENT AIR SYSTEM 291001 Valves K1.04 - The failed-valve positions for different operators (open, closed, and as-is positions; spring-loaded valves; hydraulically/pneumatically controlled valves; electric motor-driven valves). |

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| 4 | Question #70<br>292001 NEUTRONS<br>Identify characteristics of good moderators | An acceptable question could not be developed for the randomly sampled K/A due low operational relevance of the topic.<br>Randomly reselected K/A 292005 Control Rods K1.10 - State the purpose of flux shaping. |
| 4 | Question #71<br>292004 REACTIVITY COEFFICIENTS<br>K1.15 - Explain the differences between reactivity coefficients and reactivity defects | An acceptable question could not be developed for the randomly sampled K/A without testing minutia.<br>Randomly reselected K/A 292004 REACTIVITY COEFFICIENTS K1.10 - Define the void coefficient of reactivity. |
| 1 / 1 | Question #77<br>295003 PARTIAL OR COMPLETE LOSS OF AC POWER<br>2.2.41 - Ability to obtain and interpret station electrical and mechanical drawings (reference potential) | An acceptable question could not be developed at the SRO level and within Tier 1 criteria for the randomly sampled generic K/A.<br>Randomly reselected K/A 295003 PARTIAL OR COMPLETE LOSS OF AC POWER 2.4.47 - Ability to diagnose and recognize trends in an accurate and timely manner using the appropriate control room reference material (reference potential). |

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| 1 / 1 | Question #79<br>295024 HIGH DRYWELL PRESSURE<br>2.2.17 - Knowledge of the process for managing maintenance activities during power operations, such as risk assessments, work prioritization, and coordination with the transmission system operator | An acceptable question could not be developed for the combination of randomly sampled evolution and generic K/A.<br>Randomly reselected K/A 295024 HIGH DRYWELL PRESSURE 2.4.22 - Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations. |
| 1 / 1 | Question #80<br>295028 HIGH DRYWELL TEMPERATURE<br>2.4.31 - Knowledge of annunciator alarms, indications, or response procedures | An acceptable question could not be developed for the combination of randomly sampled evolution and generic K/A at the SRO level.<br>Randomly reselected K/A 295028 HIGH DRYWELL TEMPERATURE 2.4.14 - Knowledge of general guidelines for emergency and abnormal operating procedures usage. |
| 1 / 2 | Question #83<br>295007 HIGH REACTOR PRESSURE<br>2.4.42 - Knowledge of emergency response facilities | An acceptable question could not be developed for the combination of randomly sampled evolution and generic K/A.<br>Randomly reselected K/A 295007 HIGH REACTOR PRESSURE 2.4.6 - Knowledge of emergency and abnormal operating procedures major action categories. |

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|---|---|---|
| 1 / 2 | Question #84<br><br>295022 LOSS OF CONTROL ROD DRIVE PUMPS<br><br>2.2.39 - Knowledge of less than or equal to 1 hour technical specification action statements (This K/A does not include action statements of 1 hour or less that follow the expiration of a completion time for a technical specification condition for which an action statement has already been entered.) | An acceptable question could not be developed for the randomly sampled generic K/A at the SRO level.<br><br>Randomly reselected K/A 295022 LOSS OF CONTROL ROD DRIVE PUMPS 2.2.22 - Knowledge of limiting conditions for operation and safety limits. |
| 2 / 1 | Question #90<br><br>262002 UNINTERRUPTABLE POWER SUPPLY (AC/DC)<br><br>2.4.25 - Knowledge of fire protection procedures | An acceptable question could not be developed for the combination of randomly sampled generic K/A due to lack of relevant connection with the given system.<br><br>Randomly reselected K/A 262002 UNINTERRUPTABLE POWER SUPPLY (AC/DC) 2.2.36 - Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operation. |
| 2 / 2 | Question #91<br><br>233000 FUEL POOL COOLING/CLEANUP<br><br>2.1.30 - Ability to locate and operate components, including local controls | System 233000 was inadvertently oversampled and an acceptable question could not be developed for the randomly sampled generic K/A at the SRO level.<br><br>Randomly reselected K/A 272000 Radiation Monitoring System 2.3.12 - Knowledge of radiological safety principles and procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, or alignment of filters. |

| | | |
|---|---|---|
| 3 | Question #94<br>2.1.15 - Knowledge of administrative requirements for temporary management direction, such as standing orders, night orders, or operations memoranda | An acceptable question could not be developed for the randomly sampled generic K/A at the SRO level.<br>Randomly reselected K/A 2.1.23 - Ability to perform general or normal operating procedures during any plant condition. |
| 3 | Question #99<br>2.4.12 - Knowledge of operating crew responsibilities during emergency and abnormal operations | An acceptable question could not be developed for the randomly sampled generic K/A at the SRO level.<br>Randomly reselected K/A 2.4.29 - Knowledge of the emergency plan implementing procedures. |
| 3 | Question #100<br>2.4.17 - Knowledge of emergency and abnormal operating procedures terms and definitions | An acceptable question could not be developed for the randomly sampled generic K/A at the SRO level.<br>Randomly reselected K/A 2.4.51 - Knowledge of emergency operating procedure exit conditions (e.g., emergency condition no longer exists or severe accident guideline entry is required). |
| | | |
| | | |

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | | |
|---|---|---|
| | Level | RO |
| | Tier # | 1 |
| | Group # | 1 |
| | K/A # | 295001 AK1.01 |
| | Importance Rating | 3.7 |

**PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION**

**Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to the PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Natural circulation**

Proposed Question: #1

A plant startup is in progress with the following:

- The Reactor Mode Switch is in RUN.
- Reactor power is 10%.

Then, all Reactor Recirculation pumps trip.

Which one of the following describes the required control of the Reactor, in accordance with the Special Operating Procedures?

A. A manual Reactor scram is required.

B. A manual Reactor scram is NOT required, but a plant shutdown is required.

C. Reactor operation can continue as long as NO indications of thermal-hydraulic instability occur.

D. Reactor operation can continue as long as core conditions satisfy the natural circulation line of the Power/Flow map.

Proposed Answer: A

Explanation: With the plant operating at power and a trip of one or more Reactor Recirculation pumps, N1-SOP-1.3 entry is required. With less than 3 operating Recirc loops, the procedure requires manually scramming the Reactor.

B. Incorrect – A manual Reactor scram is required. Plausible because Reactor power is low to begin with and there are no indications of THI given. Also plausible because there are conditions in N1-SOP-1.5 that required a normal shutdown instead of a Reactor scram.
C. Incorrect – A manual Reactor scram is required. Plausible because one of the related conditions that requires a Reactor scram in N1-SOP-1.5 is presence of THI.
D. Incorrect – A manual Reactor scram is required. Plausible because position on the Power/Flow map is addressed in both N1-SOP-1.3 and N1-SOP-1.5 and there is a natural circulation line.

Technical Reference(s): N1-SOP-1.3, N1-SOP-1.5

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-SOP1.3C01 EO-2

Question Source: Modified Bank - JAF 14-2 NRC #39

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

| Examination Outline Cross-Reference: | Level | RO |
|---|---|---|
| | Tier # | 1 |
| | Group # | 1 |
| | K/A # | 295001 AK1.01 |
| | Importance Rating | 3.5 |

**Partial or Complete Loss of Forced Core Flow Circulation**

**Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Natural circulation**

Proposed Question: #39

A plant startup is in progress with the following:

- The Reactor Mode Switch is in STARTUP/HOT STANDBY.
- Reactor pressure is 50 psig and slowly rising.

Then, both Recirculation pumps trip.

Which one of the following describes the required control of the Reactor in accordance with AOP-8, Unexpected Change in Core Flow?

A. A manual Reactor scram is required.

B. A manual Reactor scram is NOT required, but a plant shutdown is required.

C. Reactor operation can continue as long as NO indications of thermal-hydraulic instability occur.

D. Reactor operation can continue as long as core conditions satisfy the natural circulation line of the Power-Flow map.

Proposed Answer: B

Explanation: If both Recirculation pumps trip and the Reactor Mode Switch is in STARTUP/HOT STANDBY, AOP-8 requires commencing a normal plant shutdown per OP-65 and achieving Mode 3 within 12 hours. The trip of both Recirculation pumps places the plant in natural circulation. AOP-8 does not allowed continued operation in natural circulation.

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | | |
|---|---|---|
| | Level | RO |
| | Tier # | 1 |
| | Group # | 1 |
| | K/A # | 295003 2.4.16 |
| | Importance Rating | 3.5 |

**PARTIAL OR COMPLETE LOSS OF AC POWER**

**Knowledge of emergency and abnormal operating procedures implementation hierarchy and coordination with other support procedures or guidelines such as, operating procedures, abnormal operating procedures, or severe accident management guidelines**

Proposed Question: #2

The plant is operating at 40% power with the following:

- A loss of all 115 KV power occurs.
- N1-SOP-33A.1, Loss of 115 KV, is being performed.
- A spurious Reactor scram occurs.
- Following the scram, a pressure control malfunction results in Reactor pressure rising to 1100 psig.
- The Unit Supervisor enters N1-EOP-RPV, RPV Control, due to high Reactor pressure.

Which one of the following describes the correct procedure implementation for N1-SOP-33A.1?

A. Continue performing N1-SOP-33A.1. In the event of a conflict between N1-SOP-33A.1 and N1-EOP-RPV, N1-EOP-RPV is the overriding procedure.

B. Continue performing N1-SOP-33A.1. In the event of a conflict between N1-SOP-33A.1 and N1-EOP-RPV, N1-SOP-33A.1 is the overriding procedure.

C. Exit N1-SOP-33A.1. N1-SOP-33A.1 is re-entered at the step in progress after exiting N1-EOP-RPV.

D. Exit N1-SOP-33A.1. N1-SOP-33A.1 entry conditions are re-evaluated after exiting N1-EOP-RPV.

Proposed Answer: A

Explanation: There is no requirement to exit SOPs when an EOP is entered. Both procedures are executed concurrently. The EOPs are higher-tiered procedures than SOPs, therefore in the event of a conflict, the EOP must be followed.

B. Incorrect – The EOPs are higher-tiered procedures than SOPs, therefore in the event of a conflict, the EOP must be followed. Plausible because the SOP is event-specific while the EOP is not.
C. Incorrect – There is no requirement to exit SOPs when EOPs are entered. Plausible because the EOP is the higher-tiered document. Also plausible because N1-SOP-33A.1 does have an override step that requires moving back to the beginning of the procedure due to the Main Generator trip.
D. Incorrect – There is no requirement to exit SOPs when EOPs are entered. Plausible because the EOP is the higher-tiered document. Also plausible because N1-SOP-33A.1 does have an override step that requires moving back to the beginning of the procedure due to the Main Generator trip.

Technical Reference(s): N1-EOP-B-HOT, N1-SOP-33A.1, N1-EOP-RPV

Proposed references to be provided to applicants during examination: None

Learning Objective: S101-EOP00C01 TO #1

Question Source: Bank – 2018 NRC #39

Question History: 2018 NRC #39

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | | |
|---|---|---|
| | Level | RO |
| | Tier # | 1 |
| | Group # | 1 |
| | K/A # | 295004 AK3.01 |
| | Importance Rating | 3.9 |

**PARTIAL OR COMPLETE LOSS OF DC POWER**

**Knowledge of the reasons for the following responses or actions as they apply to PARTIAL OR COMPLETE LOSS OF DC POWER: Load shedding**

Proposed Question: #3

Which one of the following describes the reason for performing Battery load reductions during a Station Blackout?

A. Support a manual dead bus transfer to MG Set 167.

B. Maintain power to Reactor instrumentation, EC controls, and to start an EDG.

C. Avoid a loss of critical Battery Board loads due to breaker trips on over-current.

D. Maintain power to the Process Computer and Annunciators for the entire coping period.

Proposed Answer: B

Explanation: During a Station Blackout, Static Battery Chargers are lost and Batteries begin to discharge. Per N1-OP-47A, B.1, load reductions are performed to preserve Battery capacity, specifically for Reactor instrumentation, Emergency Condenser control, and EDG starting.

A. Incorrect – N1-SOP-33A.2 requires load shedding MG Set 167 within 2 hours of event start. Plausible because MG Set 167 does operate from DC power during a Station Blackout.
C. Incorrect – Breakers on the SR Battery Boards have been bypassed. Fuses have been installed in their place in the back of the Battery Boards. Both the Negative and Positive legs to each load are fused. Plausible if believe lowering voltage will draw more current and cause trips.
D. Incorrect – PPC and annunciators are de-energized when MG 167 is tripped 2 hours into the 4 hour coping period. Plausible because these loads are maintained for a longer period of time than most other loads.

Technical Reference(s): N1-OP-47A, N1-SOP-33A.2

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-263000-RBO-10

Question Source: Bank - 2020 NRC #41

Question History: 2020 NRC #41

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | Level | RO |
|---|---|---|
| | Tier # | 1 |
| | Group # | 1 |
| | K/A # | 295005 AK1.03 |
| | Importance Rating | 3.8 |

**MAIN TURBINE GENERATOR TRIP**

**Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to the MAIN TURBINE GENERATOR TRIP: Reactor level control**

Proposed Question: #4

A plant startup is in progress with the following:

- Reactor power is 17%.
- The Main Generator has just been synchronized to the grid.
- Reactor water level is 72" and stable.
- Feedwater pump 12 is operating.
- Feedwater pumps 11 and 13 are secured.

Then, a Main Generator lockout results in the following:

- The Turbine trips.
- Reactor water level reaches a low of 68".
- Reactor pressure is 918 psig and stable on Turbine Bypass Valves.

Which one of the following describes the resulting status of Feedwater pump 11 and Feedwater flow control valve 12?

| | Feedwater Pump 11 | Feedwater Flow Control Valve 12 |
|---|---|---|
| A. | Remains secured | Does NOT go into HPCI mode |
| B. | Remains secured | Goes into HPCI mode |
| C. | Automatically starts | Does NOT go into HPCI mode |
| D. | Automatically starts | Goes into HPCI mode |

Proposed Answer: D

Explanation: The Turbine trip signal causes Feedwater pump 11 to automatically start. The Turbine trip signal also causes Feedwater flow control valve 12 to go into HPCI mode.

A. Incorrect – Feedwater pump 11 automatically starts. Feedwater flow control valve 12 goes into HPCI mode. Plausible because Feedwater flow control valve 12 was already in service and Reactor water level remains above 53".
B. Incorrect – Feedwater pump 11 automatically starts. Plausible because this would be correct if Feedwater pump 12 flow rose to 1.9 Mlbm/hr without a Turbine trip signal.
C. Incorrect – Feedwater flow control valve 12 goes into HPCI mode. Plausible because Feedwater flow control valve 12 was already in service and Reactor water level remains above 53".

Technical Reference(s): N1-OP-16, ARP F4-4-1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-259001-RBO-05

Question Source: Modified Bank – 2013 NRC #30

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

**HPCI**

**Ability to verify that the alarms are consistent with the plant conditions**

Proposed Question: #30

A plant startup is in progress with the following:

- Reactor power is 18%.
- Turbine startup is in progress.

Then, a Turbine overspeed signal results in the following:

- Annunciator F1-2-5, RPS CH 11 TURBINE TRIP, alarms.
- Annunciator F4-2-4, RPS CH 12 TURBINE TRIP, alarms.
- Reactor water level is 72" and stable.
- Reactor pressure is 918 psig and stable.

Which of the following annunciator(s) would also be in alarm?

(1) F1-2-1, RPS CH 11 AUTO REACTOR TRIP
(2) F4-2-8, RPS CH 12 AUTO REACTOR TRIP
(3) F4-4-1, HPCI MODE AUTO INITIATE
(4) A1-4-5, TURBINE STOP VALVES CLOSED

A. (4) only

B. (1) and (2) only

C. (3) and (4) only

D. (1), (2), (3), and (4)

Proposed Answer: C

Explanation: Turbine overspeed is a mechanical automatic Turbine trip signal, as evidenced by annunciators F1-2-5 and F4-2-4 being in alarm. Below approximately 40-45% power, a Turbine trip does not cause an automatic Reactor scram. Therefore, annunciators F1-2-1 and F4-2-8 would not be in alarm. The Turbine trip signal does cause Turbine stop valves to close (Annunciator A1-4-5). The subsequent generator trip will cause an electrical turbine trip signal and cause HPCI to auto-initiate (Annunciator F4-4-1) (3T-1, 3T-2). These functions are unaffected by the low power level.

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | | |
|---|---|---|
| | Level | RO |
| | Tier # | 1 |
| | Group # | 1 |
| | K/A # | 295006 AK2.03 |
| | Importance Rating | 3.9 |

**SCRAM**

**Knowledge of the relationship between the SCRAM and the following systems or components: CRD hydraulic system**

Proposed Question: #5

The plant is operating at 100% power with the in-service CRD flow control valve in automatic when a Reactor scram occurs.

Which one of the following describes the response of CRD total system flow and drive water pressure?

| | CRD Total System Flow | CRD Drive Water Pressure |
|---|---|---|
| A. | Lowers | Lowers |
| B. | Lowers | Remains the same |
| C. | Rises | Lowers |
| D. | Rises | Remains the same |

Proposed Answer: C

Explanation: When the scram occurs, CRD total system flow rises due to the additional flow paths from the charging water header to each of the HCUs. The flow control valve senses this higher flow and closes in an attempt to lower flow. Due to the arrangement of the system, this causes drive water pressure to lower.

A. Incorrect – Total system flow rises. Plausible because the flow control valve closes in an attempt to lower flow.
B. Incorrect – Total system flow rises. Plausible because the flow control valve closes in an attempt to lower flow. Drive water pressure lowers. Plausible because the drive water pressure control valve does not change position.
D. Incorrect – Drive water pressure lowers. Plausible because the drive water pressure control valve does not change position.

Technical Reference(s): C-18016-C Sheet 1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-201001-RBO-03

Question Source: Bank – 2008 NRC #36

Question History: 2008 NRC #36

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(6)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | | |
|---|---|---|
| | Level | RO |
| | Tier # | 1 |
| | Group # | 1 |
| | K/A # | 295016 2.4.3 |
| | Importance Rating | 3.7 |

**CONTROL ROOM ABANDONMENT**

**Ability to identify post-accident instrumentation**

Proposed Question: #6

The Control Room has been evacuated and control has been established at Remote Shutdown Panel 11.

Given the following Primary Containment parameters:

(1) Drywell pressure
(2) Torus water level
(3) Containment high range radiation

Which of these parameters can be determined using installed indicators at Remote Shutdown Panel 11?

A. (1) only

B. (2) only

C. (3) only

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

Proposed Answer: A

Explanation: Remote Shutdown Panel 11 has an installed indicator for Drywell pressure, but not for Torus water level or Containment high range radiation.

**Note:** The question meets the K/A because it requires the candidate to identify post-accident instrumentation that would be used during a Control Room evacuation. The RO job function related to identifying post-accident instrumentation includes knowledge of where alternate post-accident instrumentation can be located. All three of the given Primary Containment parameters are post-accident instruments required by Technical Specifications.

B. Incorrect – Remote Shutdown Panel 11 does not have an installed indicator for Torus water level. Plausible because it does have an installed indicator for Torus water temperature.
C. Incorrect – Remote Shutdown Panel 11 does not have an installed indicator for Containment high range radiation. Plausible because this is a required instrument in Technical Specification 3.6.11 and is a Reg Guide 1.97 instrument.
D. Incorrect – Remote Shutdown Panel 11 does not have an installed indicator for Torus water level. Plausible because it does have an installed indicator for Torus water temperature. Remote Shutdown Panel 11 does not have an installed indicator for Containment high range radiation. Plausible because this is a required instrument in Technical Specification 3.6.11 and is a Reg Guide 1.97 instrument.

Technical Reference(s): Technical Specification Table 3.6.11-1, Technical Specification Table 3.6.13-1, 1101-296000C01

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-296000-RBO-5

Question Source: Modified Bank - 2018 NRC #57

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

**Primary Containment and Auxiliaries**

**Ability to identify post-accident instrumentation.**

Proposed Question: #57

The plant is shutdown with the following:

- An accident has occurred.
- Multiple Primary Containment instruments have been damaged in the Control Room.

Given the following Primary Containment parameters:

(1) Drywell pressure
(2) Torus water temperature
(3) Containment high range radiation

Which one of the following identifies which of these parameters can be determined using installed indicators at Remote Shutdown Panel 11?

A. (1) and (2) only

B. (1) and (3) only

C. (2) and (3) only

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

Proposed Answer: A

Explanation: Remote Shutdown Panel 11 has an installed indicator for Drywell pressure and Torus water temperature, but not Containment high range radiation.

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | | |
|---|---|---|
| | Level | RO |
| | Tier # | 1 |
| | Group # | 1 |
| | K/A # | 295018 AA1.04 |
| | Importance Rating | 3.4 |

**PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER (CCW)**

**Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER (CCW): Reactor water cleanup system**

Proposed Question: #7

The plant is operating at 100% power when the following occur:

- RBCLC pumps 11 and 12 trip.
- RBCLC pump 13 is started.
- RBCLC supply temperature is 95ºF and slowly rising.

Which one of the following actions is to be performed ***first*** based on the challenge to RBCLC cooling capability, in accordance with N1-SOP-11.1, RBCLC Failure?

A. Trip the RWCU pumps

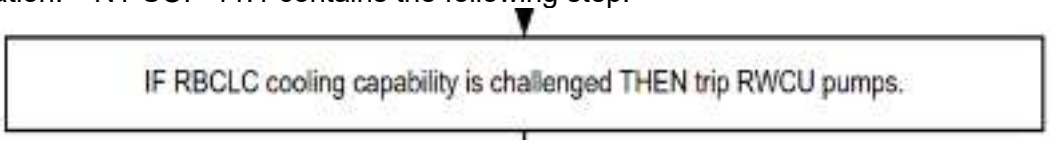
B. Secure Fuel Pool Cooling

C. Isolate RBCLC to the Drywell

D. Shutdown two Recirculation pumps

Proposed Answer: A

Explanation: N1-SOP-11.1 contains the following step:




Since the only available RBCLC pump is running and temperature is rising, cooling capability is being challenged and RWCU pumps must be tripped.

B. Incorrect – RWCU pumps must be tripped first. Plausible because N1-SOP-11.1 directs monitoring SFPC and also directs securing affected equipment if high temperatures occur.
C. Incorrect – RWCU pumps must be tripped first. Plausible because N1-SOP-11.1 directs monitoring Drywell cooling and also directs securing affected equipment if high temperatures occur.
D. Incorrect – RWCU pumps must be tripped first. Plausible because N1-SOP-11.1 directs monitoring Recirculation system temperatures and also directs securing affected equipment if high temperatures occur. Also plausible because plant operation can continue with two Recirculation pumps secured.

Technical Reference(s): N1-SOP-11.1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-208000-RBO-10

Question Source: Bank – 2018 NRC #34

Question History: 2018 NRC #34

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(4)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| | | |
|---|---|---|
| Examination Outline Cross-Reference: | Level | RO |
| | Tier # | 1 |
| | Group # | 1 |
| | K/A # | 295019 AA2.01 |
| | Importance Rating | 4.1 |

**PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR**

**Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: Instrument air pressure**

Proposed Question: #8

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

- Instrument Air Compressor (IAC) 13 is running.
- IAC 11 is red-flagged with Loading Selector Switches in Position 1.
- IAC 12 is green-flagged with Loading Selector Switches in Position 2.

Then, an Instrument Air (IA) line break results in the following:

| Time (mm:ss) | Condition(s) |
|---|---|
| 00:10 | IAC 13 loads |
| 00:20 | IAC 11 loads |
| 00:30 | IAC 12 starts and loads |
| 00:45 | Annunciator L1-4-7, INST AIR BACK – UP VALVE OPEN, alarms |
| 00:57 | Breathing Air Blocking Valve 114-02 closes<br>Automatic Dryer Bypass valves 94-164 and 94-206 open |

IA pressure has been continuously lowering throughout this transient.

Which one of the following describes Instrument Air pressure at time 00:38?

At time 00:38, Instrument Air pressure is between…

A. 98 and 101 psig

B. 93 and 98 psig

C. 90 and 93 psig

D. 85 and 90 psig

Proposed Answer: C

Explanation: IAC 12 is green-flagged as the standby compressor, therefore it starts and loads at an IA pressure of 93 psig, then attempts to control pressure 96-102 psig. Annunciator L1-4-7, INST AIR BACK – UP VALVE OPEN, alarms at 90 psig. Therefore, at time 00:38, Instrument Air pressure is between 90 and 93 psig.

A. Incorrect – At time 00:38, Instrument Air pressure is between 90 and 93 psig. Plausible because this would be correct for time 00:15.
B. Incorrect – At time 00:38, Instrument Air pressure is between 90 and 93 psig. Plausible because this would be correct for time 00:25.
D. Incorrect – At time 00:38, Instrument Air pressure is between 90 and 93 psig. Plausible because this would be correct for time 00:50.

Technical Reference(s): N1-OP-20, ARP L1-4-7

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-278001-RBO-05

Question Source: Modified Bank - 2013 NRC #8

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments:

Proposed Question: #8

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

- Instrument Air Compressor (IAC) 13 is running.
- IAC 11 is red-flagged with Loading Selector Switches in Position 1.
- IAC 12 is green-flagged with Loading Selector Switches in Position 2.

Then, an Instrument Air (IA) line break results in the following sequence of events:

| Time (mm:ss) | Event |
|---|---|
| 00:10 | Instrument Air Compressor (IAC) 13 loads. |
| 00:20 | IAC 11 loads. |
| 00:30 | IAC 12 starts and loads. |
| 00:45 | Annunciator L1-4-7, INST AIR BACK – UP VALVE OPEN, alarms. |
| 00:57 | Breathing Air Blocking Valve 114-02 closes; Automatic Dryer Bypass valves 94-164 and 94-206 open. |

IA pressure has still been continuously lowering throughout this transient.

Which one of the following describes Instrument Air pressure at time 00:25?

At time 00:25, Instrument Air pressure is between…

A. 98 and 101 psig.

B. 93 and 98 psig.

C. 90 and 93 psig.

D. 85 and 90 psig.



Proposed Answer: B

Explanation: IAC 11 is red-flagged as the backup compressor with Load Selector Switches in Position 1, therefore it loads on an IA pressure of 98#. IAC 12 is green-flagged as the standby compressor, therefore it starts and loads at an IA pressure of 93#, then attempts to control pressure 96#-102#.

**Form 4.2-1 Written Examination Question Worksheet**

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

**LOSS OF SHUTDOWN COOLING**

**Ability to determine and/or interpret the following as they apply to LOSS OF SHUTDOWN COOLING: RHR/shutdown cooling system flow**

Proposed Question: #9

The plant is shutdown with the following:

- Shutdown Cooling (SDC) loops 11 and 12 have been placed in service.
- NO related valves have been gagged.

Then, a complete loss of Instrument Air occurs.

Which one of the following describes the effect, if any, on SDC flow through the core and RBCLC flow through the SDC heat exchangers?

| | SDC Flow Through Core | RBCLC Flow Through SDC HX |
|---|---|---|
| A. | Remains the same | Remains the same |
| B. | Remains the same | Lowers |
| C. | Lowers | Remains the same |
| D. | Lowers | Lowers |

Proposed Answer: D

Explanation: The instrument air system provides motive force to operate the electropneumatic flow control valves and the minimum flow control valves. It provides motive force for RBCLC inlet to SDC. On a loss of air pressure, the flow control valves close and the minimum flow control valves open. Also, the RBCLC blocking valve fails closed. Therefore, both flows will drop to zero.

A. Incorrect – SDC flow through the core lowers. Plausible because the SDC isolation and suction valves are all motor-operated, such that they remain open on loss of Instrument Air. Also plausible because some pneumatically controlled valves fail open or as-is on loss of pressure to maintain system operation.
B. Incorrect – SDC flow through the core lowers. Plausible because the SDC isolation and suction valves are all motor-operated, such that they remain open on loss of Instrument Air. RBCLC flow through the SDC heat exchangers lowers. Plausible because the valves used to set RBCLC flow through the SDC heat exchangers are manually operated. Also plausible because some pneumatically controlled valves fail open or as-is on loss of pressure to maintain system operation.
C. Incorrect – RBCLC flow through the SDC heat exchangers lowers. Plausible because the valves used to set RBCLC flow through the SDC heat exchangers are manually operated. Also plausible because some pneumatically controlled valves fail open or as-is on loss of pressure to maintain system operation.

Technical Reference(s): C-18018-C

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-205000-RBO-8

Question Source: Bank – 2009 Cert #26

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(3)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

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

**REFUELING ACCIDENTS**

**Knowledge of the reasons for the following responses or actions as they apply to REFUELING ACCIDENTS: Ventilation isolation**

Proposed Question: #10

A refueling outage is in progress with the following:

- An irradiated fuel bundle has been dropped in the Spent Fuel Pool.
- Reactor Building ventilation has automatically isolated due to a signal from Fuel Pool/Refuel Bridge High Range ARM (RN016C-10).

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

A. Minimize air currents above the Spent Fuel Pool

B. Avoid damage to Reactor Building ventilation fans

C. Control fission product release to the environment

D. Prevent personnel exposure in lower Reactor Building elevations

Proposed Answer: C

Explanation: Reactor Building ventilation isolates to control the release of fission products to the environment by ensuring they are routed to RBEVS.

A. Incorrect – The reason is to control fission product release to the environment. Plausible because RBEVS does have a lower flow rate than RBVS and higher air flow rate above the SFP could result in more spread of contamination.
B. Incorrect – The reason is to control fission product release to the environment. Plausible because the isolation does stop flow of contaminated air through the Reactor Building ventilation fans, avoiding any possible damage.
D. Incorrect – The reason is to control fission product release to the environment. Plausible that continued operation of Reactor Building ventilation could spread contamination throughout the building, which would be undesirable.

Technical Reference(s): Technical Specification 3.4.2 Objective and bases

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-288001-RBO-01

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(9)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | | |
|---|---|---|
| | Level | RO |
| | Tier # | 1 |
| | Group # | 1 |
| | K/A # | 295024 EA1.12 |
| | Importance Rating | 4.2 |

**HIGH DRYWELL PRESSURE**

**Ability to operate and/or monitor the following as they apply to HIGH DRYWELL PRESSURE: Suppression pool spray (Mark I, II)**

Proposed Question: #11

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

- Reactor water level is -25" and lowering slowly.
- Reactor pressure is 400 psig and lowering.
- All four Containment Spray pumps are spraying the Containment.
- Drywell pressure is 3.2 psig and lowering slowly.

