Facility: SHEARON HARRIS Date of Exam: SEPTEMBER 2013

| Tier | Group | RO K/A Category Points K1 | K2 | K3 | K4 | K5 | K6 | A1 | A2 | A3 | A4 | G* | Total | SRO-Only Points A2 | G* | Total |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 1. Emergency & Abnormal Plant Evolutions | 1 | 3 | 3 | 3 | N/A | | | 3 | 3 | N/A | | 3 | 18 | 3 | 3 | 6 |
| | 2 | 1 | 1 | 2 | | | | 2 | 2 | | | 1 | 9 | 2 | 2 | 4 |
| | Tier Totals | 4 | 4 | 5 | | | | 5 | 5 | | | 4 | 27 | 5 | 5 | 10 |
| 2. Plant Systems | 1 | 2 | 2 | 3 | 3 | 2 | 3 | 3 | 3 | 2 | 3 | 2 | 28 | 3 | 2 | 5 |
| | 2 | 1 | 1 | 1 | 1 | 1 | 1 | 0 | 1 | 1 | 1 | 1 | 10 | 0 \| 2 | 1 | 3 |
| | Tier Totals | 3 | 3 | 4 | 4 | 3 | 4 | 3 | 4 | 3 | 4 | 3 | 38 | 5 | 3 | 8 |

| 3. Generic Knowledge and Abilities Categories | RO 1 | RO 2 | RO 3 | RO 4 | RO Total | SRO 1 | SRO 2 | SRO 3 | SRO 4 | SRO Total |
|---|---|---|---|---|---|---|---|---|---|---|
| | 3 | 2 | 2 | 3 | 10 | 1 | 2 | 2 | 2 | 7 |

Note:

1. Ensure that at least two topics from every applicable K/A category are sampled within each tier of the RO and SRO-only outlines (i.e., except for one category in Tier 3 of the SRO-only outline, the "Tier Totals" in each K/A category shall not be less than two).
2. The point total for each group and tier in the proposed outline must match that specified in the table. The final point total for each group and tier may deviate by ±1 from that specified in the table based on NRC revisions. The final RO exam must total 75 points and the SRO-only exam must total 25 points.
3. Systems/evolutions within each group are identified on the associated outline; systems or evolutions that do not apply at the facility should be deleted and justified; operationally important, site-specific systems/evolutions that are not included on the outline should be added. Refer to Section D.1.b of ES-401 for guidance regarding the elimination of inappropriate K/A statements.
4. Select topics from as many systems and evolutions as possible; sample every system or evolution in the group before selecting a second topic for any system or evolution.
5. Absent a plant-specific priority, only those K/As having an importance rating (IR) of 2.5 or higher shall be selected. Use the RO and SRO ratings for the RO and SRO-only portions, respectively.
6. Select SRO topics for Tiers 1 and 2 from the shaded systems and K/A categories.
7.* The generic (G) K/As in Tiers 1 and 2 shall be selected from Section 2 of the K/A Catalog, but the topics must be relevant to the applicable evolution or system. Refer to Section D.1.b of ES-401 for the applicable K/As.
8. On the following pages, enter the K/A numbers, a brief description of each topic, the topics' importance ratings (IRs) for the applicable license level, and the point totals (#) for each system and category. Enter the group and tier totals for each category in the table above; if fuel handling equipment is sampled in other than Category A2 or G* on the SRO-only exam, enter it on the left side of Column A2 for Tier 2, Group 2 (Note #1 does not apply). Use duplicate pages for RO and SRO-only exams.
9. For Tier 3, select topics from Section 2 of the K/A catalog, and enter the K/A numbers, descriptions, IRs, and point totals (#) on Form ES-401-3. Limit SRO selections to K/As that are linked to 10 CFR 55.43.

| ES-401 | | | | | | | PWR Examination Outline<br>Emergency and Abnormal Plant Evolutions - Tier 1/Group 1 (RO / SRO) | | Form ES-401-2 |
|---|---|---|---|---|---|---|---|---|---|
| E/APE # / Name / Safety Function | K 1 | K 2 | K 3 | A 1 | A 2 | G | K/A Topic(s) | IR | # |
| 000007 (BW/E02&E10; CE/E02) Reactor Trip - Stabilization - Recovery / 1 | | R | | | | | R 007 EK2.02 | | |
| 000008 Pressurizer Vapor Space Accident / 3 | | R | | | | | R 008 AK2.02 | | |
| 000009 Small Break LOCA / 3 | | R | | | | | R 009 K2.03 | | |
| 000011 Large Break LOCA / 3 | | | | | | R | R 011 EG2.4.47 | | |
| 000015/17 RCP Malfunctions / 4 | R | | | | | | R 015 K1.02 | | |
| 000022 Loss of Rx Coolant Makeup / 2 | | | | R | | | R 022AA1.08 | | |
| 000025 Loss of RHR System / 4 | | | | | S | | S 025 AA2.05 | | |
| 000026 Loss of Component Cooling Water / 8 | | | | R | | | R 026 AA1.05 | | |
| 000027 Pressurizer Pressure Control System Malfunction / 3 | | | | | | | | | |
| 000029 ATWS / 1 | R | | | | S | | R 029EK1.01 S 029 EA2.09 | | |
| 000038 Steam Gen. Tube Rupture / 3 | | | | | | S | S 038 EG2.4.30 | | |
| 000040 (BW/E05; CE/E05; W/E12) Steam Line Rupture - Excessive Heat Transfer / 4 | | | R | | | | R 040 AK3.03 | | |
| 000054 (CE/E06) Loss of Main Feedwater / 4 | | | | | R | S | R 054 AA2.02<br>S 054AG2.4.47 | | |
| 000055 Station Blackout / 6 | | | | | S | | S 055 EA2.04 | | |
| 000056 Loss of Off-site Power / 6 | | | | | | R | R 056AG2.4.45 | | |
| 000057 Loss of Vital AC Inst. Bus / 6 | | | | | | | | | |
| 000058 Loss of DC Power / 6 | | | | | R | S | R 058 AA2.01 S 058 AG2.4.3 | | |
| 000062 Loss of Nuclear Svc Water / 4 | | | | | R | | R 062 AA2.01 | | |
| 000065 Loss of Instrument Air / 8 | | | R | | | | R 065 AK3.08 | | |
| W/E04 LOCA Outside Containment / 3 | | | R | | | | R WE04 EK3.2 | | |
| W/E11 Loss of Emergency Coolant Recirc. / 4 | | | | R | | | R WE11 EA1.3 | | |
| BW/E04; W/E05 Inadequate Heat Transfer - Loss of Secondary Heat Sink / 4 | R | | | | | | R WE05 EK1.2 | | |
| 000077 Generator Voltage and Electric Grid Disturbances / 6 | | | | | | R | R 077 AG2.4.4 | | |
| | | | | | | | | | |
| K/A Category Totals: | 3 | 3 | 3 | 3 | 3 | 3 | Group Point Total: | | 18/6 |
| | | | | | 3 | 3 | | | |