Which one of the following describes the required control of Containment Spray, in accordance with the Emergency Operating Procedures?

Secure Containment Spray…

A. now. The pump control switches are required to be left green-flagged.

B. now. The pump control switches are NOT required to be left green-flagged.

C. when Drywell pressure reaches 0 psig. When secured, the pump control switches are required to be left green-flagged.

D. when Drywell pressure reaches 0 psig. When secured, the pump control switches are NOT required to be left green-flagged.

Proposed Answer: B

Explanation: Drywell pressure of 3.5 psig is the threshold for securing Containment Sprays in N1-EOP-PC. Since Drywell pressure is below this value, Containment Sprays are required to be secured now. This may be accomplished by placing the pump control switches in PULL-TO-LOCK, specifically to prevent automatic Containment Spray operation. Containment Spray initiation is controlled manually while in N1-EOP-PC.

A. Incorrect – Pump control switches may be placed in PULL-TO-LOCK. Plausible because this does defeat auto-start capability and Drywell pressure is currently low enough that green-flagging the switches would stop the pumps.
C. Incorrect – Pump control switches may be placed in PULL-TO-LOCK. Plausible because this does defeat auto-start capability and Drywell pressure is currently low enough that green-flagging the switches would stop the pumps. Containment Spray is required to be secured now. Plausible because 0 psig is the threshold used in the SAPs.
D. Incorrect – Containment Spray is required to be secured now. Plausible because 0 psig is the threshold used in the SAPs.

Technical Reference(s): N1-EOP-PC, N1-EOP-1 Attachment 17

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOPPCC01 EO-2

Question Source: Bank - 2017 Cert #58

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | Level | RO |
|---|---|---|
| | Tier # | 1 |
| | Group # | 1 |
| | K/A # | 295025 EK2.01 |
| | Importance Rating | 4.2 |

**HIGH REACTOR PRESSURE**

**Knowledge of the relationship between the HIGH REACTOR PRESSURE and the following systems or components: RPS**

Proposed Question: #12

A plant startup is in progress with the following:

- IRMs are on range 9.
- An open Turbine Bypass Valve rapidly fails closed.
- Reactor pressure rises rapidly to a peak of 1085 psig.
- Other Turbine Bypass Valves throttle open.
- Reactor pressure is restored to a normal value within 7 seconds of the start of the transient.

Which one of the following describes the resulting status of RPS, ERVs, and Emergency Condensers?

A. NO Reactor scram, ERV actuation, or Emergency Condenser initiation occurs.

B. The Reactor scrams. NO ERV actuation or Emergency Condenser initiation occurs.

C. The Reactor scrams. Two ERVs open. NO Emergency Condenser initiation occurs.

D. The Reactor scrams. Emergency Condensers initiate. NO ERV actuation occurs.

Proposed Answer: B

Explanation: The Reactor scrams based on pressure exceeding 1080 psig. There is no time delay on this signal. The lowest ERV actuation setpoint is 1090 psig, which is not reached. The Emergency Condensers initiate only if pressure is above 1080 psig for more than 12 seconds. In this question, pressure is only above normal for 7 seconds, so no Emergency condenser initiation occurs.

A. Incorrect – The Reactor scrams. Plausible because pressure goes only slightly above the scram setpoint for a short amount of time, and there are time delays on some related signals.
C. Incorrect – No ERVs open. Plausible because pressure is close to the lowest ERV initiation setpoint of 1090 psig.
D. Incorrect – ECs to not initiate. Plausible because pressure went above the setpoint of 1080 psig, but did not meet the 12 second time delay requirement

Technical Reference(s): N1-OP-40, N1-OP-13, N1-OP-1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-212000-RBO-5

Question Source: Bank - 2010 NRC #44

Question History: 2010 NRC #44

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | | |
|---|---|---|
| | Level | RO |
| | Tier # | 1 |
| | Group # | 1 |
| | K/A # | 295026 EA1.09 |
| | Importance Rating | 4.1 |

**SUPPRESSION POOL HIGH WATER TEMPERATURE**

**Ability to operate and/or monitor the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: Safety/relief valves**

Proposed Question: #13

A steam leak in the Main Steam Tunnel has resulted in the following:

- The Reactor has scrammed.
- MSIVs are closed.
- Reactor water level is 100" and stable.
- Reactor pressure is 975 psig and rising slowly.
- Torus water temperature is 115°F and rising slowly.
- All ERVs are currently closed.
- The Unit Supervisor has directed Reactor pressure controlled 800-1000 psig.

Which one of the following describes the allowed use of ERVs for pressure control under these conditions, in accordance with plant procedures?

Using ERVs for pressure control is currently…

A. NOT allowed due to Reactor water level.

B. NOT allowed due to Torus water temperature.

C. allowed. Cycle the same ERV repeatedly, as needed.

D. allowed. Cycle through all ERVs sequentially, as needed.

Proposed Answer: D

Explanation: ERVs are currently allowed to be used for pressure control despite high Reactor water level and high Torus water temperature. ERVs discharges are designed to be equally spaced at points around the Torus. N1-EOP-RPV contains direction on the use of ERVs for Reactor pressure control based on this design to ensure more equal heat load distribution around the Torus by opening ERVs in a preferred sequence, rather than just opening the same ERV repeatedly.

A. Incorrect – ERVs are currently allowed to be used for pressure control despite high Reactor water level. Plausible because this high Reactor water level results in restrictions on use of ECs and could result in carryover of moisture in the ERV discharge lines.
B. Incorrect – ERVs are currently allowed to be used for pressure control despite high Torus water temperature. Plausible because Torus water temperature is above the 110°F limit used in EOPs and use of ERVs will further raise Torus water temperature.
C. Incorrect – N1-EOP-RPV contains a note that directs opening ERVs per a pre-defined sequence, vice opening the same ERV multiple times. Plausible because Torus Cooling does discharge to one point in the Torus through the test return line, so this area will be cooled more directly.

Technical Reference(s): N1-EOP-RPV, N1-EOP-B-HOT

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-239002-RBO-10

Question Source: Modified Bank – 2023 NRC #23

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(3)

Comments:

| Form 4.2-1 | Written Examination Question Worksheet |
|---|---|

Examination Outline Cross-Reference: Level RO

| | |
|---|---|
| Tier # | 1 |
| Group # | 2 |
| K/A # | 295013 AA2.02 |
| Importance Rating | 3.5 |

*HIGH SUPPRESSION POOL TEMPERATURE*

**Ability to determine or interpret the following as they apply to HIGH SUPPRESSION POOL TEMPERATURE: Localized heating/stratification**

Proposed Question: #23

A steam leak in the Main Steam Tunnel has resulted in the following:

- The Reactor has scrammed.
- The Unit Supervisor has directed Reactor pressure controlled 800-1000 psig using ERVs.

Which one of the following describes how ERVs should be operated, in accordance with the Emergency Operating Procedures?

A. Cycle through all ERVs sequentially, as needed, to verify they are all seated properly.

B. Cycle the same ERV repeatedly, as needed, to prevent thermal cycling of multiple ERV tailpipes.

C. Cycle through all ERVs sequentially, as needed, to distribute the heat load uniformly around the Torus.

D. Cycle the same ERV repeatedly, as needed, to maintain Torus heating as close as possible to the Torus Cooling injection point.

Proposed Answer: C

Explanation: ERVs discharges are designed to be equally spaced at points around the Torus. N1-EOP-RPV contains direction on the use of ERVs for Reactor pressure control based on this design to ensure more equal heat load distribution around the Torus by opening ERVs in a preferred sequence, rather than just opening the same ERV repeatedly.

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | | |
|---|---|---|
| | Level | RO |
| | Tier # | 1 |
| | Group # | 1 |
| | K/A # | 295028 2.1.2 |
| | Importance Rating | 4.1 |

**HIGH DRYWELL TEMPERATURE**

**Knowledge of operator responsibilities during any mode of plant operation**

Proposed Question: #14

Which one of the following describes the requirement in N1-EOP-PC, Primary Containment Control, for performing an RPV Blowdown due to high Drywell temperature and the basis of this requirement?

N1-EOP-PC requires RPV Blowdown if Drywell temperature cannot be restored and maintained below the threshold of…

A. 300°F. This requirement is based on ADS qualification temperatures.

B. 300°F. This requirement is based on Recirc pump seal qualification temperatures.

C. 400°F. This requirement is based on ADS qualification temperatures.

D. 400°F. This requirement is based on Recirc pump seal qualification temperatures.

Proposed Answer: A

Explanation: N1-EOP-4 requires an RPV Blowdown if Drywell temperature cannot be restored and maintained below the threshold of 300°F. This temperature limit is based on the component qualification temperature for ERV solenoids (rounded down from 301°F).

B. Incorrect – The basis is ADS qualification temperature, not Recirc pump seal qualification temperatures. Plausible because Recirc pump seals are affected by high Drywell temperature and temperature damage to Recirc pump seals is a concern addressed in PRA.
C. Incorrect – The limit is 300°F, not 400°F. Plausible because 400°F is the upper limit on Drywell temperature indication and is part of the basis for multiple EOP limits (Detail A - Curve B and Table C, CSIL).
D. Incorrect – The limit is 300°F, not 400°F. Plausible because 400°F is the upper limit on Drywell temperature indication and is part of the basis for multiple EOP limits (Detail A - Curve B and Table C, CSIL). The basis is ADS qualification temperature, not Recirc pump seal qualification temperatures. Plausible because Recirc pump seals are affected by high Drywell temperature and temperature damage to Recirc pump seals is a concern addressed in PRA.

Technical Reference(s): N1-EOP-PC, N1-EOP-B-HOT

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOPPCC01 EO-2

Question Source: Bank - 2018 NRC #40

Question History: 2018 NRC #40

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| | | |
|---|---|---|
| Examination Outline Cross-Reference: | Level | RO |
| | Tier # | 1 |
| | Group # | 1 |
| | K/A # | 295030 EK3.05 |
| | Importance Rating | 3.6 |

**LOW SUPPRESSION POOL WATER LEVEL**

**Knowledge of the reasons for the following responses or actions as they apply to LOW SUPPRESSION POOL WATER LEVEL: Suppression pool makeup system(s) operation**

Proposed Question: #15

The plant is operating at 100% power when the following occurs:

- A leak develops in the Torus.
- Torus water level is 10.5' and lowering.
- N1-EOP-PC, Primary Containment Control, has been entered.

Which one of the following describes a reason for using N1-EOP-1 Attachment 18, Raw Water to Torus Makeup, instead of Attachment 6, Torus Make-Up from Keep Fill?

A. Higher purity water

B. Higher makeup flow rate

C. Does NOT require starting a pump

D. Does NOT require valve manipulations

Proposed Answer: B

Explanation: One reason for using N1-EOP-1 Attachment 18 instead of Attachment 6 is that it has a higher makeup flow rate.

A. Incorrect – N1-EOP-1 Attachment 18 uses lower purity water (water directly from Lake Ontario vs. water from the CSTs). Plausible because this would be a reason for using Attachment 6 instead of Attachment 18.
C. Incorrect – N1-EOP-1 Attachment 18 requires starting a Containment Spray Raw Water pump. Plausible because this would be a reason for using Attachment 6 instead of Attachment 18.
D. Incorrect – N1-EOP-1 Attachment 18 requires multiple valve manipulations. Plausible because Attachment 6 does require valve manipulations. Also plausible because one advantage of Attachment 18 is that it does not require field actions to support valve manipulations (Attachment 6 requires field action to supply power to two valves).

Technical Reference(s): N1-EOP-PC, N1-EOP-1 Attachments 6 and 18, N1-OP-14, N1-OP-2

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOPPCC01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

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

**REACTOR LOW WATER LEVEL**

**Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to the REACTOR LOW WATER LEVEL: Adequate core cooling**

Proposed Question: #16

A loss of coolant accident has resulted in the following:

- Reactor water level is -94" and lowering slowly.
- Four (4) ERVs are open.
- Reactor pressure is 100 psig and lowering.
- Condensate pump 12 is injecting to the Reactor.
- NO other injection sources are available.

Which one of the following describes the status of core cooling, in accordance with the Emergency Operating Procedures?

Adequate core cooling (ACC) is…

A. NOT currently being provided.

B. currently being provided. ACC will be first lost if Reactor water level reaches -109".

C. currently being provided. ACC will be first lost if Reactor water level reaches -126".

D. currently being provided. ACC will be first lost if Reactor water level reaches -156".

Proposed Answer: C

Explanation: The preferred method of ACC, Core Submergence, is not currently available because Reactor water level is below -84". However, ACC is still assured by Steam Cooling, which requires Reactor water level to be above -126". If Reactor water level reaches -126", this method of ACC will be lost. ACC by Spray Cooling is not available because no Core Spray pumps are available. Therefore, if Reactor water level reaches -109", ACC will no longer be assured.

A. Incorrect – Since Reactor water level is greater than -109", ACC is assured by Steam Cooling. Plausible because the preferred method of ACC, Core Submergence, is not currently available because Reactor water level is below -84".
B. Incorrect – ACC will first be lost if Reactor water level reaches -126". Plausible because -109" is the Reactor water level historically associated with Steam Cooling, however this level has recently been revised in the EOPs at NMP1.
D. Incorrect – ACC will first be lost if Reactor water level reaches -126". Plausible because -156" is the Reactor water level used in the N1-EOP-C1 Steam Cooling leg, which is not applicable since there is currently injection to the Reactor.

Technical Reference(s): N1-EOP-RPV, N1-EOP-B-HOT

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOPRPVC01 EO-2

Question Source: Bank – 2017 NRC #73

Question History: 2017 NRC #73

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| | | |
|---|---|---|
| Examination Outline Cross-Reference: | Level | RO |
| | Tier # | 1 |
| | Group # | 1 |
| | K/A # | 295037 EK2.14 |
| | Importance Rating | 4.0 |

**SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN**

**Knowledge of the relationship between the SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN and the following systems or components: RPIS (BWR 2, 3, 4, 5)**

Proposed Question: #17

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

- A Turbine trip causes a Reactor scram.
- All APRMs are unavailable.
- One control rod fails to insert and is at position 48.
- Two control rods partially insert and are at position 06.
- All other control rods fully insert.

Which one of the following describes the status of the Reactor shutdown, in accordance with the Emergency Operating Procedures?

The Maximum Subcritical Banked Withdrawal Position __(1)__ being exceeded.

The Reactor __(2)__ considered shutdown under all conditions.

| | (1) | (2) |
|---|---|---|
| A. | is | is |
| B. | is | is NOT |
| C. | is NOT | is |
| D. | is NOT | is NOT |

Proposed Answer: B

Explanation: The Maximum Subcritical Banked Withdrawal Position of 04 is being exceeded. With more than one rod withdrawn beyond the Maximum Subcritical Banked Withdrawal Position, the Reactor is NOT considered shutdown under all conditions.

A. Incorrect – The Reactor is NOT considered shutdown under all conditions. Plausible because one rod full out is allowed by Technical Specification shutdown margin calculations and the other two rods are only withdrawn one notch beyond the Maximum Subcritical Banked Withdrawal Position. Also plausible because this would be correct if only one rod was out.
C. Incorrect – The Maximum Subcritical Banked Withdrawal Position of 04 is being exceeded. The Reactor is NOT considered shutdown under all conditions. Plausible because one rod full out is allowed by Technical Specification shutdown margin calculations and the other two rods are only withdrawn one notch beyond the Maximum Subcritical Banked Withdrawal Position.
D. Incorrect – The Maximum Subcritical Banked Withdrawal Position of 04 is being exceeded. Plausible because one rod full out is allowed by Technical Specification shutdown margin calculations and the other two rods are only withdrawn one notch beyond the Maximum Subcritical Banked Withdrawal Position.

Technical Reference(s): N1-EOP-B-HOT

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOPRPVC01 EO-2

Question Source: Bank – NMP2 4/23 NRC #5

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | | |
|---|---|---|
| | Level | RO |
| | Tier # | 1 |
| | Group # | 1 |
| | K/A # | 295038 EA2.05 |
| | Importance Rating | 3.6 |

**HIGH OFFSITE RADIOACTIVITY RELEASE RATE**

**Ability to determine and/or interpret the following as they apply to HIGH OFFSITE RADIOACTIVITY RELEASE RATE: Emergency plan implementation**

Proposed Question: #18

An unisolable Emergency Condenser tube leak results in the following sequence of events:

| Time (hh:mm) | Event |
|---|---|
| 08:14 | The offsite release rate exceeds the Emergency Plan “Unusual Event” level. |
| 09:05 | The offsite release rate exceeds the Emergency Plan “Alert” level. |
| 10:10 | The offsite release rate exceeds the Emergency Plan “Site Area Emergency” level. |
| 11:23 | The offsite release rate exceeds the Emergency Plan “General Emergency” level. |

Which one of the following is the ***first*** time that entry into N1-EOP-RR, Radioactivity Release Control, is required?

A. 08:14

B. 09:05

C. 10:10

D. 11:23

Proposed Answer: B

Explanation: N1-EOP-RR entry is required when the offsite release rate is above the Emergency Plan “Alert” Level. The first time that this entry condition is met is 09:05.

A. Incorrect – N1-EOP-RR entry is required when the offsite release rate is above the Emergency Plan “Alert” Level. The first time that this entry condition is met is 09:05. Plausible that entry would be required as soon as Emergency Plan entry occurred.
C. Incorrect – N1-EOP-RR entry is required when the offsite release rate is above the Emergency Plan “Alert” Level. The first time that this entry condition is met is 09:05. Plausible that entry would first be required one level higher in the Emergency Plan.
D. Incorrect – N1-EOP-RR entry is required when the offsite release rate is above the Emergency Plan “Alert” Level. The first time that this entry condition is met is 09:05. Plausible because the General Emergency level is used in N1-EOP-RR.

Technical Reference(s): N1-EOP-RR

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOPRRC01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| | | |
|---|---|---|
| Examination Outline Cross-Reference: | Level | RO |
| | Tier # | 1 |
| | Group # | 1 |
| | K/A # | 600000 AA2.03 |
| | Importance Rating | 3.8 |

**PLANT FIRE ON SITE**

**Ability to determine and/or interpret the following as they apply to PLANT FIRE ON SITE: Fire alarm**

Proposed Question: #19

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

- Main Fire Panel Annunciator 2-1-1-3, TURB. BLDG 261 LOCAL PNL NO.3 FIRE, alarms.
- Fire Zone D-2263, TURB BLDG 289 RB CHARCOAL FILTERS, is in alarm.

Which one of the following describes the associated fire-fighting system and how it is initiated?

A. Manually initiated $CO_2$

B. Automatically initiated $CO_2$

C. Manually initiated water deluge

D. Automatically initiated water deluge

Proposed Answer: C

Explanation: This alarm is based on detectors in the RBEVS charcoal filters. These detectors initiate the given alarms, but do not automatically initiate fire suppression. The associated fire suppression system is a water deluge system that must be initiated manually by opening valves.

A. Incorrect – The associated fire suppression system is a water deluge system. Plausible because adding water to a charcoal filter is generally undesirable, such that use of $CO_2$ would be a likely alternative.
B. Incorrect – The associated fire suppression system is a water deluge system. Plausible because adding water to a charcoal filter is generally undesirable, such that use of $CO_2$ would be a likely alternative. The associated fire suppression system is manually initiated. Plausible that the same detectors that initiate the given alarms would also automatically initiate suppression, as in many other locations.
D. Incorrect – The associated fire suppression system is manually initiated. Plausible that the same detectors that initiate the given alarms would also automatically initiate suppression, as in many other locations.

Technical Reference(s): ARP 2-1-1-3, N1-OP-10

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-261000-RBO-8

Question Source: Bank – 2020 NRC #57

Question History: 2020 NRC #57

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(4)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| | | |
|---|---|---|
| Examination Outline Cross-Reference: | Level | RO |
| | Tier # | 1 |
| | Group # | 1 |
| | K/A # | 700000 AK3.01 |
| | Importance Rating | 3.8 |

**GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES**

**Knowledge of the reasons for the following responses or actions as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: Reactor and turbine trip criteria**

Proposed Question: #20

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

- A grid disturbance is in progress.
- N1-SOP-33B.1, Major 345 KV Grid Disturbances, is being executed.

Which one of the following 345 KV system conditions would be reason for an immediate Reactor scram?

A. Voltage of 340 KV

B. Voltage of 355 KV

C. Frequency of 59 Hz

D. Frequency of 62 Hz

Proposed Answer: D

Explanation: N1-SOP-33B.1 contains an override for scramming the Reactor and tripping the Main Turbine. One condition that requires a scram is 345 KV line frequency variation of greater than ±1.9 Hz from the nominal frequency of 60 Hz. Therefore, a frequency of 62 Hz would be reason for an immediate Reactor scram.

A. Incorrect – N1-SOP-33B.1 does not require a scram based on this voltage. Plausible because voltage is a major parameter monitored in the procedure. Also plausible because the related N1-SOP-33A.3 does have various steps that are required based on 115 KV voltage.
B. Incorrect – N1-SOP-33B.1 does not require a scram based on this voltage. Plausible because voltage is a major parameter monitored in the procedure. Also plausible because the related N1-SOP-33A.3 does have various steps that are required based on 115 KV voltage.
C. Incorrect – N1-SOP-33B.1 does not require an immediate scram based on this frequency. Plausible because this frequency would start a 90 minute clock for a Reactor scram.

Technical Reference(s): N1-SOP-33B.1

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-SOP33B1C01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | | |
|---|---|---|
| | Level | RO |
| | Tier # | 1 |
| | Group # | 2 |
| | K/A # | 295009 AK3.01 |
| | Importance Rating | 3.7 |

**LOW REACTOR WATER LEVEL**

**Knowledge of the reasons for the following responses or actions as they apply to LOW REACTOR WATER LEVEL: Recirculation runback**

Proposed Question: #21

Following a scram from 100% power, N1-SOP-1, Reactor Scram, requires lowering Recirculation flow.

Which one of the following describes the reason for this Recirculation flow reduction?

Avoid prolonged operation in the Power/Flow Map…

A. Restricted Zone. Operation in this region could cause unseating of fuel bundles.

B. Reactor Internals Protection Region. Operation in this region could cause unseating of fuel bundles.

C. Restricted Zone. Operation in this region could cause core shroud damage in the event of a Main Steam line rupture.

D. Reactor Internals Protection Region. Operation in this region could cause core shroud damage in the event of a Main Steam line rupture.

Proposed Answer: D

Explanation: When operating at 100% power, Recirculation flow is greater than 43 Mlbm/hr. 43 Mlbm/hr is the lower bound of the Reactor Internals Protection Region of the Power/Flow Map. When the scram occurs, Reactor power rapidly lowers, but Recirculation flow stays at the original value. This places operation in the Reactor Internals Protection Region. N1-SOP-1 directs lowering Recirculation flow to 25-43 Mlbm/hr to exit the Reactor Internals Protection Region. The concern with operation in the Reactor Internals Protection Region is the potential for damage to the core shroud in the event of a Main Steam line break.

A. Incorrect – Operation following a scram from 100% power is in the Reactor Internals Protection Region, not the Restricted Zone. Plausible because the Restricted Zone is another undesirable area of the Power/Flow Map. The concern is the potential for damage to the core shroud in the event of a Main Steam line break. Plausible because higher Recirculation Flow causes more upward flow through the core, which would provide an upward force on fuel bundles.
B. Incorrect – The concern is the potential for damage to the core shroud in the event of a Main Steam line break. Plausible because higher Recirculation Flow causes more upward flow through the core, which would provide an upward force on fuel bundles.
C. Incorrect – Operation following a scram from 100% power is in the Reactor Internals Protection Region, not the Restricted Zone. Plausible because the Restricted Zone is another undesirable area of the Power/Flow Map.

Technical Reference(s): N1-SOP-1, P/F Map, N1-OP-1

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-SOP1C01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(3)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | | |
|---|---|---|
| | Level | RO |
| | Tier # | 1 |
| | Group # | 2 |
| | K/A # | 295010 AK2.07 |
| | Importance Rating | 3.5 |

**HIGH DRYWELL PRESSURE**

**Knowledge of the relationship between the HIGH DRYWELL PRESSURE and the following systems or components: Component cooling water**

Proposed Question: #22

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

- Drywell cooling fan 11 trips.
- NO additional Drywell cooling fans are available to start.
- Drywell average temperature is 135°F and rising slowly.
- Drywell pressure is 1.2 psig and rising slowly.
- The US has directed additional Drywell cooling per N1-OP-8 Section H, Maintaining Drywell Average Temperature During Steady State Power Operations.

Which one of the following identifies the RBCLC heat exchanger to be verified in service and the reason for using this RBCLC heat exchanger for additional Drywell cooling, in accordance with N1-OP-8, Primary Containment Area Cooling System, and N1-OP-11, Reactor Building Closed Loop Cooling System?

| | RBCLC Heat Exchanger Used for Additional Drywell Cooling | Reason |
|---|---|---|
| A. | 11 | Variation in RBCLC heat exchanger design heat removal capacity |
| B. | 11 | Variation in piping configuration between RBCLC heat exchangers and Drywell cooling |
| C. | 12 | Variation in RBCLC heat exchanger design heat removal capacity |
| D. | 12 | Variation in piping configuration between RBCLC heat exchangers and Drywell cooling |

Proposed Answer: B

Explanation: For additional Drywell cooling, RBCLC heat exchanger 11 is the preferred heat exchanger. The reason for this is due to differences in piping configuration between the RBCLC heat exchangers and Drywell cooling.

A. Incorrect – The reason is due to differences in piping configuration between the RBCLC heat exchangers and Drywell cooling. Plausible because differences in heat exchanger design heat removal capacity could also cause differences in Drywell cooling (for example, if one heat exchanger had been replaced, modified, etc.).
C. Incorrect – RBCLC heat exchanger 11, not 12, is preferred for additional Drywell cooling. Plausible because there is no inherent design difference between these heat exchangers themselves. The reason is due to differences in piping configuration between the RBCLC heat exchangers and Drywell cooling. Plausible because differences in heat exchanger design heat removal capacity could also cause differences in Drywell cooling (for example, if one heat exchanger had been replaced, modified, etc.).
D. Incorrect – RBCLC heat exchanger 11, not 12, is preferred for additional Drywell cooling. Plausible because there is no inherent design difference between these heat exchangers themselves.

Technical Reference(s): N1-OP-8, N1-OP-11

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-223001-RBO-8

Question Source: Bank – 2018 NRC #49

Question History: 2018 NRC #49

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(4)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | | |
|---|---|---|
| | Level | RO |
| | Tier # | 1 |
| | Group # | 2 |
| | K/A # | 295014 AK1.09 |
| | Importance Rating | 4.4 |

**INADVERTENT REACTIVITY ADDITION**

**Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to the INADVERTENT REACTIVITY ADDITION: Reactor power**

Proposed Question: #23

The plant is operating at 100% power when the extraction non-return valve to Feedwater Heater 135 trips closed.

Which one of the following describes Reactor power, as indicated by APRMs, two (2) minutes later?

Reactor power will be…

A. 0% due to an automatic scram

B. less than 100%, but no less than 90%

C. unchanged from the initial level

D. greater than 100%

Proposed Answer: D

Explanation: When the extraction non-return valve to Feedwater Heater 135 trips closed, the heater loses heating steam. Feedwater continues to flow through the tube side of the heater, but it receives less heat input. This causes Feedwater temperature into the Reactor to lower. This reduction in temperature, although small, results in a rise in Reactor power. Reactor power will be above 100%, but the power change is too small to challenge the flow-biased scram set point.

A. Incorrect – Reactor power will be above 100%. Plausible because this would be correct for a larger power transient that exceeded the flow-biased scram set point.
B. Incorrect – Reactor power will be above 100%. Plausible because this would be correct for other reactivity transients, such as a lowering Reactor water level.
C. Incorrect – Reactor power will be above 100%. Plausible that the other Feedwater heaters would compensate for or negate the effects of this isolated loss of heating steam.

Technical Reference(s): N1-SOP-16.1

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-SOP16.1C01 EO-2

Question Source: Bank – 2010 Cert #59

Question History: 2005 NRC Retake #62

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(5)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | Level | RO |
|---|---|---|
| | Tier # | 1 |
| | Group # | 2 |
| | K/A # | 295020 AA1.04 |
| | Importance Rating | 3.6 |

**INADVERTENT CONTAINMENT ISOLATION**

**Ability to operate and/or monitor the following as they apply to INADVERTENT CONTAINMENT ISOLATION: SGTS/FRVS**

Proposed Question: #24

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

- Reactor Building Ventilation exhaust fan 11 is operating.
- Reactor Building Ventilation exhaust fan 12 is in standby.
- Reactor Building Emergency Ventilation (RBEVS) is in standby.

Then, Instrument Air is lost to just the following valves:

- 202-32, REACTOR BLDG EXHAUST ISOLATION VALVE 11
- 202-31, REACTOR BLDG EXHAUST ISOLATION VALVE 12

Which one of the following describes the response of RBEVS?

RBEVS…

A. automatically starts and has a suction path to the Reactor Building.

B. automatically starts, but does NOT have a suction path to the Reactor Building.

C. remains in standby. RBEVS can be manually started with a suction path to the Reactor Building.

D. remains in standby. RBEVS CANNOT be manually started with a suction path to the Reactor Building.

Proposed Answer: C

Explanation: The RBVS exhaust air flow path contains two IVs (202-31 and 202-32) in series. These valves fail closed on loss of Instrument Air, isolating the flow path for normal RBVS. The loss of exhaust flow will cause Reactor Building D/P to become less negative and possibly go positive. There is no automatic start of RBEVS in response to either the loss of exhaust flow, exhaust fan trip, or low building D/P. RBEVS takes suction on the RB just upstream of 202-31 and 202-32, therefore these valve closures do not make it unable to restore RB D/P.

A. Incorrect – RBEVS does not automatically start. Plausible that RBEVS would automatically start on the loss of exhaust flow, exhaust fan trip, or low building D/P.
B. Incorrect – RBEVS does not automatically start. Plausible that RBEVS would automatically start on the loss of exhaust flow, exhaust fan trip, or low building D/P.
D. Incorrect – RBEVS can be aligned with a suction path to the Reactor Building. Plausible that the closure of 202-31 and/or 202-32 would block the suction path for RBEVS since the tap off for RBEVS is just upstream of these valves. Also plausible because this would be correct if 202-36 failed closed.