| KA | NAME / SAFETY FUNCTION: | IR RO | IR SRO | K1 | K2 | K3 | K4 | K5 | K6 | A1 | A2 | A3 | A4 | G | TOPIC: |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 007EK2.02 | Reactor Trip - Stabilization - Recovery / 1 | 2.6 | 2.8 | ☐ | ☑ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | Breakers, relays and disconnects |
| 008AK2.02 | Pressurizer Vapor Space Accident / 3 | 2.7 | 2.7 | ☐ | ☑ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | Sensors and detectors |
| 009EK2.03 | Small Break LOCA / 3 | 3 | 3.3 | ☐ | ☑ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | S/Gs |
| 011EG2.4.47 | Large Break LOCA / 3 | 4.2 | 4.2 | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☑ | Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material. |
| 015AK1.02 | RCP Malfunctions / 4 | 3.7 | 4.1 | ☑ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | Consequences of an RCPS failure |
| 022AA1.08 | Loss of Rx Coolant Makeup / 2 | 3.4 | 3.3 | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☑ | ☐ | ☐ | ☐ | ☐ | VCT level |
| 026AA1.05 | Loss of Component Cooling Water / 8 | 3.1 | 3.1 | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☑ | ☐ | ☐ | ☐ | ☐ | The CCWS surge tank, including level control and level alarms and radiation alarm |
| 029EK1.01 | ATWS / 1 | 2.8 | 3.1 | ☑ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | Reactor nucleonics and thermo-hydraulics behavior<br>Please make sure not solely testing GFES; when you write -- must test plant specific knowledge |
| 040AK3.03 | Steam Line Rupture - Excessive Heat Transfer / 4 | 3.2 | 3.5 | ☐ | ☐ | ☑ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | Steam line non-return valves |
| 054AA2.02 | Loss of Main Feedwater / 4 | 4.1 | 4.4 | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☑ | ☐ | ☐ | ☐ | Differentiation between loss of all MFW and trip of one MFW pump |
| 056AG2.4.45 | Loss of Off-site Power / 6 | 4.1 | 4.3 | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☑ | Ability to prioritize and interpret the significance of each annunciator or alarm. |

| KA | NAME / SAFETY FUNCTION: | IR RO | IR SRO | K1 | K2 | K3 | K4 | K5 | K6 | A1 | A2 | A3 | A4 | G | TOPIC: |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 058AA2.01 | Loss of DC Power / 6 | 3.7 | 4.1 | | | | | | | | ✔ | | | | That a loss of dc power has occurred; verification that substitute power sources have come on line |
| 062AA2.01 | Loss of Nuclear Svc Water / 4 | 2.9 | 3.5 | | | | | | | | ✔ | | | | Location of a leak in the SWS |
| 065AK3.08 | Loss of Instrument Air / 8 | 3.7 | 3.9 | | | ✔ | | | | | | | | | Actions contained in EOP for loss of instrument air |
| 077AG2.4.4 | Generator Voltage and Electric Grid Disturbances / 6 | 4.5 | 4.7 | | | | | | | | | | | ✔ | Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures. |
| WE04EK3.2 | LOCA Outside Containment / 3 | 3.4 | 4.0 | | | ✔ | | | | | | | | | Normal, abnormal and emergency operating procedures associated with (LOCA Outside Containment). |
| WE05EK1.2 | Inadequate Heat Transfer - Loss of Secondary Heat Sink / 4 | 3.9 | 4.5 | ✔ | | | | | | | | | | | Normal, abnormal and emergency operating procedures associated with (Loss of Secondary Heat Sink). |
| WE11EA1.3 | Loss of Emergency Coolant Recirc. / 4 | 3.7 | 4.2 | | | | | | | ✔ | | | | | Desired operating results during abnormal and emergency situations. |

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## SRO T1G1 PWR EXAMINATION OUTLINE

FORM ES-401-2

| KA | NAME / SAFETY FUNCTION: | IR RO | IR SRO | K1 | K2 | K3 | K4 | K5 | K6 | A1 | A2 | A3 | A4 | G | TOPIC: |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 025AA2.05 | Loss of RHR System / 4 | 3.1 | 3.5 | | | | | | | | ✓ | | | | Limitations on LPI flow and temperature rates of change |
| 029EA2.09 | ATWS / 1 | 4.4 | 4.5 | | | | | | | | ✓ | | | | Occurrence of a main turbine/reactor trip |
| 038EG2.4.30 | Steam Gen. Tube Rupture / 3 | 2.7 | 4.1 | | | | | | | | | | | ✓ | Knowledge of events related to system operations/status that must be reported to internal orginizations or outside agencies. |
| 054AG2.4.47 | Loss of Main Feedwater / 4 | 4.2 | 4.2 | | | | | | | | | | | ✓ | Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material. |
| 055EA2.04 | Station Blackout / 6 | 3.7 | 4.1 | | | | | | | | ✓ | | | | Instruments and controls operable with only dc battery power available |
| 058AG2.4.3 | Loss of DC Power / 6 | 3.7 | 3.9 | | | | | | | | | | | ✓ | Ability to identify post-accident instrumentation. |

ES-401 PWR Examination Outline Form ES-401-2

Emergency and Abnormal Plant Evolutions - Tier 1/Group 2 RO / SRO

| E/APE # / Name / Safety Function | K 1 | K 2 | K 3 | A 1 | A 2 | G | K/A Topic(s) | IR | # |
|---|---|---|---|---|---|---|---|---|---|
| 000001 Continuous Rod Withdrawal / 1 | | | | | R | | R001 AA2.01 | | |
| 000003 Dropped Control Rod / 1 | | | | | | | | | |
| 000005 Inoperable/Stuck Control Rod / 1 | | | | | S | | S 005 AA2.03 | | |
| 000024 Emergency Boration / 1 | | | | | | | | | |
| 000028 Pressurizer Level Malfunction / 2 | | | | | | | | | |
| 000032 Loss of Source Range NI / 7 | | | | | | | | | |
| 000033 Loss of Intermediate Range NI / 7 | | | | | | | | | |
| 000036 (BW/A08) Fuel Handling Accident / 8 | | R | | | | | R 036 AK2.01 | | |
| 000037 Steam Generator Tube Leak / 3 | | | | | | R | R 037 AG2.4.34 | | |
| 000051 Loss of Condenser Vacuum / 4 | | | | R | | | R 051 AA1.04 | | |
| 000059 Accidental Liquid RadWaste Rel. / 9 | | | R | | | | R 059 AK3.04 | | |
| 000060 Accidental Gaseous Radwaste Rel. / 9 | | | | | | S | S 060AG2.2.37 | | |
| 000061 ARM System Alarms / 7 | | | | | S | | S 061 AA2.02 | | |
| 000067 Plant Fire On-site / 8 | | | | | | | | | |
| 000068 (BW/A06) Control Room Evac. / 8 | | | | | | | | | |
| 000069 (W/E14) Loss of CTMT Integrity / 5 | | | | R | | S | R WE14 EA1.1 S 069 AG2.2.25 | | |
| 000074 (W/E06&E07) Inad. Core Cooling / 4 | | | | | R | | R 074 EA2.03 | | |
| 000076 High Reactor Coolant Activity / 9 | | | | | | | | | |
| W/EO1 & E02 Rediagnosis & SI Termination / 3 | | | | | | | | | |
| W/E13 Steam Generator Over-pressure / 4 | | | | | | | | | |
| W/E15 Containment Flooding / 5 | | | R | | | | R WE15 EK3.1 | | |
| W/E16 High Containment Radiation / 9 | | | | | | | | | |
| BW/A01 Plant Runback / 1 | | | | | | | | | |
| BW/A02&A03 Loss of NNI-X/Y / 7 | | | | | | | | | |
| BW/A04 Turbine Trip / 4 | | | | | | | | | |
| BW/A05 Emergency Diesel Actuation / 6 | | | | | | | | | |
| BW/A07 Flooding / 8 | | | | | | | | | |
| BW/E03 Inadequate Subcooling Margin / 4 | | | | | | | | | |
| BW/E08; W/E03 LOCA Cooldown - Depress. / 4 | R | | | | | | R WE03 EK1.1 | | |
| BW/E09; CE/A13; W/E09&E10 Natural Circ. / 4 | | | | | | | | | |
| BW/E13&E14 EOP Rules and Enclosures | | | | | | | | | |
| CE/A11; W/E08 RCS Overcooling - PTS / 4 | | | | | | | | | |
| CE/A16 Excess RCS Leakage / 2 | | | | | | | | | |
| CE/E09 Functional Recovery | | | | | | | | | |
| K/A Category Point Totals: | 1 | 1 | 2 | 2 | 2 | 1 | Group Point Total: | | 9/4 |
| | | | | | 2 | 2 | | | |

RO



| KA | NAME / SAFETY FUNCTION: | IR RO | IR SRO | K1 | K2 | K3 | K4 | K5 | K6 | A1 | A2 | A3 | A4 | G | TOPIC: |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 001AA2.01 | Continuous Rod Withdrawal / 1 | 4.2 | 4.2 | | | | | | | | ✓ | | | | Reactor tripped breaker indicator |
| 036AK2.01 | Fuel Handling Accident / 8 | 2.9 | 3.5 | | ✓ | | | | | | | | | | Fuel handling equipment |
| 037AG2.4.34 | Steam Generator Tube Leak / 3 | 4.2 | 4.1 | | | | | | | | | | | ✓ | Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects |
| 051AA1.04 | Loss of Condenser Vacuum / 4 | 2.5 | 2.5 | | | | | | | ✓ | | | | | Rod position |
| 059AK3.04 | Accidental Liquid RadWaste Rel. / 9 | 3.8 | 4.3 | | | ✓ | | | | | | | | | Actions contained in EOP for accidental liquid radioactive-waste release |
| 074EA2.03 | Inad. Core Cooling / 4 | 3.8 | 4.1 | | | | | | | | ✓ | | | | Availability of turbine bypass valves for cooldown |
| WE03EK1.1 | LOCA Cooldown - Depress. / 4 | 3.4 | 4.0 | ✓ | | | | | | | | | | | Components, capacity, and function of emergency systems. |
| WE14EA1.1 | Loss of CTMT Integrity / 5 | 3.7 | 3.7 | | | | | | | ✓ | | | | | Components and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes and automatic and manual features. |
| WE15EK3.1 | Containment Flooding / 5 | 2.7 | 2.9 | | | ✓ | | | | | | | | | Facility operating characteristics during transient conditions, including coolant chemistry and the effects of temperature, pressure and reactivity changes and operating limitations and reasons for these operating characteristics. |

| KA | NAME / SAFETY FUNCTION: | IR RO | IR SRO | K1 | K2 | K3 | K4 | K5 | K6 | A1 | A2 | A3 | A4 | G | TOPIC: |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 005AA2.03 | Inoperable/Stuck Control Rod / 1 | 3.5 | 4.4 | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☑ | ☐ | ☐ | ☐ | Required actions if more than one rod is stuck or inoperable |
| 060AG2.2.37 | Accidental Gaseous Radwaste Rel. / 9 | 3.6 | 4.6 | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☑ | Ability to determine operability and/or availability of safety related equipment |
| 061AA2.02 | ARM System Alarms / 7 | 2.9 | 3.2 | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☑ | ☐ | ☐ | ☐ | Normal radiation intensity for each ARM system channel |
| 069AG2.2.25 | Loss of CTMT Integrity / 5 | 3.2 | 4.2 | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☑ | Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits. |