Technical Reference(s): C-18013-C, N1-OP-10

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-288001-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(9)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | | |
|---|---|---|
| | Level | RO |
| | Tier # | 1 |
| | Group # | 2 |
| | K/A # | 295032 2.1.14 |
| | Importance Rating | 3.1 |

**HIGH SECONDARY CONTAINMENT AREA TEMPERATURE**

**Knowledge of criteria or conditions that require plantwide announcements, such as pump starts, reactor trips, and mode changes**

Proposed Question: #25

Given the following events:

(1) The Reactor is scrammed due to a high Reactor Building area temperature

(2) A local area evacuation is required due a chemical spill in Radwaste

(3) The Reactor reaches criticality during a startup

Which one of the following identifies the event the requires a plant-wide announcement and sounding the **Station** Alarm?

A. (1) only

B. (2) only

C. (3) only

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

Proposed Answer: A

Explanation: The Reactor being scrammed due to a high Reactor Building area temperature would require entry into N1-EOP-SC and N1-SOP-1. This requires a plant-wide announcement with the Station Alarm per OP-NM-101-111-0001.

B. Incorrect – Local area evacuation does not require sounding the Station Alarm. Plausible because this does require a plant-wide announcement and sounding the related Evacuation Alarm.
C. Incorrect – Achieving criticality does not require sounding the Station Alarm. Plausible because this does require a plant-wide announcement and is a very significant event.
D. Incorrect – Local area evacuation does not require sounding the Station Alarm. Plausible because this does require a plant-wide announcement and sounding the related Evacuation Alarm. Achieving criticality does not require sounding the Station Alarm. Plausible because this does require a plant-wide announcement and is a very significant event.

Technical Reference(s): OP-NM-201-005, N1-SOP-34, N1-OP-43A, OP-NM-101-111-0001

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-SOP-6.1-CE-02

Question Source: Modified Bank – 2015 Cert #74

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

**Knowledge of criteria or conditions that require plant-wide announcements, such as pump starts, reactor trips, mode changes, etc.**

Proposed Question: #74

Given the following required announcements:

(1) Protected Area Evacuation announcement per CNG-EP-1.01-1009, Assembly, Evacuation, and Accountability.

(2) Medical Emergency announcement per EPIP-EPP-04, Personnel Injury or Illness.

(3) Fire Brigade Activation announcement per EPIP-EPP-28, Fire Fighting.

Which one of the following identifies which of these events require a plant-wide announcement and sounding the **Station** Alarm?

A. Event 1, only

B. Event 2, only

C. Event 3, only

D. Events 1, 2, and 3

Proposed Answer: B

Explanation: All three of these events require a plant announcement with an alarm, but only the Medical Emergency announcement uses the Station Alarm.

A. Plausible – A Protected Area Evacuation announcement requires a plant announcement and alarm, however it uses the Evacuation Alarm, not the Station Alarm.
C. Plausible – The Fire Brigade Activation announcement requires a plant announcement and alarm, however it uses the Fire Alarm, not the Station Alarm.
D. Plausible – All three of these events require a plant announcement with an alarm, but only the Medical Emergency announcement uses the Station Alarm.

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | | |
|---|---|---|
| | Level | RO |
| | Tier # | 1 |
| | Group # | 2 |
| | K/A # | 295035 EA2.03 |
| | Importance Rating | 3.7 |

**SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE**

**Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE: Lights and alarms**

Proposed Question: #26

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

- High winds are occurring.
- Annunciator L1-3-4, REACT BLDG/ATM DIFF PRESS, is in alarm
- Reactor Building differential pressure is -0.05" $H_2O$.
- An Operator has taken manual control of Reactor Building differential pressure per N1-OP-10, Reactor Building Heating, Cooling, and Ventilating System.

Which one of the following describes the required action to restore Reactor Building differential pressure, in accordance with N1-OP-10?

Adjust the Reactor Building differential pressure controller to further…

A. open the Reactor Building supply inlet damper.

B. close the Reactor Building supply inlet damper.

C. open the Reactor Building exhaust outlet damper.

D. close the Reactor Building exhaust outlet damper.

Proposed Answer: B

Explanation: The Reactor Building differential pressure controller is used to modulate the Reactor Building supply inlet damper. The given differential pressure is too low (not negative enough). To make differential pressure more negative, the supply inlet damper must be throttle further closed, restricting the flow of air into the building.

A. Incorrect – The supply inlet damper must be throttled further closed, not open. Plausible because this would be correct if D/P was too negative or if the candidate reverses the cause-effect relationship of this system.
C. Incorrect – This controller controls the supply inlet damper, not the exhaust outlet damper. Plausible because this is another possible way to design building D/P control and the exhaust outlet damper does have automatic function.
D. Incorrect – This controller controls the supply inlet damper, not the exhaust outlet damper. Plausible because this is another possible way to design building D/P control and the exhaust outlet damper does have automatic function.

Technical Reference(s): ARP L1-3-4, N1-OP-10, C-18013-C

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-288001-RBO-3

Question Source: Bank - 2018 NRC #63

Question History: 2018 NRC #63

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(9)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | | |
|---|---|---|
| | Level | RO |
| | Tier # | 2 |
| | Group # | 1 |
| | K/A # | 205000 A2.01 |
| | Importance Rating | 3.6 |

**SHUTDOWN COOLING SYSTEM**

**Ability to (a) predict the impacts of the following on the SHUTDOWN COOLING SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Recirculation loop high temperature**

Proposed Question: #27

A plant shutdown is in progress with the following:

- Shutdown Cooling (SDC) loop 12 is in service.
- Reactor water level is 90".
- Reactor coolant temperature is 340°F.
- Reactor pressure is 115 psig.

Then, a spurious valve closure results in a loss of all Reactor Building Closed Loop Cooling (RBCLC) flow to the SDC heat exchangers.

Which one of the following describes how SDC will respond and the subsequent required control of Reactor Water Cleanup (RWCU) flow, in accordance with N1-SOP-6.1, Loss of Rx Cavity/Decay Heat Removal?

| | SDC Response | Control of RWCU Flow |
|---|---|---|
| A. | SDC isolation valves (38-01, 38-02, 38-13) close and trip the SDC pump | Maximize |
| B. | SDC isolation valves (38-01, 38-02, 38-13) close and trip the SDC pump | Minimize |
| C. | SDC pump trips and SDC isolation valves (38-01, 38-02, 38-13) remain open | Maximize |
| D. | SDC pump trips and SDC isolation valves (38-01, 38-02, 38-13) remain open | Minimize |

Proposed Answer: C

Explanation: With loss of RBCLC flow to the SDC heat exchangers, Reactor coolant temperature and pressure rise. The SDC pumps trip when Reactor coolant temperature reaches 350ºF. The isolation valves remain open. N1-SOP-6.1 requires maximizing RWCU flow to provide additional decay heat removal.

A. Incorrect – The isolation valves remain open. Plausible because these valves cannot be opened until Reactor pressure is below 120 psig, but there is no isolation signal in the opposite direction.
B. Incorrect – The isolation valves remain open. Plausible because these valves cannot be opened until Reactor pressure is below 120 psig, but there is no isolation signal in the opposite direction. N1-SOP-6.1 requires maximizing RWCU flow to provide additional decay heat removal. Plausible that RWCU flow would be lowered to prevent either thermal stratification or high NRHX outlet temperatures (with Reactor coolant temperature rising).
D. Incorrect – N1-SOP-6.1 requires maximizing RWCU flow to provide additional decay heat removal. Plausible that RWCU flow would be lowered to prevent either thermal stratification or high NRHX outlet temperatures (with Reactor coolant temperature rising).

Technical Reference(s): N1-OP-4, N1-SOP-6.1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-205000-RBO-5

Question Source: Bank – 2020 NRC #1

Question History: 2020 NRC #1

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(3)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | | |
|---|---|---|
| | Level | RO |
| | Tier # | 2 |
| | Group # | 1 |
| | K/A # | 206000 K1.17 |
| | Importance Rating | 2.8 |

**HIGH PRESSURE COOLANT INJECTION SYSTEM**

**Knowledge of the physical connections and/or cause and effect relationships between the HIGH PRESSURE COOLANT INJECTION SYSTEM and the following systems: Reactor protection system**

Proposed Question: #28

The plant is operating at 100% power when RPS bus 11 de-energizes due to a sustained electrical fault.

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

A. Both Feedwater flow control valves 11 and 12 shift to HPCI mode.

B. Feedwater flow control valve 11 shifts to HPCI mode, but Feedwater flow control valve 12 does NOT.

C. NEITHER Feedwater flow control valve 11 NOR 12 shift to HPCI mode. Both Feedwater flow control valves 11 and 12 maintain the ability to shift to HPCI mode.

D. NEITHER Feedwater flow control valve 11 NOR 12 shift to HPCI mode. Feedwater flow control valve 11 loses the ability to shift to HPCI mode, but Feedwater flow control valve 12 does NOT.

Proposed Answer: C

Explanation: Feedwater level control system channel 11 is normally powered from RPS Bus 11, but automatically swaps to MG Set 167 on this loss. RPS Buses 11 and 12 supply power to the HPCI initiation logic. HPCI initiation is energize to function. Either relay 11K47 (RPS Bus 11) or 12K47 (RPS Bus 12) must energize for HPCI initiation. Only one of the two power supplies is required. With only a loss of RPS Bus 11, no HPCI initiation is received and both Feedwater flow control valves (FCVs) maintain the ability to shift to HPCI mode with just power from RPS Bus 12.

A. Incorrect – Neither FCV goes into HPCI mode. Plausible because some systems are de-energize to function.
B. Incorrect – Neither FCV goes into HPCI mode. Plausible because some systems are de-energize to function. Also plausible because just one relay is necessary to 11K47 (RPS Bus 11) or 12K47 (RPS Bus 12) initiate HPCI for both FCVs.
D. Incorrect – FCV 11 maintains the ability to shift to HPCI mode. Plausible because 11K47 cannot energize. Also plausible because the normal power supply to Feedwater level control channel 11 is RPS bus 11.

Technical Reference(s): N1-OP-16; C-19859-C sheets 3, 6; C-23077-C sheet 4

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-259002-RBO-8

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| | | |
|---|---|---|
| Examination Outline Cross-Reference: | Level | RO |
| | Tier # | 2 |
| | Group # | 1 |
| | K/A # | 207000 K6.07 |
| | Importance Rating | 3.5 |

**ISOLATION (EMERGENCY) CONDENSER**

**Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the ISOLATION (EMERGENCY) CONDENSER: AC electrical distribution**

Proposed Question: #29

The plant is operating at 100% power when Powerboard 16B de-energizes due to a sustained electrical fault.

Which one of the following describes the impact of this failure on the Emergency Condensers (ECs)?

A. A Condensate Return Valve fails open

B. A Condensate Return Valve fails closed

C. A Steam Isolation Valve loses motive power while in the open position

D. A Steam Isolation Valve loses motive power while in the closed position

Proposed Answer: C

Explanation: Powerboard 16B supplies power to Powerboard 161B, which is the power supply to 39-10R, EC STM ISOLATION VALVE 121. This valve is open while in the standby lineup.

A. Incorrect – The Condensate Return Valves are unaffected. Plausible because they do fail open on loss of air or loss of both DC solenoids.
B. Incorrect – The Condensate Return Valves are unaffected. Plausible because they do fail open on loss of air or loss of both DC solenoids. Plausible that they would fail closed to prevent spurious system initiation.
D. Incorrect – With the plant at 100% power, ECs are in a standby lineup and the Steam Isolation Valves are open. Plausible because the Condensate Return Valves are closed.

Technical Reference(s): C-19437-C Sheet 6, C-18017-C Sheet 1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-207000-RBO-8

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| | | |
|---|---|---|
| Examination Outline Cross-Reference: | Level | RO |
| | Tier # | 2 |
| | Group # | 1 |
| | K/A # | 209001 K3.01 |
| | Importance Rating | 4.3 |

**LOW PRESSURE CORE SPRAY SYSTEM**

**Knowledge of the effect that a loss or malfunction of the LOW PRESSURE CORE SPRAY SYSTEM will have on the following systems or system parameters: Reactor water level**

Proposed Question: #30

A loss of coolant accident occurs with the following:

- Reactor water level is -50” and lowering.
- Reactor pressure is 300 psig and lowering.
- Core Spray pump 111 is tagged out of service for motor replacement.
- Core Spray pump 121 has tripped and CANNOT be restarted.
- Powerboard 171B is de-energized due to a sustained electrical fault.

Which one of the following describes the availability of Core Spray loops 11 and 12 for Reactor water level control?

| | Injection with Core Spray Loop 11 | Injection with Core Spray Loop 12 |
|---|---|---|
| A. | Available | Available |
| B. | Available | NOT available |
| C. | NOT available | Available |
| D. | NOT available | NOT available |

Proposed Answer: A

Explanation: Powerboard 171B supplies power to Core Spray discharge inside IVs 40-10 (loop 11) and 40-09 (loop 12). Powerboard 161B supplies power to Core Spray discharge inside IVs 40-01 (loop 11) and 40-11 (loop 12). All of these valves are inside the Drywell and normally closed, such that they need electrical power to open and inject to the Reactor. Core Spray pump 111 is in loop 11 and Core Spray pump 121 is in loop 12. Loop 11 is still capable of injecting to the Reactor with pump 112 through inside IV 40-11. Loop 12 is still capable of injecting to the Reactor with pump 122 through inside IV 40-01.

B. Incorrect – Both Core Spray loops are still available for Reactor water level control. Plausible that the loss of Powerboard 171B, combined with the loss of Core Spray pump 111, would result in inability to inject with Core Spray loop 11.
C. Incorrect – Both Core Spray loops are still available for Reactor water level control. Plausible that the loss of Powerboard 171B, combined with the loss of Core Spray pump 111, would result in inability to inject with Core Spray loop 12.
D. Incorrect – Both Core Spray loops are still available for Reactor water level control. Plausible that the loss of Powerboard 171B, combined with the loss of Core Spray pumps 111 and 121, would result in inability to inject with both Core Spray loops.

Technical Reference(s): N1-OP-2, C-18007-C

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-209001-RBO-11

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | | |
|---|---|---|
| | Level | RO |
| | Tier # | 2 |
| | Group # | 1 |
| | K/A # | 209001 K5.02 |
| | Importance Rating | 3.4 |

**LOW PRESSURE CORE SPRAY SYSTEM**

**Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the LOW PRESSURE CORE SPRAY SYSTEM: Abnormal differential pressure indication (leak detection)**

Proposed Question: #31

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

- Annunciator K3-4-1, CORE SPRAY SPARGER DIFF PRESS, alarms.
- The alarm is determined to be due to a valid line break on Core Spray 11.

Which one of the following describes the location of the Core Spray line break?

A. Between the pumps and the discharge check valve (40-13)

B. Between the discharge check valve (40-13) and the discharge inside isolation valves (40-10 and 40-11)

C. Between the discharge inside isolation valves (40-10 and 40-11) and the core shroud

D. Inside the core shroud

Proposed Answer: C

Explanation: This alarm is caused by a change in pressure due to a leak between the injection line check valve and the core shroud.

A. Incorrect – This alarm would not detect a break in this location because closed valves prevent a change in pressure here from being detected. Plausible because this leak location is possible and would disrupt Core Spray flow in the same way as a leak in the correct location.
B. Incorrect – This alarm would not detect a break in this location because closed valves prevent a change in pressure here from being detected. Plausible because this leak location is possible and would disrupt Core Spray flow in the same way as a leak in the correct location.
D. Incorrect – This alarm would not detect a break in this location because there would be no pressure change (this section of piping is already pressurized to the same pressure as the surrounding environment). Plausible because this section of piping is directly connected to the piping that is monitored by the leak detection system.

Technical Reference(s): ARP K3-4-1, C-18007-C, C-18015-C, N1-209001-RBO-5

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-209001-RBO-5

Question Source: Bank – JAF 18-1 NRC #32

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| | | |
|---|---|---|
| Examination Outline Cross-Reference: | Level | RO |
| | Tier # | 2 |
| | Group # | 1 |
| | K/A # | 211000 A1.05 |
| | Importance Rating | 2.9 |

**STANDBY LIQUID CONTROL SYSTEM**

**Ability to predict and/or monitor changes in parameters associated with operation of the STANDBY LIQUID CONTROL SYSTEM including: Pump amps**

Proposed Question: #32

A failure to scram has occurred with the following:

- A Reactor Operator has placed the Liquid Poison keylock switch to the SYS 11 position.
- Reactor pressure is 1020 psig and stable.
- Liquid Poison header pressure is 0 psig.
- Liquid Poison pump 11 amps indicate upscale.
- The LIQUID POISON EXPL VALVE 11 CONTINUITY light is extinguished.
- The LIQUID POISON EXPL VALVE 12 CONTINUITY light is extinguished.
- NO change in Liquid Poison tank level has been observed.

Which one of the following describes the malfunction indicated by these conditions?

A. Seized pump rotor

B. Sheared pump shaft

C. Relief valve has lifted

D. Explosive valves failed to open

Proposed Answer: A

Explanation: The given indications show that the Liquid Poison pump 11 motor is drawing very high amps while the pump is not producing pressure. This indicates that the pump rotor is not spinning (locked or seized rotor).

B. Incorrect – A sheared pump shaft would result in lower than normal amps. Plausible because this would result in low discharge pressure and lack of lowering in tank level.
C. Incorrect – A lifting relief valve would not result in upscale amps or such low discharge pressure. Plausible because this would result in lack of lowering tank level and would have some effect on discharge pressure and/or pump amps.
D. Incorrect – Failure of explosive valves to open would not result in low discharge pressure or such high amps. Plausible because this would result in lack of lowering tank level and would have some effect on discharge pressure and/or pump amps.

Technical Reference(s): N1-OP-12, BC05Ir4_Motors

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-211000-RBO-6

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(6)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | | |
|---|---|---|
| | Level | RO |
| | Tier # | 2 |
| | Group # | 1 |
| | K/A # | 212000 A3.09 |
| | Importance Rating | 4.3 |

**REACTOR PROTECTION SYSTEM**

**Ability to monitor automatic operation of the REACTOR PROTECTION SYSTEM including: System actuation**

Proposed Question: #33

A plant startup is in progress with the following:

- The Reactor Mode Switch is in STARTUP.
- All IRMs indicate approximately 30 out of 125 on Range 8.
- APRMs indicate 3%.
- Reactor pressure is 550 psig.

Then, the MSIVs close due to a spurious isolation signal, resulting in the following conditions:

- IRMs spike to 100 out of 125 on Range 8.
- APRMs spike to 7%.
- Reactor pressure spikes to 650 psig.

Which one of the following describes the RPS signal that initiates a Reactor scram, if any?

A. IRM Hi-Hi

B. APRM Hi-Hi

C. MSIV Closure

D. NO Reactor scram occurs

Proposed Answer: C

Explanation: IRMs do not cause a Reactor scram on a Hi-Hi signal because they remain below 117.5 out of 125 on Range 8. APRMs do not spike high enough to cause a Reactor scram given the startup conditions. MSIV Closure does cause a direct scram due to Reactor pressure being above 600 psig with MSIVs closed.

A. Incorrect – MSIV Closure initiates the Reactor scram. Plausible because this would be correct if IRMs exceeded 117.5. Also plausible because IRMs spike well above the normal control band.
B. Incorrect – MSIV Closure initiates the Reactor scram. Plausible because APRMs spike high enough to clear the downscales.
D. Incorrect – MSIV Closure initiates the Reactor scram. Plausible because this would be correct if Reactor pressure remained below 600 psig.

Technical Reference(s): ARPs F3-1-1, F3-3-1, F3-3-6

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-212000-RBO-5

Question Source: Modified Bank – 2020 NRC #43

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Proposed Question: #43

A plant startup is in progress with the following:

- The Reactor Mode Switch is in STARTUP.
- All IRMs indicate approximately 30 out of 125 on Range 8
- APRMs indicate 3%
- Reactor pressure is 450 psig

Then, the MSIVs close due to a spurious isolation signal, resulting in the following conditions:

- IRMs spike to 120 out of 125 on Range 8
- APRMs spike to 9%
- Reactor pressure spikes to 550 psig

Which one of the following describes the RPS signal that initiates a Reactor scram, if any?

A. IRM Hi-Hi

B. APRM Hi-Hi

C. MSIV Closure

D. NO Reactor scram occurs

Proposed Answer: A

Explanation: IRMs cause a Reactor scram on a Hi-Hi signal due to exceeding 117.5 out of 125 on Range 8. APRMs do not spike high enough to cause a Reactor scram given the startup conditions. MSIV Closure does not cause a direct scram due to Reactor pressure being below 600 psig.

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | | |
|---|---|---|
| | Level | RO |
| | Tier # | 2 |
| | Group # | 1 |
| | K/A # | 215003 K3.01 |
| | Importance Rating | 4.2 |

**INTERMEDIATE RANGE MONITOR SYSTEM**

**Knowledge of the effect that a loss or malfunction of the INTERMEDIATE RANGE MONITOR SYSTEM will have on the following systems or system parameters: RPS**

Proposed Question: #34

A plant startup is in progress with the following:

- The Reactor Mode Switch is in STARTUP.
- All Intermediate Range Monitors (IRM) detectors are fully inserted and indicating mid-scale on range 2.

Then, IRM 15 fails upscale.

Which one of the following describes the response of the Reactor Protection System?

A. NO half scram is received

B. A half scram is received on RPS 11, only

C. A half scram is received on RPS 12, only

D. A full scram is received

Proposed Answer: C

Explanation: The upscale failure results in an upscale trip from IRM 15. The RPS logic is arranged such that an upscale trip from a single IRM causes a half scram. IRMs 11-14 input to RPS 11 and IRMs 15-18 input to RPS 12. Therefore, a half scram is received on RPS 12 only.

A. Incorrect – A half scram is received on RPS 12, only. Plausible that is would take more than 1 failed IRM to result in a half scram (there are 4 IRM inputs to RPS 11 and 4 IRM inputs to RPS 12). Also plausible because this would be correct if the Reactor Mode Switch was in RUN.
B. Incorrect – A half scram is received on RPS 12, only. Plausible that the odd numbered IRMs would all input to RPS 11.
D. Incorrect – A half scram is received on RPS 12, only. Plausible because this would be correct if IRM 11, 12, 13, or 14 also failed upscale or INOP.

Technical Reference(s): N1-OP-38B, ARP F2-3-6, C-19859-C Sheets 2 and 5

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-215000-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | | |
|---|---|---|
| | Level | RO |
| | Tier # | 2 |
| | Group # | 1 |
| | K/A # | 215004 A1.01 |
| | Importance Rating | 3.3 |

**SOURCE RANGE MONITOR SYSTEM**

**Ability to predict and/or monitor changes in parameters associated with operation of the SOURCE RANGE MONITOR SYSTEM including: Detector position**

Proposed Question: #35

A plant startup is in progress with the following:

- All IRMs are on Range 2.
- SRM 11 reads 9 cps with the detector fully inserted.
- SRM 12 reads 110 cps with the detector partially withdrawn.
- SRM 13 reads 95 cps with the detector partially withdrawn.
- SRM 14 reads 15 cps with the detector fully inserted.
- A control rod block is present.

Which one of the following identifies all of the SRMs that are causing the control rod block?

A. SRM 11 only

B. SRM 13 only

C. SRM 11 and 14

D. SRM 11, 13, and 14

Proposed Answer: B

Explanation: SRM indication less than 100 cps causes a rod block if the associated detector is not fully inserted. Therefore SRM 13 is causing a rod block based on not being fully inserted and <100 cps.

A. Incorrect – SRM 11 is not causing a rod block because it is fully inserted. Plausible because it is the lowest reading SRM and would be causing a rod block if it was not fully inserted.
C. Incorrect – SRMs 11 and 14 are not causing a rod block because they are fully inserted. Plausible because they are the lowest reading SRMs and would be causing a rod block if they were not fully inserted.
D. Incorrect – SRMs 11 and 14 are not causing a rod block because they are fully inserted. Plausible because they are the lowest reading SRMs and would be causing a rod block if they were not fully inserted.

Technical Reference(s): N1-OP-38A

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-215000-RBO-5

Question Source: Bank – 2013 NRC #37

Question History: 2013 NRC #37

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | | |
|---|---|---|
| | Level | RO |
| | Tier # | 2 |
| | Group # | 1 |
| | K/A # | 263000 A1.01 |
| | Importance Rating | 3.3 |

**DC Electrical Distribution**

**Ability to predict and/or monitor changes in parameters associated with operation of the DC Electrical Distribution, including: Battery charging/discharging rate**

Proposed Question: #36

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

- Static Battery Chargers 161A and 171B are in service.
- Powerboard 16B is inadvertently de-energized, then re-energized.

Which one of the following identifies the status of Batteries 11 and 12 two (2) minutes after Powerboard 16B is re-energized?

| | Battery 11 | Battery 12 |
|---|---|---|
| A. | Charging | Charging |
| B. | Charging | Discharging |
| C. | Discharging | Charging |
| D. | Discharging | Discharging |

Proposed Answer: A

Explanation: Static Battery Charger 161A receives power from Powerboard 16B and charges Battery 11. Upon loss of Powerboard 16B, Battery 11 begins discharging. Approximately 90 seconds after Powerboard 16B is re-energized, Static Battery Charger 161A will automatically align itself to Battery 11 and restore the normal float charge. Static Battery Charger 171B receives power from Powerboard 17B and charges Battery 12. Therefore, Battery 12 is in the normal float charge throughout the transient.

B. Incorrect – Battery 12 is on the normal float charge throughout the transient. Plausible that Powerboard 16B would power one battery charger for each battery for added redundance/flexibility.
C. Incorrect – By 2 minutes, Battery 11 is back to charging. Plausible because Battery 11 was discharging. Also plausible that manual action would be required to restore charging.
D. Incorrect – By 2 minutes, Battery 11 is back to charging. Plausible because Battery 11 was discharging. Also plausible that manual action would be required to restore charging. Battery 12 is on the normal float charge throughout the transient. Plausible that Powerboard 16B would power one battery charger for each battery for added redundance/flexibility.

Technical Reference(s): N1-OP-47A

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-263000-RBO-3

Question Source: Bank – 2010 NRC #13

Question History: 2010 NRC #13

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | | |
|---|---|---|
| | Level | RO |
| | Tier # | 2 |
| | Group # | 1 |
| | K/A # | 215005 K3.07 |
| | Importance Rating | 3.7 |

**AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR**

**Knowledge of the effect that a loss or malfunction of the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR will have on the following systems or system parameters: Rod block monitor system**

Proposed Question: #37

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

- Rod line is 102%.
- APRM FLOW UNIT 12 signal fails downscale.
- Actual core flow remains constant.

Which one of the following describes the plant response?

A. NO rod block or half scram is received.

B. A rod block is received, but NO half scram is received.

C. A rod block and a half scram are received. A full scram is NOT received.

D. A rod block and full scram are received.

Proposed Answer: C

Explanation: APRM 12 flow unit provides the flow input to ARPMs 15-18, which all input to RPS channel 12 only. With their flow signal failed low, 100% Reactor power exceeds both their flow-biased rod block and scram setpoints. Therefore, both a rod block and a half scram on RPS channel 12 occur. A flow comparator rod block also occurs because the difference between flow channels 11 and 12 exceeds 6.8%.

A. Incorrect – A rod block and half scram occurs. Plausible because the flow-biased rod block and half scram would not be received for an upscale failure of the flow unit.
B. Incorrect – A half scram also occurs. Plausible because this would be correct for an upscale failure of the flow unit.
D. Incorrect – A full scram does not occur. Plausible that this flow unit would input to both some RPS 11 and some RPS 12 APRMs.

Technical Reference(s): N1-OP-38C, 5-loop Power/Flow Map, ARPs F4-1-8, F3-1-1, F3-2-1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-215000-RBO-11

Question Source: Bank – 2013 NRC #38

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| | | |
|---|---|---|
| Examination Outline Cross-Reference: | Level | RO |
| | Tier # | 2 |
| | Group # | 1 |
| | K/A # | 218000 A4.01 |
| | Importance Rating | 4.5 |

**AUTOMATIC DEPRESSURIZATION SYSTEM**

**Ability to manually operate and/or monitor the AUTOMATIC DEPRESSURIZATION SYSTEM in the control room: ADS valves**

Proposed Question: #38

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

- Reactor water level is -20" and lowering slowly.
- Reactor pressure is 650 psig and lowering slowly.
- Drywell pressure is 10 psig and rising slowly.
- Annunciator F2-3-1, MAIN STM LINE AUTOMATIC DE-PRESS TIMING, came into alarm 30 seconds ago.
- The Unit Supervisor has directed bypassing ADS.
- A Reactor Operator has placed the Channel 12 ADS Inhibit keylock bypass switch in BYPASS.
- The Channel 11 ADS Inhibit keylock bypass switch CANNOT be placed in BYPASS due to a mechanical malfunction.

Which one of the following describes the status of ERVs two (2) minutes later?

A. NO ERVs are open

B. The three (3) primary ERVs are open, only

C. The three (3) secondary ERVs are open, only

D. All six (6) ERVs are open

Proposed Answer: B

Explanation: ADS channel 12 has successfully been bypassed, but channel 11 has not. ADS only needs one of these two channels to function to cause the three (3) primary ERVs to open after 111 seconds. Since the primary ERVs open, the three (3) backup ERVs remain closed.

A. Incorrect – The three (3) primary ERVs open. Plausible that both channels would be required to open ERVs. Also plausible because this would be correct prior to 111 seconds.
C. Incorrect – The three (3) primary ERVs open. Plausible because there are some malfunctions that would cause the primary ERVs to not open, in which case the backup ERVs open.
D. Incorrect – Only three (3) ERVs open. Plausible because both the primary and backup ERVs are capable of opening and EOPs require >3 ERVs to open for an RPV Blowdown.

Technical Reference(s): C-19859-C sheets 18, 18a, 24, 24a, 25, 25a, 26, 26a

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-218000-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | | |
|---|---|---|
| | Level | RO |
| | Tier # | 2 |
| | Group # | 1 |
| | K/A # | 223002 A2.13 |
| | Importance Rating | 3.3 |

**PRIMARY CONTAINMENT ISOLATION SYSTEM / NUCLEAR STEAM SUPPLY SHUTOFF**

**Ability to (a) predict the impacts of the following on the PRIMARY CONTAINMENT ISOLATION SYSTEM / NUCLEAR STEAM SUPPLY SHUTOFF and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Leak detection system failures**

Proposed Question: #39

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

- A failure of multiple Main Steam Line temperature detectors occurs.
- The failures result in an isolation signal closing the MSIVs.
- The isolation signal is sealed in and CANNOT be cleared.
- N1-SOP-1, Reactor Scram, and N1-EOP-RPV, RPV Control, are entered.
- You are directed to perform a Reactor cooldown.