ES-401 PWR Examination Outline Form ES-401-2
Plant Systems - Tier 2/Group 1 (RO) / (SRO)

| System # / Name | K 1 | K 2 | K 3 | K 4 | K 5 | K 6 | A 1 | A 2 | A 3 | A 4 | G | K/A Topic(s) | IR | # |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 003 Reactor Coolant Pump | | | | | | | | R | | | | R 003 A2.02 | | |
| 004 Chemical and Volume Control | | | | | R | | | R | | | | R 004 A2.13<br>R 004 K5.26 | | |
| 005 Residual Heat Removal | | | | | | | R | | | | | R 005 A1.05 | | |
| 006 Emergency Core Cooling | | | | | | | | | | | R | R 006 G2.1.30 | | |
| 007 Pressurizer Relief/Quench Tank | | | | R | | | | | | | R | R 007 G2.1.20<br>R 007 K4.01 | | |
| 008 Component Cooling Water | | | | | | | | | R | | | R 008 A3.08 | | |
| 010 Pressurizer Pressure Control | | | | | | R R | | | | | | R 010 K6.01, R 010 K6.03 | | |
| 012 Reactor Protection | | | | R | | | | | | | | R 012 K4.02 | | |
| 013 Engineered Safety Features Actuation | | R | | | | | | | | R | | R 013 A4.01<br>R 013 K2.01 | | |
| 022 Containment Cooling | | | | | | | | | | R | | R 022 A4.03 | | |
| 025 Ice Condenser | N/A | | | | | | | | | | | | | |
| 026 Containment Spray | | | R | | | | R | | | | | R 026 A1.04<br>R 026 K3.02 | | |
| 039 Main and Reheat Steam | | | | | | | | R S | | | | R 039 A2.01 S 039 A2.03 | | |
| 059 Main Feedwater | | | | R | | | | | | | | R 059 K4.02 | | |
| 061 Auxiliary/Emergency Feedwater | | | | | R | | R | | | | | R 061 K5.05<br>R 061 A1.01 | | |
| 062 AC Electrical Distribution | | R | | | | | | S | | | | R 062 K2.01 S 062 A2.11 | | |
| 063 DC Electrical Distribution | | | R | | | | | | R | | S | R 063 A3.01 S 063 G2.2.40<br>R 063 K3.02 | | |
| 064 Emergency Diesel Generator | | | | | | R | | | | | | R 064 K6.08 | | |
| 073 Process Radiation Monitoring | | | | | | | | | | R | | R 073 A4.02 | | |
| 076 Service Water | R | | | | | | | S | | | | R 076 K1.01 S 076 A2.01 | | |
| 078 Instrument Air | | | R | | | | | | | | | R 078 K3.01 | | |
| 103 Containment | R | | | | | | | | | | S | R 103 K1.03 S 103 G2.2.22 | | |
| | | | | | | | | | | | | | | |
| | | | | | | | | | | | | | | |
| | | | | | | | | | | | | | | |
| K/A Category Point Totals: | 2 | 2 | 3 | 3 | 2 | 3 | 3 | 3 | 2 | 3 | 2 | Group Point Total: | | 28/5 |
| | | | | | | | | 3 | | | 2 | | | |

| KA | NAME / SAFETY FUNCTION: | IR RO | IR SRO | K1 | K2 | K3 | K4 | K5 | K6 | A1 | A2 | A3 | A4 | G | TOPIC: |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 003A2.02 | Reactor Coolant Pump | 3.7 | 3.9 | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☑ | ☐ | ☐ | ☐ | Conditions which exist for an abnormal shutdown of an RCP in comparison to a normal shutdown of an RCP |
| 004A2.13 | Chemical and Volume Control | 3.6 | 3.9 | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☑ | ☐ | ☐ | ☐ | Low RWST |
| 004K5.26 | Chemical and Volume Control | 3.1 | 3.2 | ☐ | ☐ | ☐ | ☐ | ☑ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | Relationship between VCT pressure and NPSH for charging pumps |
| 005A1.05 | Residual Heat Removal | 3.3 | 3.3 | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☑ | ☐ | ☐ | ☐ | ☐ | Detection of and response to presence of water in RHR emergency sump |
| 006G2.1.30 | Emergency Core Cooling | 4.4 | 4.0 | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☑ | Ability to locate and operate components, including local controls. |
| 007G2.1.20 | Pressurizer Relief/Quench Tank | 4.6 | 4.6 | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☑ | Ability to execute procedure steps. |
| 007K4.01 | Pressurizer Relief/Quench Tank | 2.6 | 2.9 | ☐ | ☐ | ☐ | ☑ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | Quench tank cooling |
| 008A3.08 | Component Cooling Water | 3.6 | 3.7 | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☑ | ☐ | ☐ | Automatic actions associated with the CCWS that occur as a result of a safety injection signal |
| 010K6.02 | Pressurizer Pressure Control | 3.2 | 3.5 | ☐ | ☐ | ☐ | ☐ | ☐ | ☑ | ☐ | ☐ | ☐ | ☐ | ☐ | PZR |
| 010K6.03 | Pressurizer Pressure Control | 3.2 | 3.6 | ☐ | ☐ | ☐ | ☐ | ☐ | ☑ | ☐ | ☐ | ☐ | ☐ | ☐ | PZR sprays and heaters |
| 012K4.02 | Reactor Protection | 3.9 | 4.3 | ☐ | ☐ | ☐ | ☑ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | Automatic reactor trip when RPS setpoints are exceeded for each RPS function; basis for each |

| KA | NAME / SAFETY FUNCTION: | IR RO | IR SRO | K1 | K2 | K3 | K4 | K5 | K6 | A1 | A2 | A3 | A4 | G | TOPIC: |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 013A4.01 | Engineered Safety Features Actuation | 4.5 | 4.8 | | | | | | | | | | ✓ | | ESFAS-initiated equipment which fails to actuate |
| 013K2.01 | Engineered Safety Features Actuation | 3.6 | 3.8 | | ✓ | | | | | | | | | | ESFAS/safeguards equipment control |
| 022A4.03 | Containment Cooling | 3.2 | 3.2 | | | | | | | | | | ✓ | | Dampers in the CCS |
| 026A1.04 | Containment Spray | 3.1 | 3.3 | | | | | | | ✓ | | | | | Containment humidity |
| 026K3.02 | Containment Spray | 4.2 | 4.3 | | | ✓ | | | | | | | | | Recirculation spray system |
| 039A2.01 | Main and Reheat Steam | 3.1 | 3.2 | | | | | | | | ✓ | | | | Flow paths of steam during a LOCA |
| 059K4.02 | Main Feedwater | 3.3 | 3.5 | | | | ✓ | | | | | | | | Automatic turbine/reactor trip runback |
| 061A1.01 | Auxiliary/Emergency Feedwater | 3.9 | 4.2 | | | | | | | ✓ | | | | | S/G level |
| 061K5.05 | Auxiliary/Emergency Feedwater | 2.7 | 3.2 | | | | | ✓ | | | | | | | Feed line voiding and water hammer |
| 062K2.01 | AC Electrical Distribution | 3.3 | 3.4 | | ✓ | | | | | | | | | | Major system loads |
| 063A3.01 | DC Electrical Distribution | 2.7 | 3.1 | | | | | | | | | ✓ | | | Meters, annunciators, dials, recorders and indicating lights |

RO

# T2G1 PWR EXAMINATION OUTLINE



| KA | NAME / SAFETY FUNCTION: | IR RO | IR SRO | K1 | K2 | K3 | K4 | K5 | K6 | A1 | A2 | A3 | A4 | G | TOPIC: |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 063K3.02 | DC Electrical Distribution | 3.5 | 3.7 | | | ✓ | | | | | | | | | Components using DC control power |
| 064K6.08 | Emergency Diesel Generator | 3.2 | 3.3 | | | | | | ✓ | | | | | | Fuel oil storage tanks |
| 073A4.02 | Process Radiation Monitoring | 3.7 | 3.7 | | | | | | | | | | ✓ | | Radiation monitoring system control panel |
| 076K1.01 | Service Water | 3.4 | 3.3 | ✓ | | | | | | | | | | | CCW system |
| 078K3.01 | Instrument Air | 3.1 | 3.4 | | | ✓ | | | | | | | | | Containment air system |
| 103K1.03 | Containment | 3.1 | 3.5 | ✓ | | | | | | | | | | | Shield building vent system |

| KA | NAME / SAFETY FUNCTION: | IR RO | IR SRO | K1 | K2 | K3 | K4 | K5 | K6 | A1 | A2 | A3 | A4 | G | TOPIC: |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 039A2.03 | Main and Reheat Steam | 3.4 | 3.7 | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☑ | ☐ | ☐ | ☐ | Indications and alarms for main steam and area radiation monitors (during SGTR) |
| 062A2.11 | AC Electrical Distribution | 3.7 | 4.1 | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☑ | ☐ | ☐ | ☐ | Aligning standby equipment with correct emergency power source (D/G) |
| 063G2.2.40 | DC Electrical Distribution | 3.4 | 4.7 | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☑ | Ability to apply technical specifications for a system. |
| 076A2.01 | Service Water | 3.5 | 3.7 | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☑ | ☐ | ☐ | ☐ | Loss of SWS |
| 103G2.2.22 | Containment | 4.0 | 4.7 | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☑ | Knowledge of limiting conditions for operations and safety limits. |

| ES-401 | PWR Examination Outline<br>Plant Systems - Tier 2/Group 2 RO / SRO | | | | | | | | | | | | Form ES-401-2 | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| System # / Name | K 1 | K 2 | K 3 | K 4 | K 5 | K 6 | A 1 | A 2 | A 3 | A 4 | G | K/A Topic(s) | IR | # |
| 001 Control Rod Drive | | | | | | | | | | | | | | |
| 002 Reactor Coolant | | | | | | | | | | | | | | |
| 011 Pressurizer Level Control | | R | | | | | | | | | | R 011 K2.02 | | |
| 014 Rod Position Indication | | | | | | | | | | | | | | |
| 015 Nuclear Instrumentation | | | R | | | | | | | | | R 015 K3.01 | | |
| 016 Non-nuclear Instrumentation | | | | | | | | R | | | | R 016 A2.02 | | |
| 017 In-core Temperature Monitor | | | | | | | | | | | S | S 017 G2.1.7 | | |
| 027 Containment Iodine Removal | | | | | | | | | | | | | | |
| 028 Hydrogen Recombiner and Purge Control | | | | | | R | | S | | | | R 028 K6.01<br>S 028 A2.02 | | |
| 029 Containment Purge | | | | | | | | | | | | | | |
| 033 Spent Fuel Pool Cooling | | | | | | | | | | | | | | |
| 034 Fuel Handling Equipment | | | | | | | | | | | R | R 034 G2.4.31 | | |
| 035 Steam Generator | | | | | | | | | | | | | | |
| 041 Steam Dump/Turbine Bypass Control | | | | | | | | | | | | | | |
| 045 Main Turbine Generator | | | | | | | | | | | | | | |
| 055 Condenser Air Removal | | | | | | | | | | | | | | |
| 056 Condensate | | | | | | | | | | | | | | |
| 068 Liquid Radwaste | | | | | | | | S | | | | S 068 A2.04 | | |
| 071 Waste Gas Disposal | | | | | R | | | | | | | R 071 K5.04 | | |
| 072 Area Radiation Monitoring | | | | | | | | | R | | | R 072 A3.01 | | |
| 075 Circulating Water | R | | | | | | | | | | | R 075 K1.01 | | |
| 079 Station Air | | | | | | | | | | R | | R 079 A4.01 | | |
| 086 Fire Protection | | | | R | | | | | | | | R 086 K4.02 | | |
| | | | | | | | | | | | | | | |
| | | | | | | | | | | | | | | |
| | | | | | | | | | | | | | | |
| | | | | | | | | | | | | | | |
| K/A Category Point Totals: | 1 | 1 | 1 | 1 | 1 | 1 | 0 | 1 | 1 | 1 | 1 | Group Point Total: | | 10/3 |
| | | | | | | | | 2 | | | 1 | | | |