Assuming the isolation signal remains sealed in, which one of the following describes the ability to:

- use Emergency Condensers for the cooldown

  and

- place Shutdown Cooling in service once the Shutdown Cooling pressure interlock clears?

| | Emergency Condensers | Shutdown Cooling |
|---|---|---|
| A. | Available for cooldown | Will be available |
| B. | Available for cooldown | Will NOT be available |
| C. | NOT available for cooldown | Will be available |
| D. | NOT available for cooldown | Will NOT be available |

Proposed Answer: A

Explanation: Failure of the Main Steam leak detection system (Main Steam tunnel temperature detectors) has caused an isolation of the MSIVs. This is a Vessel isolation. With the MSIVs closed, TBVs are unavailable for pressure control. This Vessel isolation also causes Emergency Condenser vents and drains to close. However, this does not prevent use of the Emergency Condensers for cooldown. Additionally, this Vessel isolation does not isolate Shutdown Cooling, therefore Shutdown Cooling will be available to be placed in service once the pressure interlock clears even if the Main Steam Line temperature isolation signal remains sealed it.

B. Incorrect – Shutdown Cooling will be available to be placed in service once the pressure interlock clears. Plausible because the Shutdown Cooling system isolates on a Vessel isolation signal from Lo-Lo Reactor water level, just not this Vessel isolation signal.
C. Incorrect – Emergency Condensers are available for cooldown. Plausible because some Emergency Condenser valves do close on this isolation signal. Also plausible because Emergency Condensers do isolate on their own, separate leak detection isolation signal.
D. Incorrect – Emergency Condensers are available for cooldown. Plausible because some Emergency Condenser valves do close on this isolation signal. Also plausible because Emergency Condensers do isolate on their own, separate leak detection isolation signal. Shutdown Cooling will be available to be placed in service once the pressure interlock clears. Plausible because the Shutdown Cooling system isolates on a Vessel isolation signal from Lo-Lo Reactor water level, just not this Vessel isolation signal.

Technical Reference(s): N1-SOP-40.2

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-223002-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| | | |
|---|---|---|
| Examination Outline Cross-Reference: | Level | RO |
| | Tier # | 2 |
| | Group # | 1 |
| | K/A # | 223002 K6.09 |
| | Importance Rating | 3.2 |

**PRIMARY CONTAINMENT ISOLATION SYSTEM / NUCLEAR STEAM SUPPLY SHUTOFF**

**Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the PRIMARY CONTAINMENT ISOLATION SYSTEM / NUCLEAR STEAM SUPPLY SHUTOFF: Plant pneumatic systems**

Proposed Question: #40

Which one of the following describes the effect of a complete loss of Instrument Air on the Main Steam Isolation Valves (MSIVs)?

A. The inboard MSIVs fail open

B. The inboard MSIVs fail closed

C. The outboard MSIVs fail open

D. The outboard MSIVs fail closed

Proposed Answer: D

Explanation: The outboard MSIVs are operated by Instrument Air. They fail closed on loss of Instrument Air.

A. Incorrect – The inboard MSIVs are not operated by Instrument Air. Plausible because the outboard MSIVs are operated by Instrument Air.
B. Incorrect – The inboard MSIVs are not operated by Instrument Air. Plausible because the outboard MSIVs are operated by Instrument Air. MSIVs fail closed on loss of Instrument Air. Plausible because many air-operated valves fail open on loss of the air supply.
C. Incorrect – MSIVs fail closed on loss of Instrument Air. Plausible because many air-operated valves fail open on loss of the air supply.

Technical Reference(s): N1-SOP-20.1, C-18002-C Sheet 1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-223002-RBO-8

Question Source: Bank - 2015 Cert #32

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(9)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | | |
|---|---|---|
| | Level | RO |
| | Tier # | 2 |
| | Group # | 1 |
| | K/A # | 239002 A3.01 |
| | Importance Rating | 4.1 |

**SAFETY RELIEF VALVES**

**Ability to monitor automatic operation of the SAFETY RELIEF VALVES including: SRV operation after ADS actuation**

Proposed Question: #41

Given the following indications for ERV 111:




Which one of the following describes the status of ERV 111 indications?

ERV 111…

A. has normal indications for a valve that cycled open and then closed

B. indicates open by acoustic monitor but closed by solenoid position

C. indicates open by solenoid position but closed by acoustic monitor

D. indicates open by both acoustic monitor and solenoid position

Proposed Answer: C

Explanation: The red light above the ERV control switch is illuminated, which indicates that the valve is open by solenoid position. The red light to the right of the ERV control switch is extinguished, which indicates that the valve is closed by acoustic monitor. These conditions would occur if an ERV received an open signal, such as from ADS, with Reactor pressure too low to open the valve.

A. Incorrect – The ERV indicates open by solenoid position. Plausible because this would be correct with different light indications.
B. Incorrect – The ERV indicates open by solenoid position and closed by acoustic monitor. Plausible because this would be correct with different light indications.
D. Incorrect – The ERV indicates open by solenoid position and closed by acoustic monitor. Plausible because this would be correct with different light indications.

Technical Reference(s): N1-OP-1

Proposed references to be provided to applicants during examination: Embedded graphic

Learning Objective: N1-218000-RBO-5

Question Source: Modified Bank - JAF 20-1 NRC #15

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:

Proposed Question: #15

The plant is operating at 100% power with the following indications for SRV A:




Which one of the following describes the status of the SRV A indications?

SRV A...

A. has normal indications for a valve that cycled open then closed.

B. indicates open by acoustic monitor but closed by solenoid position.

C. indicates open by solenoid position but closed by acoustic monitor.

D. indicates open by both acoustic monitor and solenoid position.

Proposed Answer: B

Explanation: The SRV A white light being illuminated indicates the valve is open as sensed by the acoustic monitor. However, the green light being illuminated and the red light being extinguished indicate that the SRV A is not open based on solenoid position.

**Form 4.2-1 Written Examination Question Worksheet**

| | | |
|---|---|---|
| Examination Outline Cross-Reference: | Level | RO |
| | Tier # | 2 |
| | Group # | 1 |
| | K/A # | 239002 2.1.45 |
| | Importance Rating | 4.3 |

**SAFETY RELIEF VALVES**

**Ability to identify and interpret diverse indications to validate the response of another indication**

Proposed Question: #42

The plant is operating at 23% power during a startup, with the following conditions:

- Five Turbine Bypass Valves are open.
- The crew is preparing to roll the Turbine.
- Reactor pressure is 930 psig and stable.

Then, ERV 111 inadvertently opens and remains open.

Which one of the following describes:

- the ability of the Main Steam Line flow instruments to detect the flow through ERV 111

  and

- the response of Reactor pressure?

| | The Main Steam Line flow instruments… | Reactor pressure… |
|---|---|---|
| A. | detect the steam flow through ERV 111 | stabilizes below 930 psig |
| B. | do NOT detect the steam flow through ERV 111 | stabilizes below 930 psig |
| C. | detect the steam flow through ERV 111 | continually lowers as long as the ERV is open |
| D. | do NOT detect the steam flow through ERV 111 | continually lowers as long as the ERV is open |

Proposed Answer: A

Explanation: The ERVs tap off of the main steam lines after the flow restrictors. The flow restrictors are used to measure main steam line flow. Since the ERV flow passes through the flow restrictor, the ERV flow is measured as part of the total main steam line flow. The opening of the ERV lowers pressure at the turbine. EPR sensed pressure lowers, therefore the bypass valves are throttled closed. This action stabilizes pressure below the initial value of 930 psig while ERV 111 is open.

B. Incorrect – ERV flow is measured as part of the total main steam line flow. Plausible if the ERV tapped off the main steam line upstream of the flow restrictor instead of downstream.
C. Incorrect – Reactor pressure stabilizes below the initial value of 930 psig. Plausible because this would be correct if EHC did not sense the problem and adjust. Also plausible because this would be correct if the Reactor was shutdown.
D. Incorrect – ERV flow is measured as part of the total main steam line flow. Plausible if the ERV tapped off the main steam line upstream of the flow restrictor instead of downstream. Reactor pressure stabilizes below the initial value of 930 psig. Plausible because this would be correct if EHC did not sense the problem and adjust. Also plausible because this would be correct if the Reactor was shutdown.

Technical Reference(s): C-18002-C Sheet 1, N1-SOP-1.4

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-239001-RBO-8

Question Source: Bank – 2017 Cert #24

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(3)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | | |
|---|---|---|
| | Level | RO |
| | Tier # | 2 |
| | Group # | 1 |
| | K/A # | 259002 K2.03 |
| | Importance Rating | 3.1 |

**REACTOR WATER LEVEL CONTROL SYSTEM**

**Knowledge of electrical power supplies to the following: Feedwater controllers**

Proposed Question: #43

Which one of the following describes the normal power supply(ies) to the Feedwater Level Control RTP chassis?

A. I&C Bus 130

B. Computer Panel 167

C. RPS Buses 11 and 12

D. Battery Boards 11 and 12

Proposed Answer: C

Explanation: The Feedwater Level Control RTP chassis receive normal power from RPS buses 11 and 12.

A. Incorrect – The Feedwater Level Control RTP chassis receive normal power from RPS buses 11 and 12. Plausible because I&C Bus 130 supplies power to numerous instruments and controls.
B. Incorrect – The Feedwater Level Control RTP chassis receive normal power from RPS buses 11 and 12. Plausible because Computer Panel 167 supplies backup power to some Feedwater level control components.
D. Incorrect – The Feedwater Level Control RTP chassis receive normal power from RPS buses 11 and 12. Plausible because Battery Boards 11 and 12 supply power to many instruments and controls.

Technical Reference(s): N1-259002-RBO-4

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-259002-RBO-4

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(4)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | | |
|---|---|---|
| | Level | RO |
| | Tier # | 2 |
| | Group # | 1 |
| | K/A # | 510000 K1.13 |
| | Importance Rating | 2.7 |

**SERVICE WATER SYSTEM**

**Knowledge of the physical connections and/or cause and effect relationships between the Service Water System and the following systems: Feedwater system**

Proposed Question: #44

The plant has been scrammed with the following:

- A loss of both Service Water pumps has occurred.
- Both Emergency Service Water (ESW) pumps have been started.

Which one of the following describes whether ESW is supplying cooling to Feedwater pumps 11 and 12?

ESW is…

A. NOT supplying cooling to Feedwater pumps 11 and 12

B. directly supplying cooling to Feedwater pumps 11 and 12

C. indirectly supplying cooling to Feedwater pumps 11 and 12 through TBCLC

D. indirectly supplying cooling to Feedwater pumps 11 and 12 through RBCLC

Proposed Answer: D

Explanation: Feedwater pumps 11 and 12 are directly cooled by RBCLC. RBCLC is normally cooled by Service Water. With Service Water pumps unavailable, ESW pumps are started to supply an alternate source of cooling water to RBCLC. Therefore, ESW is indirectly cooling Feedwater pumps 11 and 12 through RBCLC.

A. Incorrect – ESW is indirectly cooling Feedwater pumps 11 and 12 through RBCLC. Plausible because ESW is NOT supplying cooling to Feedwater pump 13 (it is cooled by TBCLC, which normally is cooled by Service Water, but does not get ESW flow).
B. Incorrect – ESW is indirectly cooling Feedwater pumps 11 and 12 through RBCLC. Plausible because ESW flow is different than normal Service Water flow. Also plausible because Service Water does directly cool some components (including Feedwater pump area cooler).
C. Incorrect – ESW is indirectly cooling Feedwater pumps 11 and 12 through RBCLC. Plausible because some Feedwater components are normally cooled via Service Water through TBCLC (including Feedwater pump 13).

Technical Reference(s): N1-SOP-18.1, C-18022-C Sheet 1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-276000-RBO-11

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | | |
|---|---|---|
| | Level | RO |
| | Tier # | 2 |
| | Group # | 1 |
| | K/A # | 261000 K2.04 |
| | Importance Rating | 2.9 |

**STANDBY GAS TREATMENT SYSTEM**

**Knowledge of electrical power supplies to the following: Standby gas treatment system heaters**

Proposed Question: #45

Which one of the following is the power supply to the Reactor Building Emergency Ventilation System (RBEVS) 10 KW heater?

A. Powerboard H16

B. Powerboard 151

C. Powerboard 153

D. Powerboard 167

Proposed Answer: D

Explanation: The power supply to the RBEVS 10 KW heater is Powerboard 167.

A. Incorrect – The power supply to the RBEVS 10 KW heater is Powerboard 167. Plausible because this is the power supply to multiple Reactor Building duct heaters.
B. Incorrect – The power supply to the RBEVS 10 KW heater is Powerboard 167. Plausible because this is the power supply to Reactor Building Ventilation supply fans.
C. Incorrect – The power supply to the RBEVS 10 KW heater is Powerboard 167. Plausible because this is the power supply to Reactor Building Ventilation exhaust fans.

Technical Reference(s): N1-261000-RBO-4

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-261000-RBO-4

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(9)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| | | |
|---|---|---|
| Examination Outline Cross-Reference: | Level | RO |
| | Tier # | 2 |
| | Group # | 1 |
| | K/A # | 262001 A3.03 |
| | Importance Rating | 3.8 |

**AC ELECTRICAL DISTRIBUTION**

**Ability to monitor automatic operation of the AC ELECTRICAL DISTRIBUTION including: Load shedding**

Proposed Question: #46

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

- The following pumps are running:
  - CRD pump 12
  - Liquid Poison pump 12
  - RBCLC pump 12

Then, a loss of offsite power occurs. Both Emergency Diesel Generators (EDGs) start and re-energize their respective powerboards.

Which one of the following describes the status of CRD pump 12, Liquid Poison pump 12, and RBCLC pump 12 one (1) minute after the EDGs re-energize their respective powerboards?

A. All three of these pumps are running.

B. CRD pump 12 and Liquid Poison pump 12 are running. RBCLC pump 12 is shut down.

C. Liquid Poison pump 12 and RBCLC pump 12 are running. CRD pump 12 is shut down.

D. All three of these pumps are shut down.

Proposed Answer: B

Explanation: 86-16 and 86-17 actuate on the loss of power to PBs 102 and 103 to initiate load shedding to reduce the loading on the diesels when they start and energize their busses. Liquid Poison pumps are NOT stripped, therefore Liquid Poison pump 12 is operating. CRD pumps are both started, even if initially in standby, therefore both CRD pumps 11 and 12 are operating. RBCLC pumps are stripped, therefore RBCLC pump 12 is NOT operating.

A. Incorrect – RBCLC pump 12 is NOT operating. Plausible because it was initially operating and its cooling function is important. An RBCLC pump is the first load restarted in N1-SOP-33A.1 while restoring from this transient.
C. Incorrect – RBCLC pump 12 is NOT operating. Plausible because it was initially operating and its cooling function is important. An RBCLC pump is the first load restarted in N1-SOP-33A.1 while restoring from this transient. CRD pump 12 is operating. Plausible because CRD pump 12 de-energizes during the transient before the EDGs re-energize their buses and load shedding does occur.
D. Incorrect – Liquid Poison pump 12 and CRD pump 12 are operating. Plausible because they do de-energize during the transient before the EDGs re-energize their buses and load shedding does occur.

Technical Reference(s): N1-OP-30, C-19409-C Sheet 1b

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-262001-RBO-11

Question Source: Bank – 2017 Cert #25

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| | | |
|---|---|---|
| Examination Outline Cross-Reference: | Level | RO |
| | Tier # | 2 |
| | Group # | 1 |
| | K/A # | 262001 A4.01 |
| | Importance Rating | 3.7 |

**AC ELECTRICAL DISTRIBUTION**

**Ability to manually operate and/or monitor the AC ELECTRICAL DISTRIBUTION in the control room: Breakers and disconnects**

Proposed Question: #47

The plant is operating at 100% power when a Main Turbine trip occurs.

Which one of the following describes the resulting status of the Main Generator output breakers, R915 and R925, and disconnect MOD-18 three (3) minutes later?

| | Main Generator Output Breakers R915 and R925 | Disconnect MOD-18 |
|---|---|---|
| A. | Open | Open |
| B. | Open | Closed |
| C. | Closed | Open |
| D. | Closed | Closed |

Proposed Answer: A

Explanation: Both Main Generator output breakers automatically open on the Main Turbine trip. Both output breakers remain open until manual action is taken to reclose them. Main Generator disconnect MOD-18 also automatically opens on the Main Turbine trip and remains open until manual action is taken.

B. Incorrect – MOD-18 automatically opens and remains open. Plausible that it would remain closed since R915 and R925 being open separates the Main Generator from the grid.
C. Incorrect – R915 and R925 automatically open and remain open. Plausible that they would reclose to re-establish the preferred 345 KV grid alignment. Also plausible because N1-OP-31 contains direction to manually reclose these breakers to re-establish the preferred 345 KV grid alignment.
D. Incorrect – R915, R925, and MOD-18 automatically open and remain open. Plausible because the initiating signal is a Main Turbine trip, not a Main Generator lockout. Also plausible because it is desired to realign the system to re-establish the preferred 345 KV grid alignment.

Technical Reference(s): N1-SOP-31.1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-262001

Question Source: Bank - JAF 18-1 NRC #46

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(4)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | | |
|---|---|---|
| | Level | RO |
| | Tier # | 2 |
| | Group # | 1 |
| | K/A # | 262002 K4.01 |
| | Importance Rating | 3.5 |

**UNINTERRUPTABLE POWER SUPPLY (AC/DC)**

**Knowledge of UNINTERRUPTABLE POWER SUPPLY (AC/DC) design features and/or interlocks that provide for the following: Transfer of power supplies**

Proposed Question: #48

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

- Lines 1 and 4 de-energize.
- EDGs 102 and 103 re-energize their respective powerboards.

Which one of the following describes the response of UPS 172 to this electrical transient?

UPS 172…

A. loads are automatically transferred to the bypass transformer. UPS 172 loads remain on this source until manually re-transferred.

B. loads are automatically transferred to the bypass transformer. UPS 172 loads automatically transfer back after power restoration.

C. inverter is automatically supplied power from Battery Board 12. UPS 172 inverter remains supplied from this source until manually re-transferred.

D. inverter is automatically supplied power from Battery Board 12. UPS 172 inverter power supply automatically transfers back after power restoration.

Proposed Answer: D

Explanation: Loss of Lines 1 and 4 causes Powerboard 17B to de-energize. Powerboard 17B is the normal power supply for UPS 172, and also the power supply for the UPS 172 bypass transformer. With the loss of the normal AC input to UPS 172, Battery Board 12 automatically supplies power to the inverter. When normal AC power is restored (automatically after EDG start and synchronization), the UPS 172 inverter automatically swaps back to the normal power source.

A. Incorrect – UPS 172 inverter automatically receives power from Battery Board 12. Plausible because the bypass transformer is another alternate power supply that will power loads in some situations. UPS 172 automatically transfers back to normal power supply. Plausible that the transfer back would take manual action to ensure the AC source is stable before the transfer.
B. Incorrect – UPS 172 inverter automatically receives power from Battery Board 12. Plausible because the bypass transformer is another alternate power supply that will power loads in some situations.
C. Incorrect – UPS 172 automatically transfers back to normal power supply. Plausible that the transfer back would take manual action to ensure the AC source is stable before the transfer.

Technical Reference(s): N1-OP-40

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-262002-RBO-5

Question Source: Bank – 2013 NRC #47

Question History: 2013 NRC #47

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

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

**DC ELECTRICAL DISTRIBUTION**

**Knowledge of DC ELECTRICAL DISTRIBUTION design features and/or interlocks that provide for the following: Ground detection**

Proposed Question: #49

Which one of the following describes the ground detection for Battery 14?

A. Voltmeter located on Battery Board 14

B. Voltmeter located in the Main Control Room

C. Indicating lights located on Battery Board 14

D. Indicating lights located in the Main Control Room

Proposed Answer: A

Explanation: The ground detector for Battery 14 is a voltmeter located on Battery Board 14.

B. Incorrect – There is no ground detection indication for Battery 14 in the Main Control Room. Plausible because there is for Batteries 11 and 12.
C. Incorrect – The indication is a voltmeter. Plausible because this is another style of ground detector and there are many other plant indications that use indicating lights.
D. Incorrect – The indication is a voltmeter. Plausible because this is another style of ground detector and there are many other plant indications that use indicating lights. There is no ground detection indication for Battery 14 in the Main Control Room. Plausible because there is for Batteries 11 and 12.

Technical Reference(s): ARP A3-2-2, N1-OP-47A, N1-263000-RBO-5

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-263000-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| | | |
|---|---|---|
| Examination Outline Cross-Reference: | Level | RO |
| | Tier # | 2 |
| | Group # | 1 |
| | K/A # | 264000 A4.07 |
| | Importance Rating | 3.4 |

**EMERGENCY GENERATORS (DIESEL/JET)**

**Ability to manually operate and/or monitor the EMERGENCY GENERATORS (DIESEL/JET) in the control room: Transfer of control to local/remote panels**

Proposed Question: #50

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

- Annunciator A4-2-6, DSL GEN 102-103 ISO. SW. LOCAL R1012-R1013 SW. LOCKOUT, alarms.
- Investigation reveals that the EDG 102 selector switch is in LOCAL START at the EDG 102 engine control panel.

While this switch is in this position, which one of the following describes:

(1) the ability to manually start and load EDG 102 in the Main Control Room

and

(2) whether EDG 102 will start and load on a valid undervoltage signal?

| | (1) Can EDG 102 be manually started and loaded in the Main Control Room? | (2) Will EDG 102 automatically start and load on a valid undervoltage signal? |
|---|---|---|
| A. | Yes | Yes |
| B. | Yes | No |
| C. | No | Yes |
| D. | No | No |

Proposed Answer: D

Explanation: Placing the EDG 102 selector switch in LOCAL START at the EDG 102 engine control panel disables the ability to start EDG 102 in the Main Control Room. This switch position also disables field flashing for EDG 102. This prevents the EDG from being able to be loaded in any condition.

A. Incorrect – EDG 102 cannot be manually started and loaded from the Main Control Room with this switch position. Plausible that the switch would only serve to enable to local start pushbutton, but would not disable remote starting.
B. Incorrect – EDG 102 cannot be manually started and loaded from the Main Control Room with this switch position. Plausible that the switch would only serve to enable to local start pushbutton, but would not disable remote starting.
C. Incorrect – EDG 102 will not start and load on a valid undervoltage signal with this switch position. Plausible that the switch would only affect local/remote manual starting/loading of the EDG and not automatic starting/loading.

Technical Reference(s): ARP A4-2-6, N1-OP-45, C-19410-C sheets 3 and 10

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-245001-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| | | |
|---|---|---|
| Examination Outline Cross-Reference: | Level | RO |
| | Tier # | 2 |
| | Group # | 1 |
| | K/A # | 300000 291001 K1.04 |
| | Importance Rating | 2.8 |

**INSTRUMENT AIR SYSTEM**

**Valves**

**The failed-valve positions for different operators (open, closed, and as-is positions; spring-loaded valves; hydraulically/pneumatically controlled valves; electric motor-driven valves)**

Proposed Question: #51

The Reactor is scrammed with the following:

- Emergency Condenser (EC) Loop 11 is being used for pressure control.
- EC Loop 12 is unavailable.
- Instrument air pressure is 0 psig.

Which one of the following describes the response of 60-17, EC Shell Side Makeup LCV 11, and 60-04, EC Makeup Tank Makeup LCV 11?

| | 60-17, EC Shell Side Makeup LCV 11 | 60-04, EC Makeup Tank Makeup LCV 11 |
|---|---|---|
| A. | Fails open | Fails open |
| B. | Fails open | Fails closed |
| C. | Fails closed | Fails open |
| D. | Fails closed | Fails closed |

Proposed Answer: B

Explanation: 60-17 fails open on loss of IA. 60-04 fails closed due to loss of IA.

A. Incorrect – 60-04 fails closed due to loss of IA. Plausible that this valve would fail open such that makeup to the Makeup Tank was not lost.
C. Incorrect – 60-17 fails closed due to loss of IA. Plausible that this valve would fail closed such that the shell side would not be overfilled. 60-04 fails closed due to loss of IA. Plausible that this valve would fail open such that makeup to the Makeup Tank was not lost.
D. Incorrect – 60-17 fails closed due to loss of IA. Plausible that this valve would fail closed such that the shell side would not be overfilled.

Technical Reference(s): C-18017-C, N1-SOP-20.1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-278000-RBO-8

Question Source: Bank – 2017 NRC #6

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | | |
|---|---|---|
| | Level | RO |
| | Tier # | 2 |
| | Group # | 1 |
| | K/A # | 400000 A2.08 |
| | Importance Rating | 3.5 |

**COMPONENT COOLING WATER SYSTEM**

**Ability to (a) predict the impacts of the following on the COMPONENT COOLING WATER SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Residual heat removal system heat exchanger tube leak**

Proposed Question: #52

A plant shutdown is in progress with the following:

- Reactor pressure is 100 psig.
- Reactor coolant temperature is 337°F.
- Shutdown Cooling (SDC) loop 11 is in service.

Then, a tube leak develops in the SDC 11 heat exchanger.

Which one of the following describes:

- the resulting flow of leakage

  and

- the ability to isolate this tube leak, in accordance with N1-OP-4, Shutdown Cooling System?

| | Leakage is from… | Leak Isolation |
|---|---|---|
| A. | SDC into RBCLC | All three SDC loops must be isolated |
| B. | SDC into RBCLC | SDC loop 11 can be isolated without isolating SDC loops 12 and 13 |
| C. | RBCLC into SDC | All three SDC loops must be isolated |
| D. | RBCLC into SDC | SDC loop 11 can be isolated without isolating SDC loops 12 and 13 |

Proposed Answer: B

Explanation: The SDC heat exchangers are cooled by RBCLC. Normal supply pressure from RBCLC pumps is 80-90 psig. With Reactor pressure at 100 psig, SDC system pressure will be above RBCLC supply pressure. Therefore, leakage will be from SDC into RBCLC. This will cause rising temperature in RBCLC, rising rad levels in RBCLC, and discharge from the RBCLC overflow line near the CLC makeup tank. N1-OP-4 section H.3 provides guidance for isolating a SDC heat exchanger tube leak. This procedure section allows isolation of just SDC loop 11 without isolating SDC loops 12 and 13.

**Note:** The equivalent to an RHR heat exchanger tube leak at NMP Unit 1 is a SDC heat exchanger tube leak.

A. Incorrect – SDC loop 11 can be isolated without isolating SDC loops 12 and 13 per OP-4 section H.3. Plausible because use of certain isolation valves would result in isolation of all three SDC loops (38-01, 38-02, 38-13, 70-53). Also plausible because numerous conditions do isolate SDC with these valves (low Reactor water level, high SDC area temperature, manual Vessel isolation using E-console switches).
C. Incorrect – Under the given conditions, leakage is from SDC into RBCLC. Plausible because, for some systems, leakage would be in the other direction. Also plausible because leakage may be in the other direction if Reactor pressure was lower. SDC loop 11 can be isolated without isolating SDC loops 12 and 13 per OP-4 section H.3. Plausible because use of certain isolation valves would result in isolation of all three SDC loops (38-01, 38-02, 38-13, 70-53). Also plausible because numerous conditions do isolate SDC with these valves (low Reactor water level, high SDC area temperature, manual Vessel isolation using E-console switches).
D. Incorrect – Under the given conditions, leakage is from SDC into RBCLC. Plausible because, for some systems, leakage would be in the other direction. Also plausible because leakage may be in the other direction if Reactor pressure was lower.

Technical Reference(s): ARP K3-2-1, N1-OP-4 section H.3

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-208000-RBO-8

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(3)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | | |
|---|---|---|
| | Level | RO |
| | Tier # | 2 |
| | Group # | 2 |
| | K/A # | 202001 K2.01 |
| | Importance Rating | 3.5 |

**RECIRCULATION SYSTEM**

**Knowledge of electrical power supplies to the following: Recirculation pumps**

Proposed Question: #53

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

- Reactor Recirculation pump (RRP) 15 is shut down.
- The other four RRPs are operating.

Then, Powerboard 11 de-energizes due to a sustained electrical fault.

Which one of the following identifies the resulting status of RRPs?

| | RRP 11 | RRP 12 | RRP 13 | RRP 14 |
|---|---|---|---|---|
| A. | Running | Running | Running | NOT running |
| B. | Running | Running | NOT running | Running |
| C. | NOT running | Running | NOT running | Running |
| D. | NOT running | NOT running | Running | Running |

Proposed Answer: D

Explanation: RRPs 11 and 12 are powered from Powerboard 11. RRPs 14 and 15 are powered from Powerboard 12. RRP 13 is powered from Powerboard 101. The loss of Powerboard 11 results in RRPs 11 and 12 NOT running, RRPs 13 and 14 running.

A. Incorrect – RRPs 11 and 12 are NOT running and RRP 14 is running. Plausible because this would be correct for a loss of Powerboard 12.
B. Incorrect – RRPs 11 and 12 are NOT running and RRP 13 is running. Plausible because this would be correct for a loss of Powerboard 101.
C. Incorrect – RRP 12 is NOT running and RRP 13 is running. Plausible because this would be correct if the power supply alignment was different (RRPs 11 and 13 on PB 11, RRPs 12 and 14 on PB 12).

Technical Reference(s): C-19409-C sheet 1b

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-202001-RBO-4

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(3)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| | | |
|---|---|---|
| Examination Outline Cross-Reference: | Level | RO |
| | Tier # | 2 |
| | Group # | 2 |
| | K/A # | 216000 A1.05 |
| | Importance Rating | 3.3 |

**NUCLEAR BOILER INSTRUMENTATION**

**Ability to predict and/or monitor changes in parameters associated with operation of the NUCLEAR BOILER INSTRUMENTATION including: Lights and alarms**

Proposed Question: #54

The plant is experiencing a transient with the following:

- Annunciator F1-2-8, CORE LEVEL TORUS TEMP. MONITOR SYS. 11 TROUBLE, alarms.
- Annunciator F4-2-1, CORE LEVEL TORUS TEMP. MONITOR SYS. 12 TROUBLE, alarms.
- The Fuel Zone channel 11 and 12 displays begin flashing.

Which one of the following describes what is indicated by these conditions?

A. Reactor water level is below the usable range of Fuel Zone instruments

B. Reactor water level is above the usable range of Fuel Zone instruments

C. Fuel Zone instruments are unreliable due to the status of reference leg temperature

D. Fuel Zone instruments are unreliable due to the status of Reactor Recirculation pumps

Proposed Answer: C

Explanation: The given annunciators and flashing indicators indicate that Fuel Zone instruments are unreliable due to reference leg temperatures above the saturation temperature for the current Reactor pressure (indicates potential for reference leg flashing).

A. Incorrect – The given conditions indicate Fuel Zone instruments are unreliable due to reference leg temperature. Plausible because being below usable range would also be of concern and is called out in EOP Detail A.
B. Incorrect – The given conditions indicate Fuel Zone instruments are unreliable due to reference leg temperature. Plausible because being above usable range would also be of concern and is called out in EOP Detail A.
D. Incorrect – The given conditions indicate Fuel Zone instruments are unreliable due to reference leg temperature. Plausible because status of Reactor Recirculation pumps does affect Fuel Zone indicators and is called out in EOP Detail A.

Technical Reference(s): ARPs F1-2-8 and F4-2-1, N1-216000-RBO-5, EOP Detail A

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-216000-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | Level | RO |
|---|---|---|
| | Tier # | 2 |
| | Group # | 2 |
| | K/A # | 223001 K1.16 |
| | Importance Rating | 3.5 |

**PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES**

**Knowledge of the physical connections and/or cause and effect relationships between the PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES and the following systems: Containment and drywell atmosphere monitoring system**

Proposed Question: #55

A loss of coolant accident has resulted in the following:

- Drywell pressure is 12 psig and rising slowly.
- Reactor water level is 72" and stable.

Which one of the following describes the availability of the $H_2O_2$ monitors?

The $H_2O_2$ monitors are…

A. available and lined up.

B. isolated. Bypass the isolation using the CAD CHANNEL 11 and CAD CHANNEL 12 RPS BY-PASS switches.

C. isolated. Bypass the isolation using the AUTO VESSEL ISOL CH 11 and AUTO VESSEL ISOL CH 12 BYPASS switches.

D. isolated and unavailable to be placed in service under these conditions.

Proposed Answer: B

Explanation: The $H_2O_2$ monitors isolate with Drywell pressure above 3.5 psig. Per N1-OP-9 and N1-EOP-PC, to recover the $H_2O_2$ monitors it is necessary to un-isolate the system by bypassing the Containment Isolation using CAD CHANNEL 11 and 12 RPS BY-PASS switches.

A. Incorrect – The $H_2O_2$ monitors isolate with Drywell pressure above 3.5 psig. Plausible that they would be in-service under these conditions because their purpose is to give indications under accident conditions. Also plausible because Reactor water level is in the normal band.
C. Incorrect – AUTO VESSEL ISOL CH 11 and CH 12 BYPASS switches do not need to be manipulated to restore $H_2O_2$ monitors. Plausible because these switches are used to bypass other isolations.
D. Incorrect – The $H_2O_2$ monitors can be restored under these conditions. Plausible because they do have an active isolation signal present.

Technical Reference(s): N1-EOP-PC, N1-OP-9, N1-SOP-40.2

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-223002-RBO-5

Question Source: Bank – 2013 NRC #27

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(9)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | Level | RO |
|---|---|---|
| | Tier # | 2 |
| | Group # | 2 |
| | K/A # | 226001 A3.07 |
| | Importance Rating | 3.9 |

**RHR/LPCI: CONTAINMENT SPRAY MODE SYSTEM MODE**

**Ability to monitor automatic operation of the RHR/LPCI: CONTAINMENT SPRAY MODE SYSTEM MODE including: Pump start**

Proposed Question: #56

A loss of coolant accident occurs with the following:

- Drywell pressure is 4 psig and rising slowly.
- Reactor water level is 0" and stable.
- Reactor pressure is 920 psig and stable.
- NO actions are taken in the EOPs.

Which one of the following describes the status of Containment Spray one (1) minute later?

Containment Spray pumps are…

A. in standby.

B. running but are NOT spraying the Containment.

C. spraying the Containment. Containment Spray Raw Water flow is being supplied to the heat exchangers.

D. spraying the Containment. Containment Spray Raw Water flow is NOT being supplied to the heat exchangers.

Proposed Answer: D

Explanation: The Containment Spray pumps automatically start due to the combination of Drywell pressure >3.5 psig and Reactor water level <+5". The Containment Spray system valves are aligned such that this results in Containment Spray spraying the Containment with no further action. The Containment Spray Raw Water pumps do not automatically start. Further manual action is required to initiate Containment Spray Raw Water flow to the heat exchangers.

A. Incorrect – Containment Spray pumps automatically start. Plausible because this would be correct if Reactor water level was above +5". Also plausible because Reactor water level is above the ADS setpoint of -10".
B. Incorrect – Containment Spray is spraying the Containment. Plausible that further degradation of conditions would be required to initiate spray flow. Also plausible because this analogous of the current status of Core Spray pumps (running on min flow).
C. Incorrect – Containment Spray Raw Water pumps do not automatically start. Plausible that these pumps would start on the same signal as the Containment Spray pumps to ensure heat removal.

Technical Reference(s): N1-OP-14

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-226001-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(9)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | | |
|---|---|---|
| | Level | RO |
| | Tier # | 2 |
| | Group # | 2 |
| | K/A # | 233000 K5.03 |
| | Importance Rating | 3.3 |

**FUEL POOL COOLING/CLEANUP**

**Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the FUEL POOL COOLING/CLEANUP: Spent fuel decay heat generation**

Proposed Question: #57

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

- Annunciator L1-4-5, FUEL POOL ANNUNCIATOR, alarms.
- An Operator in the field reports annunciator SFP-1-2, FUEL POOL PUMPS INLET TEMP, is in alarm.
- The Operator also reports Spent Fuel Pool temperature is 102°F and rising slowly.

Which one of the following describes:

- the most restrictive Spent Fuel Pool temperature limitation that must be observed, in accordance with N1-OP-6, Fuel Pool Filtering and Cooling System,

  and

- the need to enter N1-EOP-SC, Secondary Containment Control?

| | Most Restrictive Spent Fuel Pool Temperature Limitation | N1-EOP-SC Entry |
|---|---|---|
| A. | 110°F | Required |
| B. | 110°F | NOT required |
| C. | 140°F | Required |
| D. | 140°F | NOT required |

Proposed Answer: A

Explanation: During power operation, Spent Fuel Pool temperature must be maintained less than 110ºF. N2-EOP-SC entry is required because Spent Fuel Pool temperature is above 100ºF.

B. Incorrect – N2-EOP-SC entry is required because Spent Fuel Pool temperature is above 100ºF. Plausible because this would be correct if Spent Fuel Pool temperature was slightly lower.
C. Incorrect – During power operation, Spent Fuel Pool temperature must be maintained less than 110ºF. Plausible because 140ºF is the limit during an outage with the Spent Fuel Pool in communication with the Reactor coolant.
D. Incorrect – During power operation, Spent Fuel Pool temperature must be maintained less than 110ºF. Plausible because 140ºF is the limit during an outage with the Spent Fuel Pool in communication with the Reactor coolant. N2-EOP-SC entry is required because Spent Fuel Pool temperature is above 100ºF. Plausible because this would be correct if Spent Fuel Pool temperature was slightly lower.

Technical Reference(s): N1-OP-6, N1-EOP-SC

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-233000-RBO-10

Question Source: Modified Bank - 2013 NRC #93

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference: Level SRO
Tier # 2
Group # 2
K/A # 233000 2.4.31
Importance Rating 4.1

**Fuel Pool Cooling/Cleanup**

**Knowledge of annunciator alarms, indications, or response procedures.**

Proposed Question: #92

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

- Annunciator L1-4-5, FUEL POOL ANNUNCIATOR, alarms.
- An Operator in the field reports annunciator SFP-1-2, FUEL POOL PUMPS INLET TEMP, is in alarm.
- The Operator also reports Spent Fuel Pool temperature is 100°F and rising slowly.

Which one of the following describes the most restrictive Spent Fuel Pool temperature limitation that must be observed to avoid violation of license condition?

Spent Fuel Pool temperature must be maintained less than or equal to…

A. 110°F based on Secondary Containment licensing basis.

B. 110°F based on Spent Fuel Pool licensing basis.

C. 140°F based on Secondary Containment licensing basis.

D. 140°F based on Spent Fuel Pool licensing basis.

Proposed Answer: A

Explanation: During power operation, Spent Fuel Pool temperature must be maintained less than 110°F per the licensing basis for the Secondary Containment.

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | | |
|---|---|---|
| | Level | RO |
| | Tier # | 2 |
| | Group # | 2 |
| | K/A # | 239001 K6.08 |
| | Importance Rating | 3.5 |

**MAIN AND REHEAT STEAM SYSTEM**

**Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the MAIN AND REHEAT STEAM SYSTEM: Main condenser vacuum**

Proposed Question: #58

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

- A loss of Main Condenser vacuum occurs.
- The Reactor is manually scrammed.
- The Reactor Mode Switch is placed in SHUTDOWN.
- Reactor water level reaches a low of 12” before recovering to the normal band.
- Main Condenser vacuum is 6” Hg.

Which one of the following describes the resulting status of MSIVs?

MSIVs…

A. remain open

B. are closed due to a low Reactor water level signal, only

C. are closed due to a low Main Condenser vacuum signal, only

D. are closed due to both low Reactor water level and Main Condenser vacuum signals

Proposed Answer: C

Explanation: The MSIVs close due to Main Condenser vacuum <7" Hg. The MSIVs do not receive an isolation signal on Reactor water level because Reactor water level remained above the 5" isolation setpoint.

A. Incorrect – The MSIVs are closed due to Main Condenser vacuum <7" Hg. Plausible because this would be correct if Main Condenser vacuum was slightly higher.
B. Incorrect – The MSIVs do not receive an isolation signal on Reactor water level because Reactor water level remained above the 5" isolation setpoint. Plausible because this would be correct if Reactor water went slightly lower and Main Condenser vacuum was slightly higher.
D. Incorrect – The MSIVs do not receive an isolation signal on Reactor water level because Reactor water level remained above the 5" isolation setpoint. Plausible because this would be correct if Reactor water went slightly lower.

Technical Reference(s): N1-SOP-40.2

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-239001-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(5)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | | |
|---|---|---|
| | Level | RO |
| | Tier # | 2 |
| | Group # | 2 |
| | K/A # | 245000 A2.02 |
| | Importance Rating | 3.8 |

**MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS**

**Ability to (a) predict the impacts of the following on the MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Loss of lube oil**

Proposed Question: #59

The plant is operating at 100% power when a large leak causes a complete loss of Main Turbine lube oil pressure.

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

The Main Turbine…

A. automatically trips. Reactor pressure can be controlled using the Bypass Opening Jack.

B. automatically trips. Reactor pressure CANNOT be controlled using the Bypass Opening Jack.

C. does NOT automatically trip. The Main Turbine can be manually tripped from the Main Control Room.

D. does NOT automatically trip. The Main Turbine CANNOT be manually tripped from the Main Control Room.

Proposed Answer: B

Explanation: The Main Turbine lube oil system supplies control oil for the Main Turbine Control system (MHC). With a loss of all lube oil pressure, the Main Turbine automatically trips due to the loss of MHC control oil pressure. This loss of control oil pressure will also result in loss of automatic Reactor pressure control on Turbine Bypass Valves. The Bypass Opening Jack also requires control oil pressure to open the Turbine Bypass Valves, so this method of Reactor pressure control is not available to the operators.

A. Incorrect – Reactor pressure CANNOT be controlled using the Bypass Opening Jack. Plausible that the Bypass Opening Jack would mechanically jack open Turbine Bypass Valves and not require control oil pressure.
C. Incorrect – The Main Turbine automatically trips. Plausible that the loss of control oil pressure would result in a loss of trip function rather than an automatic trip.
D. Incorrect – The Main Turbine automatically trips. Plausible that the loss of control oil pressure would result in a loss of trip function rather than an automatic trip. Also plausible that this would disable Control Room turbine trip pushbuttons and require tripping the Main Turbine from the front standard.

Technical Reference(s): N1-OP-31, N1-247000-RBO-11

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-247000-RBO-11

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | Level | RO |
|---|---|---|
| | Tier # | 2 |
| | Group # | 2 |
| | K/A # | 259001 K4.06 |
| | Importance Rating | 3.0 |

**FEEDWATER SYSTEM**

**Knowledge of FEEDWATER SYSTEM design features and/or interlocks that provide for the following: RFP lubrication**

Proposed Question: #60

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

- It is discovered that the Feedwater pump 11 Auxiliary Oil pump will NOT start.
- Feedwater pump 11 is in standby.

Then, Reactor water level lowers to 40".

Which one of the following describes the resulting status of Feedwater pump 11?

Feedwater pump 11…

A. automatically starts, but then trips

B. automatically starts and runs normally

C. does NOT automatically start, but can be manually started

D. does NOT automatically start and CANNOT be manually started

Proposed Answer: D

Explanation: The Feedwater pump 11 start circuitry includes an interlock that requires sufficient oil pressure before the pump will attempt a start, either from an automatic or manual start signal. The Auxiliary Oil pump must start for the oil pressure permissive to be met. Therefore, with the Auxiliary Oil pump unable to start, Feedwater pump 11 will not automatically start and cannot be manually started.

A. Incorrect – Feedwater pump 11 will not automatically start. Plausible that the pump would start normally, but then trip later on low oil pressure or a related signal.
B. Incorrect – Feedwater pump 11 will not automatically start. Plausible because the Feedwater pump also has a main oil pump that is driven by the Feedwater pump and supplies all required oil pressure once the Feedwater pump starts.
C. Incorrect – Feedwater pump 11 cannot be manually started. Plausible that the oil pressure permissive could be overridden by manual action.

Technical Reference(s): N1-OP-16

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-259001-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | | |
|---|---|---|
| | Level | RO |
| | Tier # | 2 |
| | Group # | 2 |
| | K/A # | 268000 2.2.43 |
| | Importance Rating | 3.0 |

**RADWASTE SYSTEM**

**Knowledge of the process used to track inoperable alarms**

Proposed Question: #61

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

- Annunciator H2-2-1, R BLDG FL DR SUMPS 11-16 AREA WTR LVL LEVEL HIGH, is alarming every few minutes due to a faulty input.
- It is desired to disable the faulty annunciator input per S-ARP-001, Disabling Nuisance Annunciators.
- It is determined that a lead must be lifted to disable the annunciator.

Which one of the following describes the proper tag to be used on this annunciator, in accordance with in accordance with S-ARP-001?

A. Caution Tag

B. Information Tag

C. Special Condition Tag

D. Temporary Configuration Change Tag

Proposed Answer: D

Explanation: S-ARP-001 provides guidance to eliminate the source of a nuisance alarm. The process is carried out by OP-AA-108-101 and utilizes a Temporary Configuration Change Tag for the annunciator.

A. Incorrect – A Temporary Configuration Change Tag is used for the annunciator. Plausible because Caution Tags are commonly used on the Control Room boards for off-normal controls or indications.
B. Incorrect – A Temporary Configuration Change Tag is used for the annunciator. Plausible because Information Tags are commonly used on the Control Room boards for off-normal controls or indications.
C. Incorrect – A Temporary Configuration Change Tag is used for the annunciator. Plausible because a Special Condition Tag is another type of tag used at the facility for other processes.

Technical Reference(s): S-ARP-001

Proposed references to be provided to applicants during examination: None

Learning Objective: NS-OJTTPE-SRO-OPAA10

Question Source: Bank – JAF 17-2 NRC #42

Question History: JAF 17-2 NRC #42

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| | | |
|---|---|---|
| Examination Outline Cross-Reference: | Level | RO |
| | Tier # | 2 |
| | Group # | 2 |
| | K/A # | 271000 A4.09 |
| | Importance Rating | 3.0 |

**OFFGAS SYSTEM**

**Ability to manually operate and/or monitor the OFFGAS SYSTEM in the control room: Offgas system controls / components**

Proposed Question: #62

The plant is operating at 100% power when a fuel defect results in the following annunciators:

- H1-1-7, OFF GAS HIGH RADIATION
- H1-2-7, OFF GAS RAD MON 11-12 FILTER DP SAMPLE FLOW
- H1-3-7, MN CONDR OFF GAS TIMER STARTED

The Off Gas rad monitors are confirmed to have tripped on high-high level.

Which one of the following describes the resulting status of:

- 76-12 and 76-13, Interstage Blocking Valves

  and

- 77-03, Stack Blocking Valve,

one (1) minute later?

| | 76-12 and 76-13, Interstage Blocking Valves | 77-03, Stack Blocking Valve |
|---|---|---|
| A. | Open | Open |
| B. | Open | Closed |
| C. | Closed | Open |
| D. | Closed | Closed |

Proposed Answer: B

Explanation: High-high Offgas radiation causes 77-03 to close immediately. The associated Offgas timers are set to 0, so no time delay occurs. 76-12 and 76-13 are unaffected by these conditions and remain open.

A. Incorrect – 77-03 is closed. Plausible that the timer would be set such that the valve would still be open after one minute.
C. Incorrect – 76-12 and 76-13 are unaffected by these conditions and remain open. Plausible because they do automatically close on high pressure or temperature. Also plausible because they also block the flow of gases from the Main Condenser to the Stack. 77-03 is closed. Plausible that the timer would be set such that the valve would still be open after one minute.
D. Incorrect – 76-12 and 76-13 are unaffected by these conditions and remain open. Plausible because they do automatically close on high pressure or temperature. Also plausible because they also block the flow of gases from the Main Condenser to the Stack.

Technical Reference(s): ARP H1-3-7, N1-OP-25

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-271000-RBO-5

Question Source: Modified Bank – 2013 NRC #62

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments:

| ES-401 | Written Examination Question Worksheet | Form ES-401-5 |
|---|---|---|

Examination Outline Cross-Reference: Level RO
Tier # 2
Group # 2
K/A # 271000 K6.09
Importance Rating 3.4

**Offgas**

**Knowledge of the effect that a loss or malfunction of the following will have on the OFFGAS SYSTEM: Fuel cladding integrity**

Proposed Question: #62

The plant is operating at 100% power when a fuel defect results in the following annunciators:

- H1-1-7, OFF GAS HIGH RADIATION
- H1-2-7, OFF GAS RAD MON 11-12 FILTER DP SAMPLE FLOW
- H1-3-7, MN CONDR OFF GAS TIMER STARTED

The Off Gas rad monitors are confirmed to have tripped on high-high level.

Which one of the following describes the resulting automatic action(s) and the required manual action?

| | Automatic Action(s) | Required Manual Action |
|---|---|---|
| A. | Stack Blocking Valve, 77-03, closes. | Immediately scram the reactor due to rapidly lowering Main Condenser vacuum. |
| B. | Interstage Blocking Valves, 76-12 and 76-13, close. | Immediately scram the reactor due to rapidly lowering Main Condenser vacuum. |
| C. | Stack Blocking Valve, 77-03, closes. | Perform an emergency power reduction to attempt to control radiation levels and stabilize Main Condenser vacuum. |
| D. | Interstage Blocking Valves, 76-12 and 76-13, close. | Perform an emergency power reduction to attempt to control radiation levels and stabilize Main Condenser vacuum. |

Proposed Answer: C

Explanation: BV 77-03 closes on high-high offgas radiation. The vacuum loss following closure of 77-03 is not rapid. Sufficient volume exists in the offgas system piping to continue to remove non-condensable gases from the main condenser for a period of time, during which vacuum will slowly degrade. N1-SOP-25.1 and 25.2 direct an emergency power reduction to control radiation/vacuum.

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | | |
|---|---|---|
| | Level | RO |
| | Tier # | 2 |
| | Group # | 2 |
| | K/A # | 290003 K5.04 |
| | Importance Rating | 3.7 |

**CONTROL ROOM VENTILATION**

**Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the CONTROL ROOM VENTILATION: Control room habitability**

Proposed Question: #63

Which one of the following identifies when manual initiation of Control Room Emergency Ventilation System (CREVS) is required, in accordance with N1-OP-49, Control Room Ventilation System?

A. High Drywell pressure or Low Low Reactor Water level at Unit 1

B. High Drywell pressure or Low Low Reactor Water level at Unit 2

C. High total Iodine concentration at Unit 1, or Unit 2 enters N2-EOP-RR, Radioactivity Release Control

D. High total Iodine concentration at Unit 2, or Unit 1 enters N1-EOP-RR, Radioactivity Release Control

Proposed Answer: C

Explanation: N1-OP-49 requires manual initiation of CREVS if either of the following occur:

- Total iodine radioactivity concentration greater than 9.47 µCi/gm (T.S. 3.2.4)
- Unit 2 enters RR (Radioactivity Release) EOP

**Note:** The question meets the K/A because an operational implication of Control Room habitability is knowing when CREVS is required to be manually initiated.

A. Incorrect – These conditions cause automatic initiation of CREVS, therefore manual initiation is not required. Plausible because these conditions require automatic initiation.
B. Incorrect – LOCA signals at Unit 2 do not require manual initiation of CREVS. Plausible because these signals at Unit 1 cause automatic initiation and there are conditions at Unit 2 that required manual initiation of CREVS at Unit 1.
D. Incorrect – The applicable Units are backwards. Plausible because this is correct if the Units are reversed.

Technical Reference(s): N1-OP-49

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-288003-RBO-10

Question Source: Bank – 2017 Cert #71

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | Level | RO |
|---|---|---|
| | Tier # | 3 |
| | Group # | |
| | K/A # | 2.1.21 |
| | Importance Rating | 3.5 |

**Ability to verify that a copy of a controlled procedure is the proper revision**

Proposed Question: #64

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

- A working copy of an 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 an electronic database that may be used to verify that the working copy of the procedure is the currently approved revision, in accordance with HU-AA-104-101, Procedure Use and Adherence?

A. FCMS

B. eSOMs

C. Maximo

D. PassPort

Proposed Answer: A

Explanation: HU-AA-104-101 requires using the Controlled Document Register for verifying current procedure revision. RM-AA-102 lists the Controlled Document Register as FCMS.

B. Incorrect – FCMS is the correct electronic database for this task. Plausible because this is another electronic database in use at the plant that is relevant to Operations and has numerous functions.
C. Incorrect – FCMS is the correct electronic database for this task. Plausible because this is another electronic database in use at the plant that is relevant to Operations and has numerous functions.
D. Incorrect – FCMS is the correct electronic database for this task. Plausible because this is another electronic database in use at the plant that is relevant to Operations and has numerous functions. Also plausible because PassPort is the Controlled Document Register at some Constellation plants.

Technical Reference(s): HU-AA-104-101, RM-AA-102

Proposed references to be provided to applicants during examination: None

Learning Objective: N-AN-OP-HU-LP-OBJ-29

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | | |
|---|---|---|
| | Level | RO |
| | Tier # | 3 |
| | Group # | |
| | K/A # | 2.1.32 |
| | Importance Rating | 3.8 |

**Ability to explain and apply system precautions, limitations, notes, or cautions**

Proposed Question: #65

Which one of the following identifies the limitation on the timing of positive Reactivity additions, in accordance with N1-OP-43B, Normal Power Operations?

Between positive reactivity additions, a minimum wait period of…

A. one (1) minute must be observed. This limitation applies at all Reactor power levels.

B. three (3) minutes must be observed. This limitation applies at all Reactor power levels.

C. one (1) minute must be observed. This limitation applies at Reactor power levels of 98% or greater, only.

D. three (3) minutes must be observed. This limitation applies at Reactor power levels of 98% or greater, only.

Proposed Answer: D

Explanation: N1-OP-43B contains the following requirement:

“When operating at 98% of rated power or greater, positive reactivity additions whether with rods or recirculation flow will be followed by a minimum 3-minute wait period to ensure reactor conditions stabilize prior to making another positive reactivity addition.”

A. Incorrect – The minimum wait period is 3 minutes. Plausible because 1 minute is of the same magnitude. This restriction only applies at Reactor powers of 98% or greater. Plausible because positive reactivity additions must be performed in a cautious manner at all power levels.
B. Incorrect – This restriction only applies at Reactor powers of 98% or greater. Plausible because positive reactivity additions must be performed in a cautious manner at all power levels.
C. Incorrect – The minimum wait period is 3 minutes. Plausible because 1 minute is of the same magnitude.

Technical Reference(s): N1-OP-43B

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-OP43AC01-EO-05

Question Source: Bank – NMP2 11/23 NRC #65

Question History: NMP2 11/23 NRC #65

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | | |
|---|---|---|
| | Level | RO |
| | Tier # | 3 |
| | Group # | |
| | K/A # | 2.2.2 |
| | Importance Rating | 4.6 |

**Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels**

Proposed Question: #66

A Reactor startup is in progress with the following:

- The next control rod in the startup sequence is to be moved from position 00 to 48.
- Initial and current SRM readings are as follows:

| SRM | Initial Reading (cps) | Current Reading (cps) |
|---|---|---|
| 11 | 240 | 700 |
| 12 | 270 | 840 |
| 13 | 220 | 730 |
| 14 | 260 | 750 |

Which one of the following describes the requirement, if any, to use notch withdrawal for this control rod based on SRM readings, in accordance with N1-OP-43A, Plant Startup?

Notch withdrawal…

A. is required at all positions

B. is NOT required at any position

C. is required from positions 04 to 16 only

D. is required from positions 00 to 36 only

Proposed Answer: B

Explanation: N1-OP-43A P&L 2.6 requires notch control rod movement from positions 00 to 36 once 3 count rate doublings are approached until criticality is achieved. The given SRM data shows two SRMs have tripled, but no SRMs are approaching 3 count rate doublings. Therefore, there is no requirement to use notch movement for the next control rod.

A. Incorrect – There is no requirement to use notch movement for the next control rod. Plausible because if 3 count rate doublings were approached, notch withdrawal would be required, but from 00 to 36 only.
C. Incorrect – There is no requirement to use notch movement for the next control rod. Plausible because if 3 count rate doublings were approached, notch withdrawal would be required, but from 00 to 36 only. 04 to 16 are the positions identified as having the highest control rod worth per N1-OP-43A P&L 2.2.
D. Incorrect – There is no requirement to use notch movement for the next control rod. Plausible because if 3 count rate doublings were approached, notch withdrawal would be required from 00 to 36 only.

Technical Reference(s): N1-OP-43A

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-212000-RBO-10

Question Source: Bank – 2018 NRC #69

Question History: 2018 NRC #69

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | | |
|---|---|---|
| | Level | RO |
| | Tier # | 3 |
| | Group # | |
| | K/A # | 2.2.38 |
| | Importance Rating | 3.6 |

**Knowledge of conditions and limitations in the facility license**

Proposed Question: #67

The plant has experienced a transient with the following:

- Peak Reactor pressure was 1255 psig.
- Lowest Reactor water level was -30".

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

| | Reactor Pressure | Reactor Water Level |
|---|---|---|
| A. | Violated a SL | Violated a SL |
| B. | Violated a SL | Did NOT violate a SL |
| C. | Did NOT violate a SL | Violated a SL |
| D. | Did NOT violate a SL | Did NOT violate a SL |

Proposed Answer: C

Explanation: The peak Reactor pressure remained below the Safety Limit of 1375 psig. The lowest Reactor water level violated the Safety Limit of -10".

**Note:** The question meets the K/A because Technical Specifications, including Safety Limits, are part of the facility license (Appendix A).

A. Incorrect – Reactor pressure did not exceed a Safety Limit. Plausible because this would be correct if Reactor pressure rose higher to above 1375 psig. Also plausible because Reactor pressure is well above the 1080 psig limit in LSSS 2.1.2.c and slightly above the design pressure of 1250 psig.
B. Incorrect – Reactor pressure did not exceed a Safety Limit. Plausible because this would be correct if Reactor pressure rose higher to above 1375 psig. Also plausible because Reactor pressure is well above the 1080 psig limit in LSSS 2.1.2.c and slightly above the design pressure of 1250 psig. Reactor water level violated the Safety Limit of -10". Plausible because it remained well above top of active fuel (-84").
D. Incorrect – Reactor water level violated the Safety Limit of -10". Plausible because it remained well above top of active fuel (-84").

Technical Reference(s): Technical Specifications 2.1.1 and 2.2.1

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-TECSPEC01-EO-3.0

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | Level | RO |
|---|---|---|
| | Tier # | 3 |
| | Group # | |
| | K/A # | 2.3.11 |
| | Importance Rating | 3.8 |

**Ability to control radiation releases**

Proposed Question: #68

A loss of coolant accident has resulted in the following:

- Primary Containment venting is required due to approaching the Primary Containment Pressure Limit (PCPL).
- Torus water level is 12.5' and stable.

Which one of the following describes the path required to be used to vent the Containment and the reason why, in accordance with N1-EOP-PC, Primary Containment Control?

Vent from the…

A. Drywell to more quickly reduce Containment pressure

B. Torus to minimize cycling of the Torus to Drywell vacuum breakers

C. Drywell because Torus water level is too high for venting from the Torus

D. Torus to better scrub fission products from the Containment atmosphere before release

Proposed Answer: D

Explanation: N1-EOP-PC requires venting from the Torus as long as Torus water level is below 27'. The reason is to better scrub fission products from the Containment atmosphere before release.

A. Incorrect – Venting should be from the Torus. Plausible because Drywell pressure is expected to be higher than Torus pressure under LOCA conditions, which would result in a faster pressure reduction if the Drywell was vented.
B. Incorrect – The reason is not to minimize cycling of vacuum breakers. Plausible because lowering Torus pressure first would minimize cycling of vacuum breakers and cycling vacuum breakers does raise the risk of them not re-seating properly, which would be undesirable.
C. Incorrect – Venting should be from the Torus. Plausible because Torus water level is elevated above the EOP entry condition of 11.25', and if it was above 27', then Drywell venting would be required.

Technical Reference(s): N1-EOP-PC, N1-EOP-B-HOT

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOPPCC01 EO-2

Question Source: Bank - NMP2 2017 NRC #70

Question History: NMP2 2017 NRC #70

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | | |
|---|---|---|
| | Level | RO |
| | Tier # | 3 |
| | Group # | |
| | K/A # | 2.4.45 |
| | Importance Rating | 4.1 |

**Ability to prioritize and interpret the significance of each annunciator or alarm**

Proposed Question: #69

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

- An evolution is in progress.
- The operator controlling the evolution identifies an alarm that will be received after the next manipulation.