| KA | NAME / SAFETY FUNCTION: | IR RO | IR SRO | K1 | K2 | K3 | K4 | K5 | K6 | A1 | A2 | A3 | A4 | G | TOPIC: |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 011K2.02 | Pressurizer Level Control | 3.1 | 3.2 | | ✓ | | | | | | | | | | PZR heaters |
| 015K3.01 | Nuclear Instrumentation | 3.9 | 4.3 | | | ✓ | | | | | | | | | RPS |
| 016A2.02 | Non-nuclear Instrumentation | 2.9 | 3.2 | | | | | | | | ✓ | | | | Loss of power supply |
| 028K6.01 | Hydrogen Recombiner and Purge Control | 2.6 | 3.1 | | | | | | ✓ | | | | | | Hydrogen recombiners |
| 034G2.4.31 | Fuel Handling Equipment | 4.2 | 4.1 | | | | | | | | | | | ✓ | Knowledge of annunciators alarms, indications or response procedures |
| 071K5.04 | Waste Gas Disposal | 2.5 | 3.1 | | | | | ✓ | | | | | | | Relationship of hydrogen/oxygen concentrations to flammability |
| 072A3.01 | Area Radiation Monitoring | 2.9 | 3.1 | | | | | | | | | ✓ | | | Changes in ventilation alignment |
| 075K1.01 | Circulating Water | 2.5 | 2.5 | ✓ | | | | | | | | | | | SWS |
| 079A4.01 | Station Air | 2.7 | 2.7 | | | | | | | | | | ✓ | | Cross-tie valves with IAS |
| 086K4.02 | Fire Protection | 3.0 | 3.4 | | | | ✓ | | | | | | | | Maintenance of fire header pressure |

ES-401, REV 9

# SRO T2G2 PWR EXAMINATION OUTLINE

FORM ES-401-2

| KA | NAME / SAFETY FUNCTION: | IR RO | IR SRO | K1 | K2 | K3 | K4 | K5 | K6 | A1 | A2 | A3 | A4 | G | TOPIC: |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 017G2.1.7 | In-core Temperature Monitor | 4.4 | 4.7 | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☑ | Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior and instrument interpretation. |
| 028A2.02 | Hydrogen Recombiner and Purge Control | 3.5 | 3.9 | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☑ | ☐ | ☐ | ☐ | LOCA condition and related concern over hydrogen |
| 068A2.04 | Liquid Radwaste | 3.3 | 3.3 | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☑ | ☐ | ☐ | ☐ | Failure of automatic isolation |

Facility: Harris Date of Exam: September 2013

| Category | K/A # | Topic | RO IR | RO # | SRO-Only IR | SRO-Only # |
|---|---|---|---|---|---|---|
| 1. Conduct of Operations | 2.1. 13 | Facility Req'ts for vital/controlled access | 2.5 | | | |
| | 2.1. 15 | Temporary Mgmt Directives | 2.7 | | | |
| | 2.1. 19 | Use plant computer | 3.9 | | | |
| | 2.1. | | | | | |
| | 2.1. 41 | Knowledge of refueling processes | | | 3.7 | |
| | 2.1. | | | | | |
| | Subtotal | | 3 | | 1 | |
| 2. Equipment Control | 2.2. 20 | Process for troubleshooting | 2.6 | | | |
| | 2.2. 39 | less than or equal to 1 hr TS actions | 3.9 | | | |
| | 2.2. | | | | | |
| | 2.2. 13 | Tagging and Clearance procedures | | | 4.3 | |
| | 2.2. 40 | Apply Tech Specs for a system | | | 4.7 | |
| | 2.2. | | | | | |
| | Subtotal | | 2 | | 2 | |
| 3. Radiation Control | 2.3. 11 | Control radiation releases | 3.8 | | | |
| | 2.3. 12 | Radiological principles wrt licensed duties | 3.2 | | | |
| | 2.3. | | | | | |
| | 2.3. 4 | Rad exposure limits during normal/emerg | | | 3.7 | |
| | 2.3. 14 | Radiation/Contamination hazards - N, A, or E | | | 3.8 | |
| | 2.3. | | | | | |
| | Subtotal | | 2 | | 2 | |
| 4. Emergency Procedures / Plan | 2.4. 22 | Basis for prioritizing safety functions | 3.6 | | | |
| | 2.4. 27 | Fire in the plant procedures | 3.4 | | | |
| | 2.4. 29 | E-plan Knowledge | 3.1 | | | |
| | 2.4. | | | | | |
| | 2.4. 40 | SRO responsibilities in E-plan implementation | | | 4.5 | |
| | 2.4. 46 | Verify alarms consistent w/ plt conditions | | | 4.2 | |
| | Subtotal | | 3 | | 2 | |
| Tier 3 Point Total | | | 10 | 10 | 7 | 7 |

RO

ES-401, REV 9 **T3 PWR EXAMINATION OUTLINE** FORM ES-401-2

| KA | NAME / SAFETY FUNCTION: | IR RO | IR SRO | K1 | K2 | K3 | K4 | K5 | K6 | A1 | A2 | A3 | A4 | G | TOPIC: |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| G2.1.13 | Conduct of operations | 2.5 | 3.2 | | | | | | | | | | | ✓ | Knowledge of facility requirements for controlling vital / controlled access. |
| G2.1.15 | Conduct of operations | 2.7 | 3.4 | | | | | | | | | | | ✓ | Knowledge of administrative requirements for temporary management directives such as standing orders, night orders, Operations memos, etc. |
| G2.1.19 | Conduct of operations | 3.9 | 3.8 | | | | | | | | | | | ✓ | Ability to use plant computer to evaluate system or component status. |
| G2.2.20 | Equipment Control | 2.6 | 3.8 | | | | | | | | | | | ✓ | Knowledge of the process for managing troubleshooting activities. |
| G2.2.39 | Equipment Control | 3.9 | 4.5 | | | | | | | | | | | ✓ | Knowledge of less than one hour technical specification action statements for systems. |
| G2.3.11 | Radiation Control | 3.8 | 4.3 | | | | | | | | | | | ✓ | Ability to control radiation releases. |
| G2.3.12 | Radiation Control | 3.2 | 3.7 | | | | | | | | | | | ✓ | Knowledge of radiological safety principles pertaining to licensed operator duties |
| G2.4.22 | Emergency Procedures/Plans | 3.6 | 4.4 | | | | | | | | | | | ✓ | Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations. |
| G2.4.27 | Emergency Procedures/Plans | 3.4 | 3.9 | | | | | | | | | | | ✓ | Knowledge of "fire in the plant" procedures. |
| G2.4.29 | Emergency Procedures/Plans | 3.1 | 4.4 | | | | | | | | | | | ✓ | Knowledge of the emergency plan. |

| KA | NAME / SAFETY FUNCTION: | IR RO | IR SRO | K1 | K2 | K3 | K4 | K5 | K6 | A1 | A2 | A3 | A4 | G | TOPIC: |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| G2.1.41 | Conduct of operations | 2.8 | 3.7 | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☑ | Knowledge of the refueling processes |
| G2.2.13 | Equipment Control | 4.1 | 4.3 | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☑ | Knowledge of tagging and clearance procedures. |
| G2.2.40 | Equipment Control | 3.4 | 4.7 | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☑ | Ability to apply technical specifications for a system. |
| G2.3.14 | Radiation Control | 3.4 | 3.8 | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☑ | Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities |
| G2.3.4 | Radiation Control | 3.2 | 3.7 | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☑ | Knowledge of radiation exposure limits under normal and emergency conditions |
| G2.4.40 | Emergency Procedures/Plans | 2.7 | 4.5 | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☑ | Knowledge of the SRO's responsibilities in emergency plan implementation. |
| G2.4.46 | Emergency Procedures/Plans | 4.2 | 4.2 | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☐ | ☑ | Ability to verify that the alarms are consistent with the plant conditions. |

| | |
|---|---|
| Facility: Harris Nuclear Plant | Date of Examination: September 9, 2013 |
| Examination Level: RO ■ SRO ☐ | Operating Test Number: 05000400/2013301 |

| Administrative Topic (see Note) | Type Code* | Describe activity to be performed |
|---|---|---|
| Conduct of Operations | P, R | Determine Rod Misalignment Using Thermocouples (JPM ADM-062-c)<br>*K/A G2.1.7*<br>**2013 NRC RO / SRO A1-1** |
| Conduct of Operations | D, R | Determine Average RCS Boron Concentration per EOP-ECA-0.1 (JPM ADM-020-a) **Common**<br>*K/A G 2.1.20*<br>**2013 NRC RO / SRO A1-2** |
| Equipment Control | M, R | Perform a Quadrant Power Tilt Ratio (QPTR) calculation with a control rod misaligned. (JPM ADM-010-e)<br>*K/A G 2.2.12*<br>**2013 NRC RO A2** |
| Radiation Control | N, R | Using Valve Maps And HP Room Survey Maps Determine Stay Time During Refueling. (JPM ADM-065-a) **Common**<br>*K/A G2.3.4*<br>**2013 NRC RO / SRO A3** |
| Emergency Procedures/Plan | N/A | NOT SELECTED FOR RO<br>**2013 NRC RO A4** |

NOTE: All items (5 total) are required for SROs. RO applicants require only 4 items unless they are retaking only the administrative topics, when all 5 are required.

| * Type Codes & Criteria: | | |
|---|---|---|
| | (C)ontrol room, (S)imulator, or Class(R)oom | **(4)** |
| | (D)irect from bank (≤ 3 for ROs; ≤ 4 for SROs & RO retakes) | **(2)** |
| | (N)ew or (M)odified from bank (≥ 1) | **(2)** |
| | (P)revious 2 exams (≤ 1; randomly selected) | **(1)** |

Rec'd 3/25/13

# 2013 NRC RO Admin JPM Summary

**2013 NRC RO A1-1** - Determine Rod Misalignment Using Thermocouples
(JPM ADM-062-c) Previous - 2011 NRC Exam JPM *randomly selected from bank

*K/A G2.1.7 - Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.*
*(CFR: 41.5 / 43.5 / 45.12 / 45.13) RO 4.4 SRO 4.7*

The plant is at 90% power with a load reduction in progress when a control rod is observed indicating 24 steps higher than group demand. The candidate must perform Attachment 2 of AOP-001, Malfunction of Rod Control and Indication System, to calculate the temperature difference between the affected thermocouple and its symmetric thermocouples. For RO's this JPM requires the candidate to identify the control rod is misaligned and notifies the CRS.

**2013 NRC RO A1-2** - (**Common**) - Determine Average RCS Boron Concentration per EOP-ECA-0.1
(JPM ADM-020-b)

*K/A G2.1.20 - Ability to interpret and execute procedure steps.*
*(CFR: 41.10 / 43.5 / 45.12) RO 4.6 / SRO 4.6*

The candidate must perform a calculation to determine average RCS boron concentration in order to complete a Shutdown Margin calculation as required by EOP-ECA-0.1, Loss Of All AC Power Recovery Without SI Required. The candidate is provided a list of plant conditions and is required to calculate the average RCS boron concentration for these conditions IAW EOP-ECA-0.1, Attachment 1.

**2013 NRC RO A2** - Perform a Quadrant Power Tilt Ratio (QPTR) calculation with a control rod misaligned.
(JPM ADM-010-e) **MODIFIED**

*K/A G2.2.12 - Knowledge of surveillance procedures.*
*(CFR: 41.10 / 45.13) RO 3.7 SRO 4.1*

The candidate must perform a QPTR calculation in accordance with surveillance procedure OST-1039, Calculation of Quadrant power Tilt Ratio, Weekly Interval and as required by the AOP-001, Malfunction of Rod Control and Indication System for a misaligned rod at 95% power. For SRO's this JPM requires the candidate to identify applicable Tech Spec LCOs.

**NOTE:** This JPM will be modified by changing the initial reactor power, the control rod that is dropped into the reactor, and the values of the PRNI upper and lower detectors. These changes result in the QPTR value that exceeds 1.09. The Tech Spec action is now different due to the value exceeding 1.09.



 Rec'd 3/25/13

**2013 NRC RO A3** - (**Common**) - Using Valve Maps And HP Room Survey Maps Determine Stay Time During Refueling.
(JPM-ADM-065-a) **NEW**

*K/A G2.3.4 - Knowledge of radiation exposure limits under normal or emergency conditions. (CFR: 41.12 / 43.4 / 45.10) RO 3.2 / SRO 3.7*

The candidate will be supplied a survey map of a location in the RAB containing various radiation levels and several hot spots. In this area work must be performed by a refueling team shared resource AO. The AO will be required to have continuous HP coverage. The candidates will also have a copy of HP administrative procedures to use during this JPM. The candidate must determine the individual stay times for work in the area. The candidate should determine that the HP Technician cannot stay long enough to complete the task without either exceeding the RWP maximum radiation dose or the administrative yearly dose limit. Since the HP coverage cannot be provided for the work to be completed the job will have to be stopped before it is complete.