Given the following possible actions:

(1) Discuss the alarm with the Unit Supervisor before receipt of the alarm
(2) Review the associated ARP before and after receipt of the alarm
(3) Hang a flagging tool on the alarm window after receipt of the alarm

Which one of the following identifies which of these actions are requirements to classify the alarm as an “Expected Alarm”, in accordance with OP-AA-103-102, Watchstanding Practices?

A. (1) and (2) only

B. (1) and (3) only

C. (2) and (3) only

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

Proposed Answer: A

Explanation: OP-AA-103-102 classifies an “Expected Alarm” as “an alarm that occurs in the normal course of equipment operation or testing shall be considered expected if the alarm is discussed with the responsible Control Room Supervisor and the Annunciator Response Procedure/Alarm Response Card has been reviewed before receipt.”

B. Incorrect – Reviewing the ARP is required. Plausible that the ARP wouldn’t need to be review since the operator understands why it is going to be received ahead of time. Hanging a flagging tool is not required. Plausible that some visual marking would be required, such as for a disabled annunciator.
C. Incorrect – Discussing the alarm with the Unit Supervisor is required. Plausible that the Unit Supervisor would only need to be informed once the alarm is received that it is expected. Hanging a flagging tool is not required. Plausible that some visual marking would be required, such as for a disabled annunciator.
D. Incorrect – Hanging a flagging tool is not required. Plausible that some visual marking would be required, such as for a disabled annunciator.

Technical Reference(s): OP-AA-103-102

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-SOP-5.1-CE-02

Question Source: Bank – NMP2 2014 NRC #72

Question History: PB 2017 NRC #74

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

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

**Control Rods**

**State the purpose of flux shaping**

Proposed Question: #70

The plant is operating at 80% power for a control rod sequence exchange.

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

A. Equalize burn up of control rods

B. Minimize local power peaks in the core

C. Prevent conversion of uranium into plutonium in the core

D. Reduce excessive localized Reactor vessel neutron irradiation

Proposed Answer: B

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

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

Technical Reference(s): BR0SIr4_Control_Rods

Proposed references to be provided to applicants during examination: None

Learning Objective: BR0SIr4_Control_Rods 10

Question Source: Bank – NRC GFES B1557 & B1656

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(1)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| | | |
|---|---|---|
| Examination Outline Cross-Reference: | Level | RO |
| | Tier # | 4 |
| | Group # | |
| | K/A # | 292004 K1.10 |
| | Importance Rating | 3.2 |

**REACTIVITY COEFFICIENTS**

**Define the void coefficient of reactivity**

Proposed Question: #71

The plant is operating at 50% power.

Which one of the following describes the effect of the void coefficient of reactivity if Recirculation flow is ***raised***?

If Recirculation flow is **raised**, the void coefficient of reactivity will cause…

A. positive reactivity to be added due to lower moderator density

B. positive reactivity to be added due to higher moderator density

C. negative reactivity to be added due to lower moderator density

D. negative reactivity to be added due to higher moderator density

Proposed Answer: B

Explanation: As Recirculation flow is raised, the moderator spends less time in the core, resulting in boiling occurring higher in the fuel channels. This results in fewer voids in the Reactor, or higher moderator density. The increase in moderation results in addition of positive reactivity.

A. Incorrect – Moderator density rises. Plausible because the change is due to a change in moderator density and density would lower given lowering Recirculation flow.
C. Incorrect – Positive reactivity is added. Plausible because negative reactivity would be added given lowering Recirculation flow. Moderator density rises. Plausible because the change is due to a change in moderator density and density would lower given lowering Recirculation flow.
D. Incorrect – Positive reactivity is added. Plausible because negative reactivity would be added given lowering Recirculation flow.

Technical Reference(s): BR08Ir4_Operational_Physics

Proposed references to be provided to applicants during examination: None

Learning Objective: BR08Ir4_Operational_Physics 21

Question Source: Bank – Exelon GFES Bank Reactor Theory #15

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(1)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

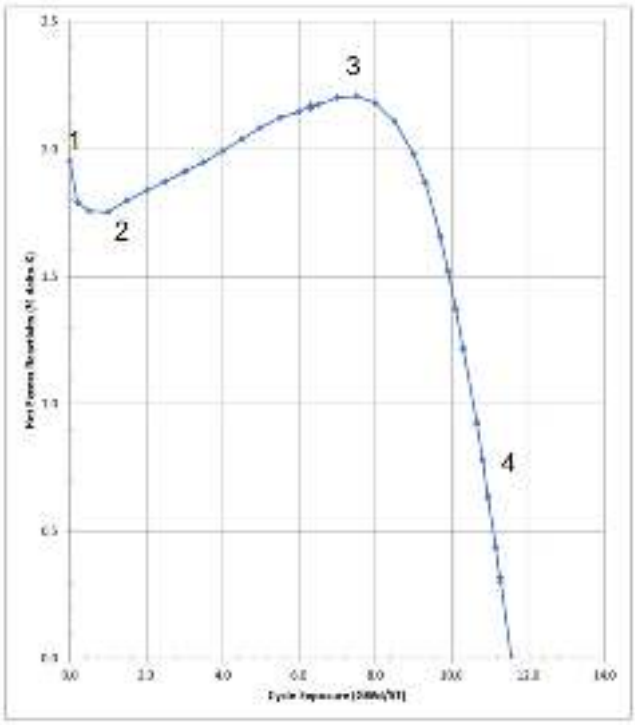
| | | |
|---|---|---|
| Examination Outline Cross-Reference: | Level | RO |
| | Tier # | 4 |
| | Group # | |
| | K/A # | 292007 K1.03 |
| | Importance Rating | 2.7 |

**FUEL DEPLETION AND BURNABLE POISONS**

**Given a curve of K-effective versus core age, state the reasons for maximum, minimum, and inflection points**

Proposed Question: #72

Given the following drawing of $K_{eff}$ versus core age from the Nine Mile Point Unit 1 Cycle Management report:




Which one of the following describes the major cause for the change in $K_{eff}$ from point 2 to point 3?

A. Depletion of fuel

B. Depletion of control rods

C. Burnout of burnable poisons

D. Burnout of fission product poisons

Proposed Answer: C

Explanation: From point 2 to 3, the major cause for the change in $K_{eff}$ is the burnout of burnable poisons. The burnable poisons deplete faster than the fuel, raising $K_{eff}$.

A. Incorrect – From point 2 to 3, the major cause for the change in $K_{eff}$ is the burnout of burnable poisons. Plausible because depletion of fuel is the major cause for the change in $K_{eff}$ from point 3 to 4.
B. Incorrect – From point 2 to 3, the major cause for the change in $K_{eff}$ is the burnout of burnable poisons. Plausible because the neutron absorbers in control rods do deplete over time.
D. Incorrect – From point 2 to 3, the major cause for the change in $K_{eff}$ is the burnout of burnable poisons. Plausible because the buildup of fission product poisons is the major cause for the change in $K_{eff}$ from point 1 to 2. Also plausible because fission product poisons do burnout in some conditions.

Technical Reference(s): BR07Ir4_Fuel_Depletion, Cycle Management Report

Proposed references to be provided to applicants during examination: Embedded graphic

Learning Objective: BR07Ir4_Fuel_Depletion Objective 3

Question Source: Bank – NMP2 Nov 2023 NRC #72

Question History: NMP2 Nov 2023 NRC #72

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(1)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | | |
|---|---|---|
| | Level | RO |
| | Tier # | 4 |
| | Group # | |
| | K/A # | 293003 K1.16 |
| | Importance Rating | 2.8 |

**STEAM**

**Define the following term: Subcooling**

Proposed Question: #73

The plant is operating at 100% power when extraction steam to a Feedwater heater isolates.

Which one of the following describes:

(1) the resulting change in Feedwater inlet temperature to the Reactor

and

(2) the amount of subcooling of the Feedwater entering the Reactor?

| | Feedwater Inlet Temperature | Amount of Subcooling |
|---|---|---|
| A. | Rises | Rises |
| B. | Rises | Lowers |
| C. | Lowers | Rises |
| D. | Lowers | Lowers |

Proposed Answer: C

Explanation: Loss of extraction steam to a Feedwater heater results in less heating of the Feedwater before it enters the Reactor. This results in a lower Feedwater inlet temperature, which equates to more subcooling.

A. Incorrect – Feedwater inlet temperature lowers. Plausible because this would be correct for restoring extraction steam to a Feedwater heater.
B. Incorrect – Feedwater inlet temperature lowers and subcooling rises. Plausible because this would be correct for restoring extraction steam to a Feedwater heater.
D. Incorrect – Subcooling rises. Plausible if the definition of subcooling is not understood.

Technical Reference(s): BT03Ir4_Steam

Proposed references to be provided to applicants during examination: None

Learning Objective: BT03Ir4_Steam 11.o

Question Source: Bank – Exelon Fundamentals Thermo #2

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(14)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

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

**FLUID STATICS AND DYNAMICS**

**Explain operational implications of fluid/water hammer**

Proposed Question: #74

The plant has experienced a scram with the following:

- Reactor water level is 100” and stable.
- Emergency Condensers are NOT in service.

Which one of the following identifies the operational concern with manually initiating an Emergency Condenser under these conditions?

A. Water hammer

B. Thermal shock

C. Tube uncovering

D. Valve pressure binding

Proposed Answer: A

Explanation: With Reactor water level above 95”, the concern with initiating an Emergency Condenser is system damage due to water hammer.

B. Incorrect – The concern is water hammer. Plausible because the concern is a shock from a pressure transient. Also plausible because thermal shock is a concern in other applications in the plant.
C. Incorrect – The concern is water hammer. Plausible because tube uncovering is a concern related to other Emergency Condenser conditions (low shell level).
D. Incorrect – The concern is water hammer. Plausible because valve pressure binding is a concern related to other Emergency Condenser conditions (closing steam IVs at normal Reactor pressure).

Technical Reference(s): N1-OP-13, N1-EOP-HC attachment 9

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-207000-RBO-9

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(14)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| | | |
|---|---|---|
| Examination Outline Cross-Reference: | Level | RO |
| | Tier # | 4 |
| | Group # | |
| | K/A # | 293010 K1.06 |
| | Importance Rating | 4.1 |

**BRITTLE FRACTURE AND VESSEL THERMAL STRESS**

**State the operational concerns of uncontrolled cooldown**

Proposed Question: #75

A normal plant shutdown and cooldown is in progress in accordance with N1-OP-43C, Plant Shutdown.

Which one of the following is the cooldown rate limit and the basis for this limit?

| | Cooldown Rate Limit | Basis for Limit |
|---|---|---|
| A. | 25°F/hr | Avoid excessive stress to the Reactor vessel |
| B. | 25°F/hr | Avoid degradation of Recirculation pump seals |
| C. | 100°F/hr | Avoid excessive stress to the Reactor vessel |
| D. | 100°F/hr | Avoid degradation of Recirculation pump seals |

Proposed Answer: C

Explanation: N1-OP-43C and the Pressure and Temperature Limits Report (PTLR) require maintaining the cooldown rate ≤ 100°F/hr. This limit is based on avoiding excessive stress to the Reactor vessel.

A. Incorrect – The limit is 100°F/hr. Plausible because the PTLR gives a limit of 25°F/hr during hydrostatic and class 1 leak testing.
B. Incorrect – The limit is 100°F/hr. Plausible because the PTLR gives a limit of 25°F/hr during hydrostatic and class 1 leak testing. This limit is based on avoiding excessive stress to the Reactor vessel. Plausible because Recirculation pump seals would also be affected by changes in Reactor coolant temperature and degradation of Recirculation pump seals is a concern related to loss of coolant.
D. Incorrect – This limit is based on avoiding excessive stress to the Reactor vessel. Plausible because Recirculation pump seals would also be affected by changes in Reactor coolant temperature and degradation of Recirculation pump seals is a concern related to loss of coolant.

Technical Reference(s): BT10Ir4_Brittle_Stress, N1-OP-43C, PTLR, Technical Specification 3.2.1

Proposed references to be provided to applicants during examination: None

Learning Objective: BT10Ir4_Brittle_Stress 11

Question Source: Bank – Exelon Fundamentals Thermo #56

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(14)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | | |
|---|---|---|
| | Level | SRO |
| | Tier # | 1 |
| | Group # | 1 |
| | K/A # | 295001 AA2.13 |
| | Importance Rating | 3.9 |

**PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION**

**Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Thermal limits**

Proposed Question: #76

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

- Two Reactor Recirculation pumps (RRPs) trip.
- N1-SOP-1.3, Recirc Pump Trip at Power, is entered
- The associated RRP discharge valves are closed in accordance with N1-SOP-1.3.
- Reactor power is now approximately 86%.

Which one of the following describes:

- the status of Reactor power, in accordance with N1-SOP-1.3,

  and

- the need to apply a thermal limit penalty, in accordance with Technical Specifications and/or the Core Operating Limits Report?

| | Reactor Power | Thermal Limit Penalty |
|---|---|---|
| A. | Must be lowered | Required |
| B. | Must be lowered | NOT required |
| C. | May be raised | Required |
| D. | May be raised | NOT required |

Proposed Answer: C

Explanation: Technical Specification 3.1.7.e requires Reactor power to be limited to 90% when operating with only three Reactor Recirculation pumps in service. With power currently at 86%, it can be raised. Technical Specification 3.1.7.e also requires referencing the COLR for APLHGR and MCPR values for three loop operation. The COLR requires a thermal limit penalty to be applied for MCPR while in three loop operation.

A. Incorrect – Reactor power is allowed to be as high as 90%. Plausible because this would be correct if Reactor power was >90% or if the limit was lower.
B. Incorrect – Reactor power is allowed to be as high as 90%. Plausible because this would be correct if Reactor power was >90% or if the limit was lower. A thermal limit penalty must be applied. Plausible because the MAPLHGR multiplier for three loop operation is currently 1.00, meaning there is no actual penalty on that thermal limit, despite it being referenced by Technical Specification 3.1.7.
D. Incorrect – A thermal limit penalty must be applied. Plausible because the MAPLHGR multiplier for three loop operation is currently 1.00, meaning there is no actual penalty on that thermal limit, despite it being referenced by Technical Specification 3.1.7.

Technical Reference(s): N1-SOP-1.3, Technical Specification 3.1.7, COLR

Proposed references to be provided to applicants during examination: Technical Specification 3.1.7, COLR

Learning Objective: 1101-SOP1.3C01 EO-2

Question Source: Modified Bank – 2008 NRC #93

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

| Examination Outline Cross-reference: | Level | RO | | SRO |
|---|---|---|---|---|
| | Tier # | 2 | | |
| | Group # | 2 | | |
| | K/A # | 202001 2.2.22 | | |
| | Importance Rating | | | 4.7 |

(K&A Statement) Equipment Control - Knowledge of limiting conditions for operations and safety limits.(Recirc)

Proposed Question: SRO 93

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

- Reactor Recirc Pump 14 is secured and isolated
- Reactor Recirc Pump 11 develops a failure of the high pressure mechanical seal
- Reactor Recirc Pump 11 will be removed from service and isolated

Which one of the following is true for these conditions?

A. Power must be reduced to less than 90% and thermal limits verified

B. The reactor must be manually scrammed to prevent exceeding MCPR limits

C. Power can be restored to 100% provided the MCPR limit is increased by 0.02

D. Power must be reduced to less than 90.5% within ten minutes to prevent exceeding the APLHGR limits

Proposed Answer: A.

Explanation (Optional):

A. Correct – Based on the conditions, the plant will be in three loop operation. In accordance with TS 3.1.7.e, when operating with three recirculation loops and the two remaining loops isolated or unisolated, the reactor may operate at 90% of full power. Both APLHGR and MCPR limits must be adjusted per the Core Operating Limits report.

B. Incorrect – Three loop operation is permitted. A MCPR penalty must be applied to thermal limits. Nothing in the question implies that MCPR is violated. Even if MCPR were being violated, 15 minutes is allowed per Tech Specs to take action to restore operation to within limits.

C. Incorrect – Tech Specs limit 3 loop operation to 90%.

D. Incorrect – Power must be less than 90% for 3 loop operation. 90.5% is based on 4 loop operating limits.

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | Level | SRO |
|---|---|---|
| | Tier # | 1 |
| | Group # | 1 |
| | K/A # | 295003 2.4.47 |
| | Importance Rating | 4.2 |

**PARTIAL OR COMPLETE LOSS OF AC POWER**

**Ability to diagnose and recognize trends in an accurate and timely manner using the appropriate control room reference material (reference potential)**

Proposed Question: #77

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

- Annunciator A8-1-3, 115 KV. BUS LOW VOLTAGE, alarms.
- N1-SOP-33A.3, Major 115 KV Grid Disturbances, is entered.
- Investigation reveals 115 KV system voltage momentarily dropped to a low of 105 KV.
- 115 KV system voltage has recovered slightly and is now 109 KV and stable.
- Power Control reports that the Low Voltage Contingency Computer is NOT available.

Which one of the following describes the need to enter an LCO in:

- Technical Specification 3.1.8, High Pressure Coolant Injection,

  and

- Technical Specification 3.6.3, Emergency Power Sources,

in accordance with N1-SOP-33A.3?

| | Technical Specification 3.1.8 LCO Entry | Technical Specification 3.6.3 LCO Entry |
|---|---|---|
| A. | Required | Required |
| B. | Required | NOT required |
| C. | NOT required | Required |
| D. | NOT required | NOT required |

Proposed Answer: A

Explanation: N1-SOP-33A.3 contains multiple overrides based on status of 115 KV system voltage and status of the low voltage post contingency alarm. With voltage below 109.3 KV, Arrow C applies. With this procedure section entered, HPCI, Line 1, and Line 4 must be declared inoperable. This requires entering LCOs in both Technical Specifications 3.1.8 and 3.6.3.

B. Incorrect – Technical Specification 3.6.3 must be entered. Plausible because voltage has recovered back towards the limit. Also plausible because Lines 1 and 4 are still connected to the plant.
C. Incorrect – Technical Specification 3.1.8 must be entered. Plausible because voltage is just below the 109.3 KV threshold associated with this Technical Specification entry. Also plausible because supported systems are usually not declared inoperable solely due to degradation of the supporting system (cascading Tech Specs).
D. Incorrect – Technical Specification 3.1.8 must be entered. Plausible because voltage is just below the 109.3 KV threshold associated with this Technical Specification entry. Also plausible because supported systems are usually not declared inoperable solely due to degradation of the supporting system (cascading Tech Specs). Technical Specification 3.6.3 must be entered. Plausible because voltage has recovered back towards the limit. Also plausible because Lines 1 and 4 are still connected to the plant.

Technical Reference(s): N1-SOP-33A.3

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-SOP33A3C01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2 & 5)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| | | |
|---|---|---|
| Examination Outline Cross-Reference: | Level | SRO |
| | Tier # | 1 |
| | Group # | 1 |
| | K/A # | 295005 AA2.06 |
| | Importance Rating | 2.9 |

**MAIN TURBINE GENERATOR TRIP**

**Ability to determine and/or interpret the following as they apply to MAIN TURBINE GENERATOR TRIP: Feedwater temperature**

Proposed Question: #78

The plant is operating at 20% power during a startup with the following:

- Ten minutes ago, the Main Generator was synchronized to the grid and loaded to 60 MWe per N1-OP-32, Generator.
- No subsequent steps in the startup have been performed yet.
- Then, a Main Turbine trip occurs.
- The Reactor does NOT scram.
- Reactor power begins to slowly rise.

Which one of the following describes whether this is the correct Reactor response to the Main Turbine trip and a required action to be directed, in accordance with the SOPs and EOPs?

This is…

A. the correct Reactor response. Enter N1-SOP-31.1, Turbine Trip, and direct a manual Reactor scram.

B. the correct Reactor response. Enter N1-SOP-31.1, Turbine Trip, and direct a Turbine shutdown. A manual Reactor scram is NOT required.

C. NOT the correct Reactor response. Enter N1-SOP-1.5, Unplanned Power Change, to determine the cause of the power rise. A manual Reactor scram is NOT required.

D. NOT the correct Reactor response. Enter N1-EOP-RPV, RPV Control, and direct a manual Reactor scram. If the Reactor still does NOT scram, transition to N1-EOP-C5, Failure to Scram.

Proposed Answer: B

Explanation: This is the correct Reactor response. The Reactor does NOT scram on the Main Turbine trip because Reactor power is below 35%. Reactor power rises due to the loss of Feedwater heating on the Main Turbine trip. N1-SOP-31.1 entry is required. The subsequent actions of N1-SOP-31.1 require directing a Turbine shutdown, but not a Reactor shutdown or Reactor scram.

A. Incorrect – A manual Reactor scram is not required. Plausible because a Reactor scram would be required if Reactor power was above 35%.
C. Incorrect – This is the correct response to a Turbine trip <35% power. Plausible because an automatic scram would be expected if >35% power. Also plausible that Reactor power should not be rising if the Feedwater temperature effect is not understood. Also plausible because N1-SOP-1.5 entry is warranted for the rising Reactor power.
D. Incorrect – This is the correct response to a Turbine trip <35% power. Plausible because an automatic scram would be expected if >35% power. Also plausible that Reactor power should not be rising if the Feedwater temperature effect is not understood. Also plausible because this would be the correct response if >35% power and the Reactor failed to scram.

Technical Reference(s): N1-SOP-1.5, N1-SOP-31.1, N1-EOP-RPV

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-SOP31.1C01 EO-2

Question Source: Bank – 2015 NRC #78

Question History: 2015 NRC #78

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| | | |
|---|---|---|
| Examination Outline Cross-Reference: | Level | SRO |
| | Tier # | 1 |
| | Group # | 1 |
| | K/A # | 295024 2.4.22 |
| | Importance Rating | 4.4 |

**HIGH DRYWELL PRESSURE**

**Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations**

Proposed Question: #79

The plant has experienced an earthquake with the following:

- Reactor water level is 50” and lowering slowly.
- Drywell pressure is 4 psig and rising slowly.
- Spent Fuel Pool water level is 3” below the level of the weirs and slowly lowering.

Which one of the following describes the required prioritization, in accordance with N1-EOP-B-HOT, NMP1 Emergency Operating Procedures Basis Document, and OP-NM-101-111-1001, Transient Mitigation Guidelines?

The EOP Director…

A. must make Reactor injection the priority

B. must make Containment Control actions the priority

C. must make makeup to the Spent Fuel Pool the priority

D. is allowed to set the priority based on plant conditions

Proposed Answer: D

Explanation: N1-EOP-B-HOT and OP-NM-101-111-1001 both specify that there is no pre-determined prioritization for equipment use between EOPs. EOPs are to be implemented concurrently and priorities are set as determined by the EOP Director based on plant conditions.

A. Incorrect – Reactor injection does NOT always take priority over Containment Control activities and Spent Fuel Pool makeup. Plausible because adequate core cooling, if challenged, is given priority over other concerns in the EOPs.
B. Incorrect – Containment Control activities do NOT always take priority over Reactor injection and Spent Fuel Pool makeup. Plausible there are situations in the SAPs where Containment Control activities are given the priority.
C. Incorrect – Spent Fuel Pool makeup does NOT always take priority over Reactor injection and Containment Control activities. Plausible because low Spent Fuel Pool level can lead to a direct release to the environment, whereas the Reactor is inside a Primary Containment.

Technical Reference(s): N1-EOP-B-HOT, OP-NM-101-111-1001

Proposed references to be provided to applicants during examination: None

Learning Objective: NS-TMG-EO-01

Question Source: Modified Bank – 2020 NRC #100

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference: Level SRO
Tier # 3
Group #
K/A # 2.4.22
Importance Rating 4.4

**Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations.**

Proposed Question: #100

The plant has experienced an earthquake with the following:

- The Diesel Fire pump is the only available pump.
- Firefighting activities are in progress.
- Reactor water level is 50" and slowly lowering.
- Spent Fuel Pool water level is 3" below the level of the weirs and slowly lowering.

Which one of the following describes the prioritization for use of the Diesel Fire pump, in accordance with NER-1M-095, NMP1 Emergency Operating Procedures and Severe Accident Procedures Basis Document, and OP-NM-101-111-1001, Transient Mitigation Guidelines?

The EOP Director…

A. is allowed to set the priority for use of the Diesel Fire pump based on plant conditions.

B. must make lining up the Diesel Fire pump for Reactor injection the priority.

C. must make lining up the Diesel Fire pump for Spent Fuel Pool makeup the priority.

D. must make maintaining the Diesel Fire pump lined up to the Fire Water header the priority.

Proposed Answer: A

Explanation: NER-1M-095 and OP-NM-101-111-1001 both specify that there is no pre-determined prioritization for pump use or between EOPs. EOPs are to be implemented concurrently and priorities are set as determined by the EOP Director based on plant conditions.

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | | |
|---|---|---|
| | Level | SRO |
| | Tier # | 1 |
| | Group # | 1 |
| | K/A # | 295028 2.4.14 |
| | Importance Rating | 4.5 |

**HIGH DRYWELL TEMPERATURE**

**Knowledge of general guidelines for emergency and abnormal operating procedures usage**

Proposed Question: #80

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

- The Reactor has been manually scrammed.
- N1-EOP-PC, Primary Containment Control, has been entered due to high Drywell temperature and pressure, only.

Which one of the following describes how to implement N1-EOP-PC, in accordance with the EOPs and bases?

A. All legs are entered. There is a pre-determined order for which leg to execute first.

B. All legs are entered. There is NOT a pre-determined order for which leg to execute first.

C. Only the DWT and PCP legs are entered. There is a pre-determined order for which leg to execute first.

D. Only the DWT and PCP legs are entered. There is NOT a pre-determined order for which leg to execute first.

Proposed Answer: B

Explanation: When an EOP entry condition is met, the EOP is entered and all legs are entered concurrently. There is no pre-determined prioritization between the legs. Prioritization is left up to the EOP director.

A. Incorrect – There is no pre-determined prioritization between the legs. Prioritization is left up to the EOP director. Plausible that a pre-determined order would be given to ensure consistent implementation.
C. Incorrect – When an EOP entry condition is met, the EOP is entered and all legs are entered concurrently. Plausible because not all legs have associated entry conditions met. There is no pre-determined prioritization between the legs. Prioritization is left up to the EOP director. Plausible that a pre-determined order would be given to ensure consistent implementation.
D. Incorrect – When an EOP entry condition is met, the EOP is entered and all legs are entered concurrently. Plausible because not all legs have associated entry conditions met.

Technical Reference(s): N1-EOP-PC, N1-EOP-B-HOT

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP4C01 EO-2

Question Source: Bank – 2020 NRC #99

Question History: 2020 NRC #99

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(5)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | | |
|---|---|---|
| | Level | SRO |
| | Tier # | 1 |
| | Group # | 1 |
| | K/A # | 295030 EA2.02 |
| | Importance Rating | 3.8 |

**LOW SUPPRESSION POOL WATER LEVEL**

**Ability to determine and/or interpret the following as they apply to LOW SUPPRESSION POOL WATER LEVEL: Suppression pool temperature**

Proposed Question: #81

The plant was operating at 100% power when a seismic event resulted in the following:

- Reactor power is 6%.
- Reactor water level has been lowered in accordance with N1-EOP-C5, RPV Control - ATWS.
- Reactor pressure is 750 psig and slowly rising.
- Reactor pressure is being controlled between 700 and 900 psig with Emergency Condensers and ERVs.
- Torus water temperature is 117°F and slowly rising.
- N1-EOP-PC, Primary Containment Control, has been entered.
- Torus makeup is in service in accordance with N1-EOP-1 Attachment 18, Raw Water to Torus Makeup.
- Torus water level is 8.1 feet and lowering slowly.

**Note:** Heat Capacity Temperature Limit curve is provided on the next page.

Which one of the following describes the required control of Reactor pressure, in accordance with the EOPs?

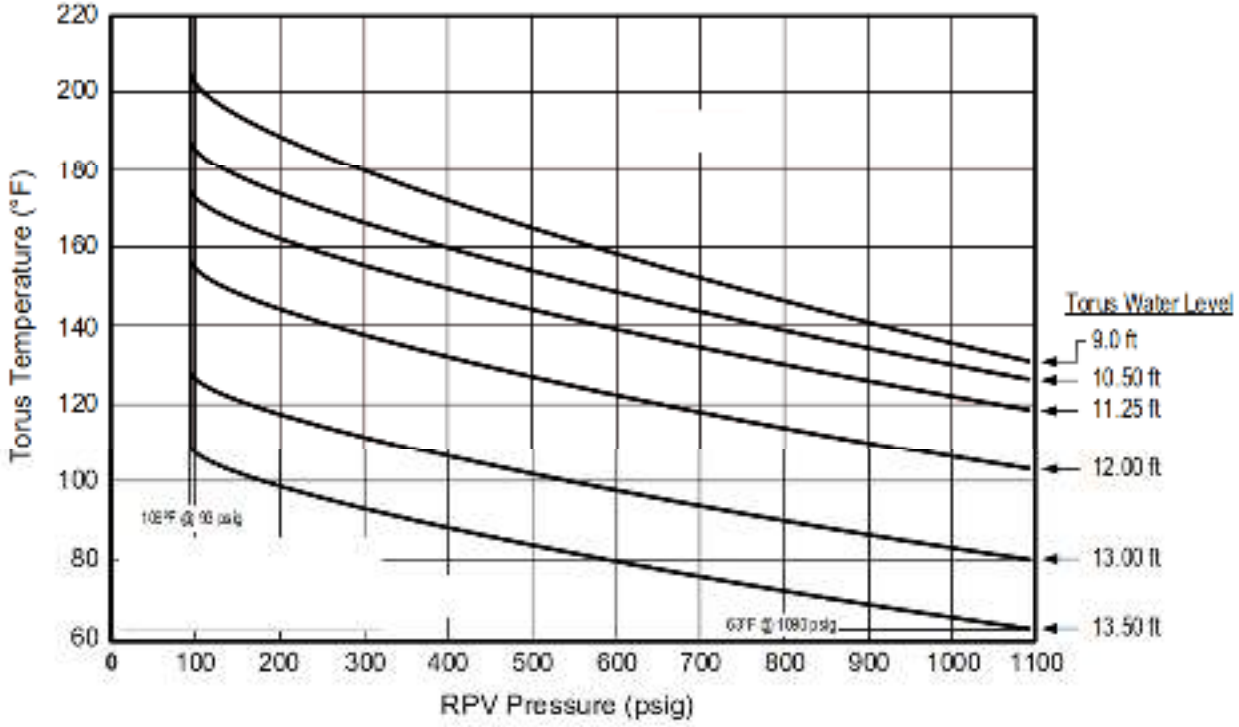
A. Continued use of the entire current Reactor pressure band is acceptable.

B. An RPV Blowdown is required based on the Torus Level leg of N1-EOP-PC.

C. An RPV Blowdown is required based on the Torus Temperature leg of N1-EOP-PC.

D. Reactor pressure must be maintained in a lower Reactor pressure band, but an RPV Blowdown is NOT required.

# M Heat Capacity Temperature Limit



Proposed Answer: B

Explanation: The Torus Level leg of N1-EOP-PC requires an RPV Blowdown if you cannot maintain Torus water level above 8 feet. Since the highest capacity Torus makeup method is already in service and Torus water level is 8.1 feet and lowering, Torus water level cannot be maintained above 8 feet. This requires an RPV Blowdown.

A. Incorrect – An RPV Blowdown is required. Plausible because, using the 9.0' HCTL curve, this would be correct.
C. Incorrect – An RPV Blowdown is not required based on Torus temperature. Plausible because this would be correct for higher Torus water levels or if the Good/Bad sides of the HCTL curves were reversed.
D. Incorrect – An RPV Blowdown is required. Plausible because this would be correct for some of the higher Torus water level HCTL curves.

Technical Reference(s): N1-EOP-PC, N1-EOP-1

Proposed references to be provided to applicants during examination: Embedded graphic

Learning Objective: 1101-EOP4C01 EO-2

Question Source: Bank – 2015 Cert #75

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | | |
|---|---|---|
| | Level | SRO |
| | Tier # | 1 |
| | Group # | 1 |
| | K/A # | 295037 EA2.06 |
| | Importance Rating | 4.3 |

**SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN**

**Ability to determine and/or interpret the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: Reactor pressure**

Proposed Question: #82

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

- Multiple control rods are fully withdrawn.
- Reactor power is 12% and stable.
- Reactor water level is being controlled -84” to -41” using Feedwater.
- Reactor pressure is 750 psig and lowering on Turbine Bypass Valves (TBVs).

Then, the following occurs:

- An RPV Blowdown becomes required due to Secondary Containment temperatures.
- Emergency Condensers (ECs) are initiated.

Which one of the following describes the required control of Reactor pressure, in accordance with N1-EOP-C2, RPV Blowdown?

A. Use TBVs to lower Reactor pressure. Reduce Reactor pressure to as low as possible.

B. Use TBVs to lower Reactor pressure. Control ECs and TBVs to maintain Reactor pressure above 350 psig but as low as possible.

C. Open 4 ERVs to lower Reactor pressure. Then, reduce Reactor pressure to as low as possible.

D. Open 4 ERVs to lower Reactor pressure. Then, control ECs and ERVs to maintain Reactor pressure above 350 psig but as low as possible.

Proposed Answer: B

Explanation: With multiple control rods still fully withdrawn, the Reactor will not stay shutdown without boron, therefore the ATWS leg of N1-EOP-C2 must be executed. This leg requires lowering Reactor pressure to as low as possible above 350 psig using ECs and TBVs.

A. Incorrect – Reactor pressure must be maintained above 350 psig. Plausible because N1-EOP-C2 requires lowering Reactor pressure to as low as possible under non-ATWS conditions.
C. Incorrect – 4 ERVs are not opened. Plausible because this would be correct under non-ATWS conditions. Reactor pressure must be maintained above 350 psig. Plausible because N1-EOP-C2 requires lowering Reactor pressure to as low as possible under non-ATWS conditions.
D. Incorrect – 4 ERVs are not opened. Plausible because this would be correct under non-ATWS conditions.

Technical Reference(s): N1-EOP-C2

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOPC2C01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| | | |
|---|---|---|
| Examination Outline Cross-Reference: | Level | SRO |
| | Tier # | 1 |
| | Group # | 2 |
| | K/A # | 295007 2.4.6 |
| | Importance Rating | 4.7 |

**HIGH REACTOR PRESSURE**

**Knowledge of emergency and abnormal operating procedures major action categories**

Proposed Question: #83

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

- A Station Blackout occurs.
- All control rods insert.
- N1-SOP-33A.2, Station Blackout, is entered.
- N1-EOP-RPV, RPV Control, is entered due to low Reactor water level and high Reactor pressure.
- Both Emergency Condensers automatically initiate.
- Reactor pressure is 1000 psig and lowering slowly.

Which one of the following describes the required control of Reactor pressure, in accordance with plant procedures?

A. Cycle the Emergency Condensers to stabilize Reactor pressure near the current value

B. Cycle the Emergency Condensers to establish a cooldown, but maintain the rate less than 100°F/hr

C. Maintain one Emergency Condenser in continuous service and secure the other Emergency Condenser

D. Maintain both Emergency Condensers in continuous service.

Proposed Answer: D

Explanation: EOPs are higher tiered documents than SOPs, and therefore when a conflict exists between the direction in EOPs and SOPs, the EOP direction normally is followed. However, Reactor pressure control during a Station Blackout is a special case in which the SOP direction is followed over the normal EOP direction. Under normal N1-EOP-RPV direction, the Emergency Condensers would need to be cycled in and out of service to maintain the cooldown rate less than 100°F/hr. However, since N1-SOP-33A.2 is the overriding document in this special case, the Emergency Condensers are required to be maintained in continuous service, even though the normal EOP cooldown rate is expected to be violated.

A. Incorrect – N1-SOP-33A.2 requires maintaining both ECs in continuous service. Plausible because there are conditions where Reactor pressure is stabilized (such as initial ATWS response) and stabilizing pressure would assist in stable level control.
B. Incorrect – N1-SOP-33A.2 requires maintaining both ECs in continuous service. Plausible because this is the normal direction in N1-EOP-RPV.
C. Incorrect – N1-SOP-33A.2 requires maintaining both ECs in continuous service. Plausible because this would lower the cooldown rate while still conserving some amount of power.

Technical Reference(s): N1-SOP-33A.2, N1-EOP-RPV, N1-EOP-B-HOT

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-SOP33A.2C01 EO-2

Question Source: Bank – 2017 NRC #80

Question History: 2017 NRC #80

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(5)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| | | |
|---|---|---|
| Examination Outline Cross-Reference: | Level | SRO |
| | Tier # | 1 |
| | Group # | 2 |
| | K/A # | 295022 2.2.22 |
| | Importance Rating | 4.7 |

**LOSS OF CONTROL ROD DRIVE PUMPS**

**Knowledge of limiting conditions for operation and safety limits**

Proposed Question: #84

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

- CRD pump 11 is in service.
- Annunciator F3-1-5, CRD CHARGING WTR PRESSURE HI/LO, alarms.
- Charging Water pressure is 900 psig and stable.
- CRD system flow is $10x10^3$ lbm/hr (20 gpm) and stable.
- An Operator reports from the field that a leak has developed from the discharge flange of CRD pump 11.

Which one of the following describes the need to enter an LCO in:

- Technical Specification 3.1.1, Control Rods,

  and

- Technical Specification 3.1.6, Control Rod Coolant Injection?

| | Technical Specification 3.1.1 LCO Entry | Technical Specification 3.1.6 LCO Entry |
|---|---|---|
| A. | Required | Required |
| B. | Required | NOT required |
| C. | NOT required | Required |
| D. | NOT required | NOT required |

Proposed Answer: C

Explanation: Technical Specification 3.1.6 and the associated bases require both CRD pumps to be able to supply at least 50 gpm. The given leak is preventing CRD pump 11 from being able to produce this flow. Therefore, CRD pump 11 must be declared inoperable and LCO entry is required in Technical Specification 3.1.6. The given conditions do not require LCO entry in Technical Specification 3.1.1.

A. Incorrect – LCO entry in Technical Specification 3.1.1 is not required. Plausible because Charging Water header pressure is below the 940 psig required accumulator pressure (LCO entry would be required if Annunciator F3-2-5 alarmed and accumulator pressures were less than 940 psig).
B. Incorrect – LCO entry in Technical Specification 3.1.1 is not required. Plausible because Charging Water header pressure is below the 940 psig required accumulator pressure (LCO entry would be required if Annunciator F3-2-5 alarmed and accumulator pressures were less than 940 psig). LCO entry is required in Technical Specification 3.1.6. Plausible because CRD pump 11 is still running and capable of providing significant flow.
D. Incorrect – LCO entry is required in Technical Specification 3.1.6. Plausible because CRD pump 11 is still running and capable of providing significant flow.

Technical Reference(s): ARPs F3-1-5 and F3-2-5, Technical Specifications 3.1.1 and 3.1.6 and associated bases

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-201001-RBO-14

Question Source: Modified Bank – 2010 Cert #85

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

Emergency Procedures / Plan: Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures (Loss of CRD pumps).

Proposed Question: SRO Question # 85

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

- CRD pump 11 is in service
- Annunciator F3-1-5, CRD CHARGING WTR PRESSURE HI/LO, is in alarm
- Charging Water pressure lowers and stabilizes at 900 psig
- CRD system flow lowers and stabilizes at $10 \times 10^3$ lbm/hr (20 gpm)
- An operator reports from the field that a leak has developed from the discharge flange of CRD pump 11

Which one of the following describes the entry requirements for SOP-5.1, Loss of CRD, and the operability of CRD pump 11 per Technical Specification 3.1.6, Control Rod Drive Pump Coolant Injection?

Entry into SOP-5.1 is __(1)__. CRD pump 11 is __(2)__ per Technical Specification 3.1.6.

| | | (1) | (2) |
|---|---|---|---|
| A. | | Required due to low Charging Water pressure | Operable |
| B. | | Required due to low Charging Water pressure | Inoperable |
| C. | | NOT required unless CRD system flow lowers below 10 gpm | Operable |
| D. | | NOT required unless CRD system flow lowers below 10 gpm | Inoperable |

Proposed Answer: B

Explanation (Optional):

| | | |
|---|---|---|
| A. | | Incorrect - TS 3.1.6 bases require a CRD pump to produce at least 50 gpm flow for operability. With CRD pump 11 only providing a system flow of 20 gpm, it must be declared inoperable. |
| B. | | Correct - Annunciator F3-1-5 is an entry condition for SOP-5.1. TS 3.1.6 bases require a CRD pump to produce at least 50 gpm flow for operability. With CRD pump 11 only providing a system flow of 20 gpm, it must be declared inoperable. |
| C. | | Incorrect - Annunciator F3-1-5 is an entry condition for SOP-5.1. TS 3.1.6 bases require a CRD pump to produce at least 50 gpm flow for operability. With CRD pump 11 only providing a system flow of 20 gpm, it must be declared inoperable. |
| D. | | Incorrect - Annunciator F3-1-5 is an entry condition for SOP-5.1. |

**Form 4.2-1 Written Examination Question Worksheet**

| | | |
|---|---|---|
| Examination Outline Cross-Reference: | Level | SRO |
| | Tier # | 1 |
| | Group # | 2 |
| | K/A # | 295029 EA2.03 |
| | Importance Rating | 3.6 |

**HIGH SUPPRESSION POOL WATER LEVEL**

**Ability to determine and/or interpret the following as they apply to HIGH SUPPRESSION POOL WATER LEVEL: Drywell/containment water level**

Proposed Question: #85

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

- Reactor water level is 30" and rising slowly.
- Reactor pressure is 350 psig and lowering slowly.
- Core Spray is injecting to the Reactor.
- Containment Sprays are in service per N1-EOP-1 Attachment 17, Auto or Manual Initiation of Containment Spray.
- Drywell pressure is 5 psig and lowering slowly.
- Water is being pumped from the Torus to Radwaste per N1-EOP-1 Attachment 15, Torus Water to Waste Collector.
- Torus water level is 13.6' and rising slowly.
- Torus water temperature is 125°F and rising slowly.

Which one of the following is required, in accordance with the Emergency Operating Procedures?

A. Perform an RPV Blowdown

B. Vent the Primary Containment

C. Secure Containment Sprays and initiate Torus Cooling

D. Stop injection from Core Spray not needed for adequate core cooling

Proposed Answer: A

Explanation: Torus water level is above 13.5' and continues to rise with Torus pump down (N1-EOP-1 attachment 15) already in progress. This requires an RPV Blowdown per the Torus level leg of N1-EOP-PC.

B. Incorrect – Venting the Primary Containment is not required. Plausible if the candidate believed the concern on Torus water level reaching 13.5' had to do with operability of Primary Containment venting, which is of concern at higher containment water levels.
C. Incorrect – Securing Containment Sprays is not required. Plausible because Containment Sprays are required to be secured when Drywell pressure lowers further, and Torus temperature is elevated, which makes Torus Cooling appropriate.
D. Incorrect – There is no need to secure injection from Core Spray because it injects from the Torus. Plausible because this would be correct if the injection sources was from outside of the Containment, such as Feedwater.

Technical Reference(s): N1-EOP-PC

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOPPCC01 EO-2

Question Source: Modified Bank – 2020 NRC #85

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

**High Suppression Pool Water Level**

**Ability to determine and/or interpret the following as they apply to HIGH SUPPRESSION POOL WATER LEVEL: Drywell/containment water level**

Proposed Question: #85

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

- Reactor water level is 30" and rising.
- Reactor pressure is 450 psig and lowering.
- Feedwater and CRD are injecting to the Reactor.
- Containment Sprays are in service.
- Drywell pressure is 5 psig and lowering.
- Torus water level is 13.4' and slowly rising.
- Torus water temperature is 125°F and slowly rising.

Which one of the following is required, in accordance with the Emergency Operating Procedures?

A. Perform an RPV Blowdown.

B. Vent the Primary Containment.

C. Secure Containment Sprays and initiate Torus Cooling.

D. Stop injection from Feedwater and CRD not needed for adequate core cooling.



Proposed Answer: D

Explanation: Torus water level is approaching 13.5'. Per N1-EOP-4, this requires stopping injection into the RPV from sources outside the Primary Containment (which includes Feedwater and CRD) not needed for adequate core cooling. With Reactor water level at 30" and rising, adequate core cooling is currently assured and there is more injection than needed to maintain adequate core cooling, therefore at least some of the injection from Feedwater and CRD can be stopped to help mitigate the rise in Torus water level.

**Form 4.2-1 Written Examination Question Worksheet**

| | | |
|---|---|---|
| Examination Outline Cross-Reference: | Level | SRO |
| | Tier # | 2 |
| | Group # | 1 |
| | K/A # | 205000 A2.05 |
| | Importance Rating | 4.0 |

**SHUTDOWN COOLING SYSTEM**

**Ability to (a) predict the impacts of the following on the SHUTDOWN COOLING SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: System isolation**

Proposed Question: #86

The plant is shutdown with the following:

- Shutdown Cooling (SDC) loop 11 is in service.
- Reactor coolant temperature is 140°F and slowly lowering.

Then, 38-01, REACTOR SHUTDOWN COOLING SUPPLY ISOL VAL 11, spuriously closes.

- SDC pump 11 trips.
- N1-SOP-6.1, Loss of Rx Cavity Level / Decay Heat Removal, has been entered.
- All attempts to re-open 38-01 have been unsuccessful.

Which one of the following describes:

- the ability to place an alternate loop of Shutdown Cooling in service while 38-01 is closed

  and

- a required action,

in accordance with N1-SOP-6.1?

| | Can an alternate loop of SDC be placed in service while 38-01 is closed? | Required Action |
|---|---|---|
| A. | Yes | Verify Secondary Containment is established |
| B. | Yes | Secure all Water Inventory Control (WIC) activities |
| C. | No | Verify Secondary Containment is established |
| D. | No | Secure all Water Inventory Control (WIC) activities |

Proposed Answer: C

Explanation: 38-01 is in the common suction path for all 3 SDC cooling loops. There is no alternate or parallel suction path. Therefore, with 38-01 closed, an alternate loop of SDC cannot be placed in service. N1-SOP-6.1 requires Secondary Containment be set given the loss of decay heat removal.

A. Incorrect – An alternate loop of SDC cannot be placed in service with 38-01 closed. Plausible that there would be an alternate or parallel suction path. Also plausible because this would be correct for closure of 38-03.
B. Incorrect – An alternate loop of SDC cannot be placed in service with 38-01 closed. Plausible that there would be an alternate or parallel suction path. Also plausible because this would be correct for closure of 38-03. It is not required to secure all WIC activities. Plausible because maintaining Reactor cavity inventory maximizes the available heat capacity for removal of decay heat.
D. Incorrect – It is not required to secure all WIC activities. Plausible because maintaining Reactor cavity inventory maximizes the available heat capacity for removal of decay heat.

Technical Reference(s): C-18018-C, N1-SOP-6.1

Proposed references to be provided to applicants during examination: None

Learning Objective: 1010-SOP6.1C01 EO-2

Question Source: Modified Bank – 2009 NRC #78

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

| QUESTION | 78 |
|---|---|

The plant is in a refueling outage with Core Shuffle Phase II in progress, when the following events occur:

- Shutdown Cooling (SDC) is lost
- Actual temperature measurements indicate Reactor water temperature is projected to exceed 140°F

Which one of the following is required, in accordance with N1-SOP-6.1, Loss of SFP/Rx Cavity Level/Decay Heat Removal?

| | |
|---|---|
| A. | Perform a time to boil estimation. |
| B. | Verify Secondary Containment established. |
| C. | Secure all Operations with the Potential for Draining the Reactor Vessel (OPDRVs). |
| D. | Return any core component being transferred to the nearest storage location in Spent Fuel Pool OR Core. |

| | |
|---|---|
| K&A # | 295021 AA2.04 |
| Importance Rating | 3.6 |
| QUESTION 78 | |
| K&A Statement: | Ability to determine and/or interpret the following as they apply to LOSS OF SHUTDOWN COOLING: Reactor water temperature. |
| Justification: | |
| A. | Correct – Per N1-SOP-6.1 if the safety assessment level for DHR <N+1 and temperature is predicted to exceed 140F, then perform a time to boil. |
| B. | Incorrect because the procedure under pool or vessel inventory lowering requires this action. But plausible if the candidate is not familiar with actions in N1-SOP-6.1. |
| C. | Incorrect but plausible because N1-SOP-6.1 also covers actions for loss of vessel inventory while refueling. |
| D. | Incorrect because the procedure under pool or vessel inventory lowering requires this action. But plausible if the candidate is not familiar with actions in N1-SOP-6.1. |

| | | | |
|---|---|---|---|
| References: | N1-SOP-6.1 | Student Ref: | None |
| Learning Objective: | N/A | | |
| Question source: | NMP1 | | |
| Question History: | NRC 2004 Exam | | |

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | | |
|---|---|---|
| | Level | SRO |
| | Tier # | 2 |
| | Group # | 1 |
| | K/A # | 212000 A2.18 |
| | Importance Rating | 3.9 |

**REACTOR PROTECTION SYSTEM**

**Ability to (a) predict the impacts of the following on the REACTOR PROTECTION SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: SCRAM air header low pressure**

Proposed Question: #87

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

- Annunciator L1-3-7, INST AIR SYSTEM, is in alarm.
- Computer point A193, IA DRYERS OR FILTERS, is in alarm due to both high differential pressure across the filters and low pressure downstream of the filters.
- Instrument Air Header pressure indicates 100 psig and stable in the Control Room.
- Annunciator F3-3-2, CRD CONTROL AIR PRESSURE HI-LO, is in alarm.
- Scram Air Header pressure indicates 59 psig and slowly lowering.
- NO other abnormal CRD or RPS conditions have been observed yet.

Which one of the following describes:

- the potential impact of the low Scram Air Header pressure

  and

- whether a Reactor scram is currently required,

in accordance with N1-SOP-20.1, Instrument Air Failure?

| | Potential Impact | Is a Reactor scram currently required? |
|---|---|---|
| A. | Loss of scram capability | Yes |
| B. | Loss of scram capability | No |
| C. | Control rod drifts or scrams | Yes |
| D. | Control rod drifts or scrams | No |

Proposed Answer: D

Explanation: The concern with lowering Scram Air Header pressure is control rod drifts or scrams. N1-SOP-20.1 requires scramming the Reactor one or more control rods drift or scram. Since no other abnormal CRD or RPS conditions have been observed, a Reactor scram is not currently required.

A. Incorrect – The concern with lowering Scram Air Header pressure is control rod drifts or scrams. Plausible that the low air pressure would cause water leaking into the SDV and not draining properly, which could lead to loss of scram function due to hydraulic lock. A Reactor scram is not currently required. Plausible because Scram Air Header pressure is below the alarm setpoint, below the pressure listed in the N1-SOP-20.1 discussion section, and below the value requiring a scram in previous revisions of N1-SOP-20.1.
B. Incorrect – The concern with lowering Scram Air Header pressure is control rod drifts or scrams. Plausible that the low air pressure would cause water leaking into the SDV and not draining properly, which could lead to loss of scram function due to hydraulic lock.
C. Incorrect – A Reactor scram is not currently required. Plausible because Scram Air Header pressure is below the alarm setpoint, below the pressure listed in the N1-SOP-20.1 discussion section, and below the value requiring a scram in previous revisions of N1-SOP-20.1.

Technical Reference(s): N1-SOP-20.1

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-SOP20.1C01 EO-2

Question Source: Modified Bank – 2023 NRC #79

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

**PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR**

**Ability to interpret and execute procedure steps**

Proposed Question: #79

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

- Annunciator L1-3-7, INST AIR SYSTEM, is in alarm.
- Computer point A193, IA DRYERS OR FILTERS, is in alarm due to both high differential pressure across the filters and low pressure downstream of the filters.
- Annunciator F3-3-2, CRD CONTROL AIR PRESSURE HI-LO, is in alarm.
- Scram Air Header pressure indicates 65 psig and slowly lowering.
- Instrument Air Header pressure indicates 100 psig and stable in the Control Room.

Which one of the following describes the procedure to be entered and the required action?

| | Procedure to Be Entered | | Required Action |
|---|---|---|---|
| A. | N1-SOP-5.1,<br>Loss of Control Rod Drive | | Scram the Reactor if Scram Air Header pressure reaches 60 psig |
| B. | N1-SOP-5.1,<br>Loss of Control Rod Drive | | Scram the Reactor if one or more control rods drift or scram |
| C. | N1-SOP-20.1,<br>Instrument Air Failure | | Scram the Reactor if Scram Air Header pressure reaches 60 psig |
| D. | N1-SOP-20.1,<br>Instrument Air Failure | | Scram the Reactor if one or more control rods drift or scram |



Proposed Answer: D

Explanation: N1-SOP-5.1 covers multiple different CRD system malfunctions, however lowering Instrument Air and Scram Air Header pressure requires entry into N1-SOP-20.1, not N1-SOP-5.1. N1-SOP-20.1 does not require a scram unless one or more control rods drift or scram

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | Level | SRO |
|---|---|---|
| | Tier # | 2 |
| | Group # | 1 |
| | K/A # | 215003 2.2.21 |
| | Importance Rating | 4.1 |

**INTERMEDIATE RANGE MONITOR SYSTEM**

**Knowledge of pre- and post-maintenance operability requirements**

Proposed Question: #88

Preparations are underway for a plant startup with the following:

- All control rods are full in.
- The Reactor Mode Switch is in SHUTDOWN.
- Maintenance was performed on IRMs during the shutdown.
- Post-maintenance testing reveals:
  - The upscale trip setpoint for IRM 13 is too high.
  - All other IRM trip setpoints are SAT.

Which one of the following describes the impact of these conditions on the startup, in accordance with Technical Specifications?

Control rod withdrawal may…

A. NOT begin until IRM 13 is fixed.

B. begin before IRM 13 is fixed. If IRM 13 is NOT fixed before SRMs reach three count rate doublings, then control rod withdrawal must be stopped.

C. begin before IRM 13 is fixed. If IRM 13 is NOT fixed within 6 hours, then control rod withdrawal must be stopped.

D. begin before IRM 13 is fixed. IRM 13 does NOT need to be fixed in order to complete the startup.

Proposed Answer: D

Explanation: N1-OP-43A, Plant Startup, requires verifying the requirements of Technical Specification Table 3.6.2.a for IRMs prior to commencing a startup. Technical Specification Table 3.6.2.a requires at least 3 operable IRMs in each trip system. With only IRM 13 inoperable, this requirement is met. Therefore, control rod withdrawal may begin and there is no requirement to fix IRM in order to complete the startup.

A. Incorrect – Control rod withdrawal may begin with IRM 13 inoperable. Plausible because this would be correct if another IRM in the same trip system was inoperable or if the requirement was for 4 operable IRMs per trip system.
B. Incorrect – IRM 13 does NOT need to be fixed in order to complete the startup. Plausible because IRMs are required to be operable for a startup and three count rate doublings is related to a control rod withdrawal requirement in N1-OP-43A.
C. Incorrect – IRM 13 does NOT need to be fixed in order to complete the startup. Plausible because IRMs are required to be operable for a startup and 6 hours is a related time requirement in Technical Specification 3.6.2 note (o).

Technical Reference(s): N1-OP-43A, Technical Specification 3.6.2

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-215000-RBO-14

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| | | |
|---|---|---|
| Examination Outline Cross-Reference: | Level | SRO |
| | Tier # | 2 |
| | Group # | 1 |
| | K/A # | 218000 2.2.19 |
| | Importance Rating | 3.4 |

**AUTOMATIC DEPRESSURIZATION SYSTEM**

**Knowledge of maintenance work order requirements**

Proposed Question: #89

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

- A failure has occurred in the ADS Channel 11 Inhibit keylock bypass switch.
- The failure is causing ADS Channel 11 to be bypassed.
- Maintenance is preparing to remove the switch and install a new one.
- Operations has determined a tagout will be placed to support the work.

Which one of the following describes:

- whether this job can be performed as “Tool Pouch Maintenance”, in accordance with MA-AA-716-003, Tool Pouch / Minor Maintenance,

  and

- whether a formal risk assessment needs to be performed, in accordance with OP-AA-107, Integrated Risk Management?

| | Can this job be performed as “Tool Pouch Maintenance”? | Is a formal risk assessment required to be performed? |
|---|---|---|
| A. | Yes | Yes |
| B. | Yes | No |
| C. | No | Yes |
| D. | No | No |

Proposed Answer: C

Explanation: This job cannot be performed as “Tool Pouch Maintenance” because it meets two of the criteria that exclude use of this classification: it is a job to restore Tech Spec equipment to an operable status and it is a power block activity which requires a clearance. This job does need a risk assessment to be performed, since it does not meet any of the exclusion criteria in OP-AA-107.

A. Incorrect – This job cannot be performed as “Tool Pouch Maintenance”. Plausible because it is replacement of a single component that can be accessed from the exterior of a panel. Also plausible because the component is already causing inoperability, such that the maintenance itself will not make the system become inoperable.
B. Incorrect – This job cannot be performed as “Tool Pouch Maintenance”. Plausible because it is replacement of a single component that can be accessed from the exterior of a panel. Also plausible because the component is already causing inoperability, such that the maintenance itself will not make the system become inoperable. A risk assessment must be performed. Plausible because the system is already inoperable due to the component malfunction and the work is isolated to a single system and component.
D. Incorrect – A risk assessment must be performed. Plausible because the system is already inoperable due to the component malfunction and the work is isolated to a single system and component.

Technical Reference(s): MA-AA-716-003, WC-AA-106, OP-AA-107

Proposed references to be provided to applicants during examination: None

Learning Objective: NS-OJTTPE-SRO-WCAA10

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(5)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | | |
|---|---|---|
| | Level | SRO |
| | Tier # | 2 |
| | Group # | 1 |
| | K/A # | 262002 2.2.36 |
| | Importance Rating | 4.2 |

**UNINTERRUPTABLE POWER SUPPLY (AC/DC)**

**Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operation**

Proposed Question: #90

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

| Time (hh:mm) | Event |
|---|---|
| 08:00 | • UPS 172A is supplying RPS Bus 12.<br>• UPS 172B is tagged out for maintenance. |
| 12:00 | • UPS 172A de-energizes due to a sustained electrical fault. |
| 12:15 | • RPS Bus 12 is re-energized from I&C Bus 130A. |

Which one of the following describes the status of Technical Specification 3.6.3, Emergency Power Sources, during these events?

Technical Specification 3.6.3 LCO entry is ***first*** required at time…

A. 08:00. After time 12:15, this LCO can be exited.

B. 08:00. After time 12:15, this LCO entry must remain in effect.

C. 12:00. After time 12:15, this LCO can be exited.

D. 12:00. After time 12:15, this LCO entry must remain in effect.

Proposed Answer: D

Explanation: When UPS 172B is tagged out, there is no need to enter an LCO in Technical Specification 3.6.3. When both UPS 172A and 172B become unavailable (at time 12:00) Technical Specification 3.6.3 LCO entry is required. Once the RPS Bus is transferred to I&C Bus 130A, TS LCO 3.6.3 entry is still required due to loss of emergency power source (N1-OP-40 P&L D.10.a).

A. Incorrect – When UPS 172B is tagged out, there is no need to enter an LCO in Technical Specification 3.6.3. Plausible because this does reduce the available power sources to the RPS bus. At time 12:15, TS LCO 3.6.3 entry is still required. Plausible because I&C Bus 130A re-energizes RPS Bus 12 and all functions are restored.
B. Incorrect – When UPS 172B is tagged out, there is no need to enter an LCO in Technical Specification 3.6.3. Plausible because this does reduce the available power sources to the RPS bus.
C. Incorrect – At time 12:15, TS LCO 3.6.3 entry is still required. Plausible because I&C Bus 130A re-energizes RPS Bus 12 and all functions are restored.

Technical Reference(s): N1-OP-40, Technical Specification 3.6.3

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-262002-RBO-14

Question Source: Modified Bank – 2017 Cert #89

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

**UPS (AC/DC)**

**Ability to explain and apply all system limits and precautions.**

Proposed Question: #89

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

- UPS 172A is supplying RPS Bus 12.
- Then, Annunciator A3-1-3, RPS UPS 172 TROUBLE, alarms.
- No half scram occurs.
- Computer point G184, UPS 172A SYSTEM ____TRBL, is received.
- An operator is dispatched to UPS 172A and reports the following local indications:
  - UPS 172A output voltage is 105 volts.
  - UPS 172A output frequency is 55 hertz.
- Then, RPS Bus 12 is manually transferred to I&C Bus 130A.

Which one of the following describes the Technical Specification impact for these events, if any?

A. No Technical Specification LCO entries are required.

B. LCO entry is required into Technical Specification 3.6.3, but NOT Technical Specification 3.6.12.

C. LCO entry is required into Technical Specification 3.6.12, but NOT Technical Specification 3.6.3.

D. LCO entry is required into both Technical Specifications 3.6.3 and 3.6.12.

Proposed Answer: D

Explanation: The UPS has low output voltage and frequency. The undervoltage and underfrequency protection for the RPS Bus has failed and this requires entry into TS LCO 3.6.12.b. Once the RPS Bus is transferred to I&C Bus 130A, TS LCO 3.6.3 entry is required due to loss of emergency power source (N1-OP-40 P&L D.10.a).