**2013 NRC RO A4** – Not selected

Facility: Harris Nuclear Plant Date of Examination: September 9, 2013

Examination Level: RO ☐ SRO ■ Operating Test Number: 05000400/2013301

| Administrative Topic (see Note) | Type Code* | Describe activity to be performed |
|---|---|---|
| Conduct of Operations | P, R | Determine Rod Misalignment Using Thermocouples and Evaluate Tech Specs<br>(JPM ADM-062-d)<br>*K/A G2.1.7*<br>**2013 NRC RO / SRO A1-1** |
| Conduct of Operations | D, R | Determine Average RCS Boron Concentration per EOP-ECA-0.1<br>(JPM ADM-020-b) **Common**<br>*K/A G 2.1.20*<br>**2013 NRC RO / SRO A1-2** |
| Equipment Control | M, R | Perform a Quadrant Power Tilt Ratio (QPTR) calculation with a control rod misaligned and Evaluate Tech Specs.<br>(JPM ADM-010-f)<br>*K/A G 2.2.12*<br>**2013 NRC SRO A2** |
| Radiation Control | N, R | Using Valve Maps And HP Room Survey Maps Determine Stay Time During Refueling.<br>(JPM ADM-065-a) **Common**<br>*K/A G2.3.4*<br>**2013 NRC RO / SRO A3** |
| Emergency Procedures/Plan | N, R | Given a Set of Plant Conditions, Classify an Event.<br>(JPM ADM-064-a)<br>*K/A G2.4.41*<br>**2013 NRC SRO A4** |

NOTE: All items (5 total) are required for SROs. RO applicants require only 4 items unless they are retaking only the administrative topics, when all 5 are required.

| * Type Codes & Criteria: | | |
|---|---|---|
| | (C)ontrol room, (S)imulator, or Class(R)oom | **(5)** |
| | (D)irect from bank (≤ 3 for ROs; ≤ 4 for SROs & RO retakes) | **(2)** |
| | (N)ew or (M)odified from bank (≥ 1) | **(3)** |
| | (P)revious 2 exams (≤ 1; randomly selected) | **(1)** |



Rec'd 3/25/13

# 2013 NRC SRO Admin JPM Summary

**<u>2013 NRC SRO A1-1</u>** - Determine Rod Misalignment Using Thermocouples and Evaluate Tech Specs
(JPM ADM-062-d) Previous - 2011 NRC Exam JPM *randomly selected from bank

*K/A G2.1.7 - Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.*
*(CFR: 41.5 / 43.5 / 45.12 / 45.13) RO 4.4 SRO 4.7*

The plant is at 90% power with a load reduction in progress when a control rod is observed indicating 24 steps higher than group demand. The candidate must perform Attachment 2 of AOP-001, Malfunction of Rod Control and Indication System, to calculate the temperature difference between the affected thermocouple and its symmetric thermocouples. For SRO's this JPM requires the candidate to identify applicable Tech Spec LCOs.

**<u>2013 NRC SRO A1-2</u>** - (**Common**) - Determine Average RCS Boron Concentration per EOP-ECA-0.1
(JPM ADM-020-b) **DIRECT**

*K/A G2.1.20 - Ability to interpret and execute procedure steps.*
*(CFR: 41.10 / 43.5 / 45.12) RO 4.6 / SRO 4.6*

The candidate must perform a calculation to determine average RCS boron concentration in order to complete a Shutdown Margin calculation as required by EOP-ECA-0.1, Loss Of All AC Power Recovery Without SI Required. The candidate is provided a list of plant conditions and is required to calculate the average RCS boron concentration for these conditions IAW EOP-ECA-0.1, Attachment 1.

**<u>2013 NRC SRO A2</u>** - Perform a Quadrant Power Tilt Ratio (QPTR) calculation with a control rod misaligned and Evaluate Tech Specs
(JPM ADM-010-f) **MODIFIED**

*K/A G2.2.12 - Knowledge of surveillance procedures.*
*(CFR: 41.10 / 45.13) RO 3.7 SRO 4.1*

The candidate must perform a QPTR calculation in accordance with surveillance procedure OST-1039, Calculation of Quadrant power Tilt Ratio, Weekly Interval and as required by the AOP-001, Malfunction of Rod Control and Indication System for a misaligned rod at 95% power. For SRO's this JPM requires the candidate to identify applicable Tech Spec LCOs.

**NOTE:** This JPM will be modified by changing the initial reactor power, the control rod that is dropped into the reactor, and the values of the PRNI upper and lower detectors. These changes result in the QPTR value that exceeds 1.09. The Tech Spec action is now different due to the value exceeding 1.09.



Rec'd 3/25/13

# 2013 NRC SRO Admin JPM Summary (continued)

**<u>2013 NRC SRO A3</u>** - **(Common)** - Using Valve Maps And HP Room Survey Maps Determine Stay Time During Refueling.
(JPM-ADM-065-a) **NEW**

*K/A G2.3.4 - Knowledge of radiation exposure limits under normal or emergency conditions. (CFR: 41.12 / 43.4 / 45.10) RO 3.2 SRO 3.7*

The candidate will be supplied a survey map of a location in the RAB containing various radiation levels and several hot spots. In this area work must be performed by a refueling team shared resource AO. The AO will be required to have continuous HP coverage. The candidates will also have a copy of HP administrative procedures to use during this JPM. The candidate must determine the individual stay times for work in the area. The candidate should determine that the HP Technician cannot stay long enough to complete the task without either exceeding the RWP maximum radiation dose or the administrative yearly dose limit. Since the HP coverage cannot be provided for the work to be completed the job will have to be stopped before it is complete.

**<u>2013 NRC SRO A4</u>** - Given a set of conditions, Classify an Event
(JPM-ADM-064-a) **NEW**

*K/A G2.4.41 - Knowledge of the emergency action level thresholds and classifications (CFR: 41.10 / 43.5 / 45.11