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | Level | SRO |
|---|---|---|
| | Tier # | 2 |
| | Group # | 2 |
| | K/A # | 272000 2.3.12 |
| | Importance Rating | 3.7 |

**Radiation Monitoring System**

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

Proposed Question: #91

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

- A malfunction during backwash of a Spent Fuel Pool Cooling filter is causing elevated radiation levels in the Reactor Building.
- Reactor Building Ventilation exhaust radiation monitors are indicating 25 mR/hr and slowly rising.
- The Reactor Building Continuous Air Monitor (CAM) is in alarm.
- The Stack radiation monitors are indicating 350 cps and slowly rising.
- Multiple Reactor Building Area Radiation Monitors (ARMs) are in alarm and indicate upscale.
- Radiation Protection reports the following general area survey results:
    - Reactor Building 261 east – 1.5 R/hr
    - Reactor Building 261 west – 0.8 R/hr
    - Reactor Building 281 east – 12.0 R/hr
    - Reactor Building 281 west – 9.5 R/hr

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

A. The Reactor may continue to operate at the current power level.

B. A Reactor shutdown is required, but a Reactor scram is NOT required.

C. A Reactor scram is required, but an RPV Blowdown is not required.

D. A Reactor scram and RPV Blowdown are required.

Proposed Answer: B

Explanation: N1-EOP-SC entry is required based on high Reactor Building ventilation exhaust radiation and high area radiation levels. The cause of the high radiation levels is a Spent Fuel Pool Cooling filter, therefore the discharge is from a non-primary system. This determines the flow path in N1-EOP-SC. Two general areas (RB 281 east and west) are above Maximum Safe Values in the same parameter (radiation levels above 8 R/hr). Therefore, N1-EOP-SC requires shutting down the Reactor per N1-OP-43C. This is the normal shutdown procedure.

A. Incorrect – A Reactor shutdown is required because two general areas are above the Maximum Safe Value for radiation (8 R/hr). Plausible because the source of the high radiation levels is not directly connected to the Reactor. Also plausible because if only one general area were above the Maximum Safe Value, continued Reactor operation would be allowed.
C. Incorrect – A Reactor shutdown is required, but a scram is not required because the source of the high radiation levels is a non-primary system. This would be the correct answer if the discharge was from a primary system and only one general area radiation level was above the Maximum Safe Value.
D. Incorrect – A Reactor shutdown is required, but a scram and RPV Blowdown is not required because the source of the high radiation levels is a non-primary system. This would be the correct answer if the discharge was from a primary system.

Technical Reference(s): N1-EOP-SC

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOPSCC01 EO-2

Question Source: Bank – 2017 Cert #83

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(4)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | | |
|---|---|---|
| | Level | SRO |
| | Tier # | 2 |
| | Group # | 2 |
| | K/A # | 256000 A2.01 |
| | Importance Rating | 3.9 |

**CONDENSATE SYSTEM**

**Ability to (a) predict the impacts of the following on the CONDENSATE SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Pump trips**

Proposed Question: #92

The plant is operating at 10% power during a startup with the following:

- Condensate pump 13 is operating.
- Condensate pumps 11 and 12 are in standby.

Then, Condensate pump 13 trips.

- Condensate pumps 11 and 12 automatically start.
- Reactor water level is 72” and stable.

Which one of the following describes the need for LCO entry in Technical Specification 3.1.8, High Pressure Coolant Injection?

LCO entry is…

A. required

B. NOT required because Reactor power is below 25%

C. NOT required because Condensate pump 13 is NOT a HPCI component

D. NOT required because any two Condensate pumps satisfy minimum operability requirements

Proposed Answer: A

Explanation: Condensate pump 13 is one of the two Condensate pumps associated with the HPCI mode of Condensate/Feedwater operation. With Reactor power at 10%, Reactor pressure is above 110 psig. Technical Specification 3.1.8 requires all HPCI-related pumps to be operable with Reactor pressure above 110 psig. Therefore, LCO 3.1.8.b entry is required.

B. Incorrect – LCO 3.1.8.b entry is required. Plausible because Technical Specification 3.1.8 allows keeping on Feedwater pump blocking valve closed up to 25% power.
C. Incorrect – LCO 3.1.8.b entry is required. Plausible because this would be correct for Condensate pump 12.
D. Incorrect – LCO 3.1.8.b entry is required. Plausible because only two Condensate pumps are required (but they are specific ones). Also plausible because all three Condensate pumps are capable of providing the necessary flow for the Feedwater pumps.

Technical Reference(s): N1-OP-16, Technical Specification 3.1.8

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-256000-RBO-14

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | | |
|---|---|---|
| | Level | SRO |
| | Tier # | 2 |
| | Group # | 2 |
| | K/A # | 290002 2.4.21 |
| | Importance Rating | 4.6 |

**REACTOR VESSEL INTERNALS**

**Knowledge of the parameters and logic used to assess the status of emergency operating procedures critical safety functions or shutdown critical safety functions**

Proposed Question: #93

The plant has experienced a loss of coolant accident from 100% power with the following:

- N1-EOP-RPV, RPV Control, and N1-EOP-PC, Primary Containment Control, are being implemented.
- N1-EOP-C2, RPV Blowdown, has been performed due to low Reactor water level.
- All available Reactor injection systems have been maximized.
- NO indications of core damage have been observed.
- Core Spray loop 11 flow is 250 x $10^4$ lbm/hr and stable.
- Core Spray loop 12 flow is 230 x $10^4$ lbm/hr and stable.
- Reactor pressure is 50 psig and stable.
- Reactor water level is -129” and slowly lowering.

Which one of the following describes the required procedure implementation?

A. Remain in N1-EOP-RPV and N1-EOP-PC.

B. Exit N1-EOP-RPV and N1-EOP-PC.
Enter N1-SAP-1, RPV and Primary Containment Injection.
Entry into N1-SAP-2, RPV, Containment, and Radioactivity Release Control, is NOT required.

C. Exit N1-EOP-RPV and N1-EOP-PC.
Enter N1-SAP-2, RPV, Containment, and Radioactivity Release Control.
Entry into N1-SAP-1, RPV and Primary Containment Injection, is NOT required.

D. Exit N1-EOP-RPV and N1-EOP-PC.
Enter N1-SAP-1, RPV and Primary Containment Injection, and N1-SAP-2, RPV, Containment, and Radioactivity Release Control.

Proposed Answer: A

Explanation: Reactor water level below -126” indicates that adequate core cooling is challenged, However, since both Core Spray loop flows are above 180 x $10^4$ lbm/hr, adequate core cooling from Spray Cooling exists. Therefore, SAP entry is NOT required and the crew must remain in N1-EOP-RPV and N1-EOP-PC.

B. Incorrect – SAP entry is not required. Plausible because adequate core cooling is challenged. Also plausible that SAPs would be entered individually based on specific plant conditions, as the EOPs are.
C. Incorrect – SAP entry is not required. Plausible because adequate core cooling is challenged. Also plausible that SAPs would be entered individually based on specific plant conditions, as the EOPs are.
D. Incorrect – SAP entry is not required. Plausible because this would be correct if either Core Spray loop flow was below 180 x $10^4$ lbm/hr.

Technical Reference(s): N1-EOP-RPV

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOPRPVC01 EO-2

Question Source: Modified Bank – 2023 NRC #83

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | | |
|---|---|---|
| | Level | SRO |
| | Tier # | 3 |
| | Group # | |
| | K/A # | 2.1.23 |
| | Importance Rating | 4.4 |

**Ability to perform general or normal operating procedures during any plant condition**

Proposed Question: #94

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

- A Technical Specification shutdown becomes required at time 12:00.
- The Technical Specification LCO states, “a normal orderly shutdown shall be initiated within one hour, and the reactor shall be in the cold shutdown condition within 10 hours.”

Which one of the following describes the required timing of the shutdown in order to meet the 10-hour time requirement, in accordance with N1-OP-43C, Plant Shutdown?

Insert a manual Reactor scram by approximately ___(1)___ and have Shutdown Cooling in service by approximately ___(2)___.

| | (1) | (2) |
|---|---|---|
| A. | 13:00 | 19:00 |
| B. | 13:00 | 20:30 |
| C. | 15:00 | 19:00 |
| D. | 15:00 | 20:30 |

Proposed Answer: C

Explanation: N1-OP-43C provides guidance on how to implement a 10-hour LCO shutdown in order to meet the time requirement. The procedure states, “When required to shutdown and cooldown to meet a 10-hour LCO requirement, it is necessary to perform a manual scram at approximately 3 hours into the LCO and have shutdown cooling in service approximately 7 hours into the LCO if cold shutdown is to be achieved in 10 hours.” Therefore, it is required to insert a manual Reactor scram by approximately 15:00 (12:00 + 3 hours) and to have Shutdown Cooling in service by approximately 19:00 (12:00 + 7 hours).

A. Incorrect – It is required to insert a manual Reactor scram by approximately 15:00. Plausible because 13:00 is when the shutdown is required to be initiated by. Also plausible because this would provide extra time to perform the cooldown and balance of plant shutdown.
B. Incorrect – It is required to insert a manual Reactor scram by approximately 15:00. Plausible because 13:00 is when the shutdown is required to be initiated by. Also plausible because this would provide extra time to perform the cooldown and balance of plant shutdown. It is required to have Shutdown Cooling in service by approximately 19:00. Plausible because 20:30 would provide 1.5 hours to complete the cooldown with Shutdown Cooling, which would allow enough time to keep the cooldown rate <100$^{o}$F/hr.
D. Incorrect – It is required to have Shutdown Cooling in service by approximately 19:00. Plausible because 20:30 would provide 1.5 hours to complete the cooldown with Shutdown Cooling, which would allow enough time to keep the cooldown rate <100$^{o}$F/hr.

Technical Reference(s): N1-OP-43C

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-OP-43C-CT-01

Question Source: Bank – 2009 Cert #95

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | | |
|---|---|---|
| | Level | SRO |
| | Tier # | 3 |
| | Group # | |
| | K/A # | 2.1.42 |
| | Importance Rating | 3.4 |

**Knowledge of new and spent fuel movement procedures (SRO Only)**

Proposed Question: #95

A refueling outage is in progress with the following:

- Core shuffle is in progress.
- A four cell section of the core is ***displayed on the following page***.

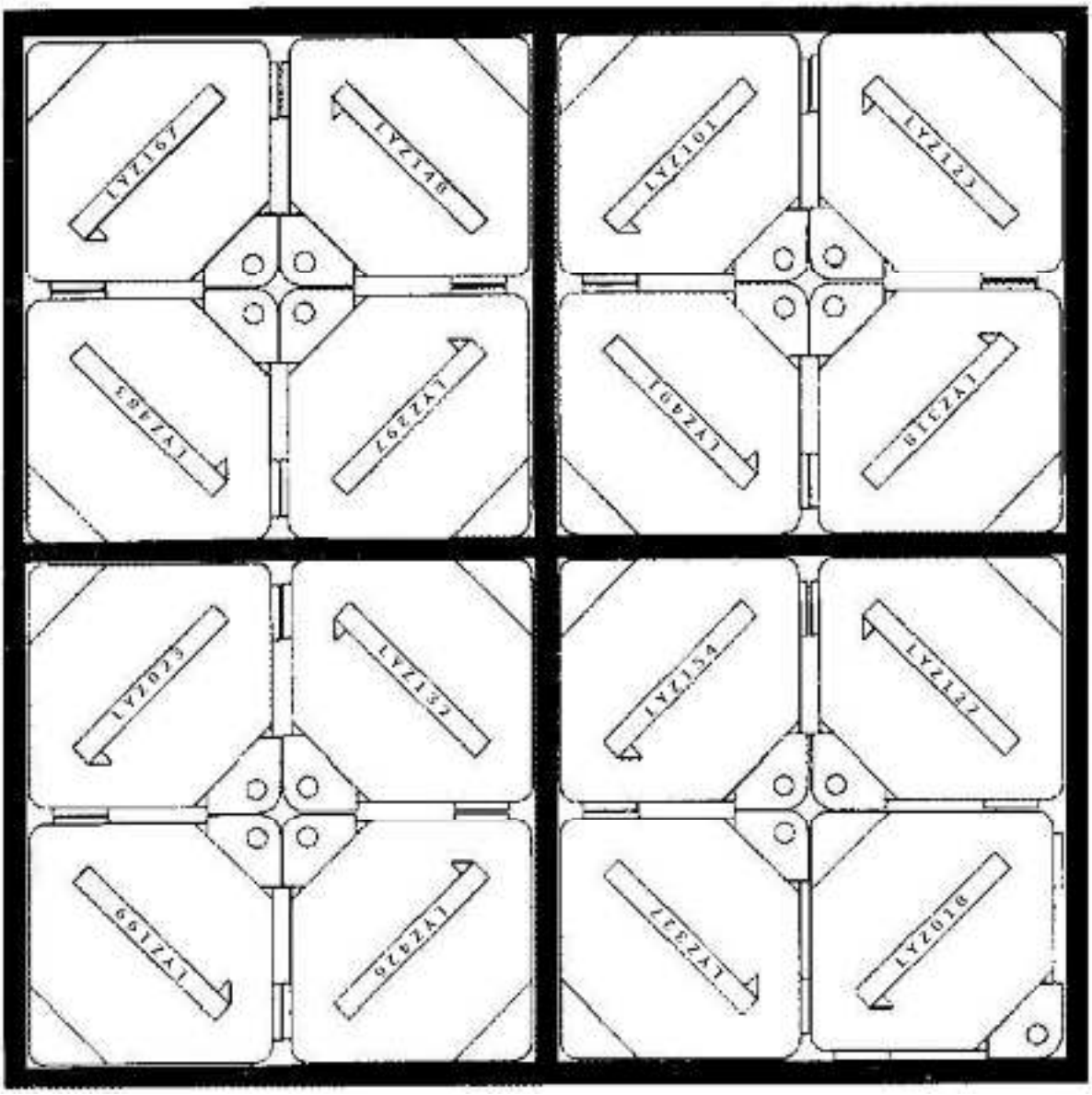
Which one of the following describes the status of this portion of the core and the required action(s), if any, in accordance with N1-FHP-25, General Description of Fuel Moves?

A. The fuel is loaded correctly. Core shuffle may continue with no additional required actions.

B. A discrepancy exists in the fuel loading. Core shuffle may continue without interruption as long as the discrepancy is fixed prior to core verification.

C. A discrepancy exists in the fuel loading. The discrepancy must be immediately fixed and then fuel movements may continue with permission from the Refuel Floor SRO.

D. A discrepancy exists in the fuel loading. Fuel movements must be immediately stopped. Completion of a formal evaluation is required before fuel movement may resume.


LYZ167
LYZ148
LYZ082
LYZ297
LYZ101
LYZ123
LYZ401
LYZ319
LYZ025
LYZ132
LYZ109
LYZ276
LYZ154
LYZ127
LYZ237
LYZ036

Proposed Answer: D

Explanation: The fuel assembly in the lower right corner is oriented incorrectly, as evidenced by the bail handle indicator pointing away from the center of the fuel cell, the position of the channel fastener, and the orientation of the text on the handle. N1-FHP-25 defines this as a Fuel Movement Discrepancy (FMD). The procedure requires fuel movement to be stopped and requires completion of a formal evaluation prior to resuming fuel movements.

A. Incorrect – There is a discrepancy in the fuel loading. Plausible because 15 of the 16 assemblies are loaded correctly and the overall shape is symmetrical.
B. Incorrect – Core shuffle cannot continue with the given error. Plausible because no significant adverse consequence would be expected from this fuel movement discrepancy until Reactor operation recommenced.
C. Incorrect – The discrepancy cannot be immediately fixed without a formal evaluation. Plausible because the issue would be fairly simple to fix.

Technical Reference(s): N1-FHP-25

Proposed references to be provided to applicants during examination: Embedded graphic

Learning Objective: S-101002-RBO-9

Question Source: Bank – NMP2 2021 NRC #95

Question History: 2013 NRC #80

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(6)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | | |
|---|---|---|
| | Level | SRO |
| | Tier # | 3 |
| | Group # | |
| | K/A # | 2.2.6 |
| | Importance Rating | 3.6 |

**Knowledge of the process for making changes to procedures**

Proposed Question: #96

A Temporary Change is being processed for an operating procedure per AD-AA-101, Processing of Procedures and T&RMs.

Which one of the following describes:

(1) the individual(s) in Operations that are allowed to approve the Temporary Change for initial use,

and

(2) when full review, approval, and authorization of the Temporary Change must be completed by,

in accordance with AD-AA-101?

| | (1) | (2) |
|---|---|---|
| A. | Any qualified SRO | 14 days |
| B. | Any qualified SRO | 31 days |
| C. | Operations Director, only | 14 days |
| D. | Operations Director, only | 31 days |

Proposed Answer: A

Explanation: AD-AA-101 allows any qualified SRO to perform the initial approval of a Temporary Change. The full review, approval, and authorization must be completed within 14 days.

B. Incorrect – 14 days are allowed for full review, approval, and authorization. Plausible because 31 days is a common completion time and of similar magnitude as 14 days.
C. Incorrect – Any qualified SRO may perform the initial approval. Plausible because many processes require Operations Director approval.
D. Incorrect – Any qualified SRO may perform the initial approval. Plausible because many processes require Operations Director approval. 14 days are allowed for full review, approval, and authorization. Plausible because 31 days is a common completion time and of similar magnitude as 14 days.

Technical Reference(s): AD-AA-101, AD-AA-101-F-10

Proposed references to be provided to applicants during examination: None

Learning Objective: NS-OJTTPE-SRO-ADAA10

Question Source: Bank – JAF 20-1 NRC #96

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(3)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | Level | SRO |
|---|---|---|
| | Tier # | 3 |
| | Group # | |
| | K/A # | 2.2.37 |
| | Importance Rating | 4.6 |

**Ability to determine operability or availability of safety-related equipment (SRO Only)**

Proposed Question: #97

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

- It is discovered that a surveillance has NOT been completed on time for a Technical Specification required system.
- The surveillance frequency is quarterly (92 days).
- The surveillance was last performed 120 days ago.
- A risk evaluation has been performed and the associated impact is being managed.

Which one of the following describes the requirement for this missed surveillance, in accordance with Technical Specifications?

A. The associated LCO must be declared NOT met at this time.

B. Complete the surveillance within a maximum of 24 hours from the time of discovery or then the associated LCO must be declared NOT met.

C. Complete the surveillance within a maximum of 23 days from the time of discovery or then the associated LCO must be declared NOT met.

D. Complete the surveillance within a maximum of 92 days from the time of discovery or then the associated LCO must be declared NOT met.

Proposed Answer: D

Explanation: Surveillance Requirement (SR) Applicability 4.0.3 applies given discovery of a missed surveillance after the required frequency has elapsed. The requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified frequency (in this case, 92 days), whichever is greater. Therefore, up to 92 days are allowed to perform the missed surveillance before being required to declare the LCO not met.

A. Incorrect – SR 4.0.3 allows a delay time to perform the missed surveillance before being required to declare the LCO not met. Plausible because the surveillance is currently over the 125% limit of SR 4.0.2.
B. Incorrect – SR 4.0.3 allows the longer of 24 hours or the specified frequency (92 days). Plausible because in some cases 24 hours would be the limit for a missed surveillance.
C. Incorrect – SR 4.0.3 allows the longer of 24 hours or the specified frequency (92 days). 23 days is based on the grace period (25% of 92 days) allowed by SR 4.0.2.

Technical Reference(s): Technical Specification Surveillance Requirement 4.0.3

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-TECSPEC01-EO-6.0

Question Source: Bank – NMP2 4/23 NRC #97

Question History: NMP2 4/23 NRC #97

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | Level | SRO |
|---|---|---|
| | Tier # | 3 |
| | Group # | |
| | K/A # | 2.3.6 |
| | Importance Rating | 3.8 |

**Ability to approve liquid or gaseous release permits**

Proposed Question: #98

A loss of coolant accident has resulted in the following:

- An RPV Blowdown has been performed due to exceeding the Pressure Suppression Pressure.
- Reactor pressure is 75 psig and stable.
- Reactor water level is being controlled 53” to 95” using Condensate.
- Core Spray injection has been secured by closing the inside isolation valves.
- Drywell pressure is 45 psig and slowly rising.
- Torus pressure is 42 psig and slowly rising.
- Torus water level is 17 feet and slowly rising.
- Chemistry reports that venting of the Primary Containment will exceed release rate limits.

**Note:** The Primary Containment Pressure Limit (PCPL) curve is provided on the next page.

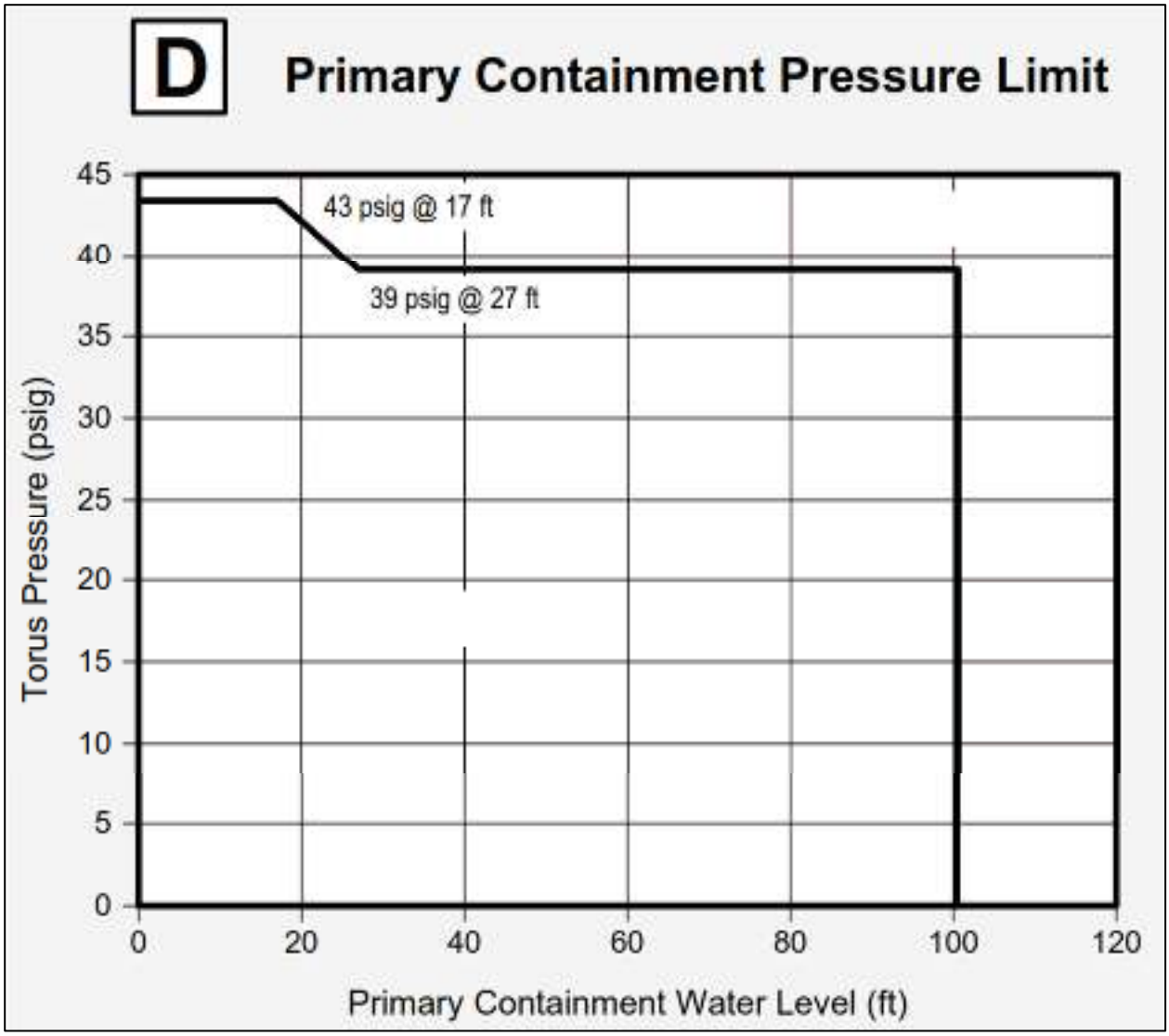
Which one of the following describes the required control of Condensate injection and Primary Containment venting, in accordance with the Emergency Operating Procedures?

A. Continue injection with Condensate. Direct Primary Containment venting.

B. Continue injection with Condensate. Do NOT direct Primary Containment venting.

C. Transition injection from Condensate to another system. Direct Primary Containment venting.

D. Transition injection from Condensate to another system. Do NOT direct Primary Containment venting.


D
Primary Containment Pressure Limit
43 psig @ 17 ft
39 psig @ 27 ft
Torus Pressure (psig)
45
40
35
30
25
20
15
10
5
0
0
20
40
60
80
100
120
Primary Containment Water Level (ft)

Proposed Answer: C

Explanation: The given values of Torus pressure and Torus water level show that the Primary Containment Pressure Limit (PCPL) is about to be exceeded. N1-EOP-RPV requires stopping injection from sources outside the Primary Containment not needed for core cooling if you cannot stay below PCPL. Reactor water level is well above top of active fuel, Core Spray is available for injection, and Condensate injects water from outside the Primary Containment. Therefore, injection must be transitioned from Condensate to another system, such as Core Spray. N1-EOP-PC requires venting the Primary Containment even if release rate limits will be exceeded.

**Note:** This question meets the K/A by testing the operator's knowledge of when to approve a release through Primary Containment venting in excess of normal release rate limits.

A. Incorrect – Condensate injection must be terminated because it injects from outside the Primary Containment and PCPL is being challenged. Plausible because PCPL has not yet been exceeded. Also plausible because this would be correct if other injection systems were not available and/or adequate core cooling were challenged.
B. Incorrect – Condensate injection must be terminated because it injects from outside the Primary Containment and PCPL is being challenged. Plausible because PCPL has not yet been exceeded. Also plausible because this would be correct if other injection systems were not available and/or adequate core cooling were challenged. Primary Containment venting is required. Plausible because PCPL has not yet been exceeded. Also plausible because release rates will be exceeded.
D. Incorrect – Primary Containment venting is required. Plausible because PCPL has not yet been exceeded. Also plausible because release rates will be exceeded.

Technical Reference(s): N1-EOP-RPV, N1-EOP-PC

Proposed references to be provided to applicants during examination: Embedded graphic

Learning Objective: 1101-EOPPCC01 EO-2

Question Source: Bank – 2015 Cert #100

Question History: JAF 17-2 NRC #99

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(4)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | Level | SRO |
|---|---|---|
| | Tier # | 3 |
| | Group # | |
| | K/A # | 2.4.29 |
| | Importance Rating | 4.4 |

**Knowledge of the emergency plan implementing procedures**

Proposed Question: #99

An emergency is in progress with the following:

- An Alert has been declared.
- A condition requires a worker to enter a high radiation area.
- The required dose to perform the job is 17,000 mRem.

Which one of the following describes the ability to authorize this exposure, in accordance with EP-AA-113, Personnel Protective Actions?

This exposure…

A. can be authorized for either protecting valuable plant equipment or life-saving activities.

B. can be authorized for life-saving activities, but NOT for protecting valuable plant equipment.

C. CANNOT be authorized unless the emergency is upgraded to a higher declaration.

D. CANNOT be authorized for either protecting valuable plant equipment or life-saving activities regardless of the emergency declaration level.

Proposed Answer: B

Explanation: EP-AA-113 allows emergency dose authorization of up to 10 Rem for protecting valuable plant equipment. A dose authorization of 25 Rem is allowed for life-saving activities. Therefore, the given job that requires 17 Rem can be authorized for life-saving activities, but not for protecting valuable plant equipment.

A. Incorrect – The given dose is too high for protecting valuable plant equipment. Plausible because an emergency dose up to 10 Rem could be authorized for this purpose.
C. Incorrect – Emergency dose authorization is allowed at the Alert level. Plausible because this is the 2nd lowest declaration level and some E-plan actions do not happen until higher levels (eg. PARs, KI).
D. Incorrect – The given dose can be authorized for life-saving activities. Plausible because this is above the max dose for other activities and is well above the normal federal limit.

Technical Reference(s): EP-AA-113

Proposed references to be provided to applicants during examination: None

Learning Objective: S100-EPL001C07

Question Source: Bank – NMP2 11/23 NRC #100

Question History: JAF 17-2 NRC #98

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(4)

Comments:

**Form 4.2-1 Written Examination Question Worksheet**

| Examination Outline Cross-Reference: | | |
|---|---|---|
| | Level | SRO |
| | Tier # | 3 |
| | Group # | |
| | K/A # | 2.4.51 |
| | Importance Rating | 4.0 |

**Knowledge of emergency operating procedure exit conditions (e.g., emergency condition no longer exists or severe accident guideline entry is required)**

Proposed Question: #100

The plant has experienced a loss of coolant accident and N1-EOP-RPV, RPV Control, is being executed.

Which one of the following indications would require exiting N1-EOP-RPV, in accordance with N1-EOP-RPV?

A. Drywell radiation at 400 R/hr

B. Reactor coolant activity at 500 µCi/gm

C. Drywell hydrogen concentration at 1.2%

D. Field survey at the site boundary at 200 mR/hr

Proposed Answer: B

Explanation: EOP Detail V, Core Damage Indications, defines the parameters and values used to determine if core damage is occurring in N1-EOP-RPV. If core damage is occurring, N1-EOP-RPV must be exited. Of the given indications, Reactor coolant activity at 500 µCi/gm is the one that exceeds a value in Detail V and requires exiting N1-EOP-RPV.

A. Incorrect – This does not require exiting N1-EOP-RPV. Plausible because Drywell radiation ≥1100 R/hr would require exiting N1-EOP-RPV.
C. Incorrect – This does not require exiting N1-EOP-RPV. Plausible because Drywell hydrogen concentration ≥4% would require exiting N1-EOP-RPV.
D. Incorrect – This does not require exiting N1-EOP-RPV. Plausible because this does indicate significant release of radioactivity from the plant and is above the value for declaration of a Site Area Emergency.

Technical Reference(s): N1-EOP-RPV

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOPRPVC01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments: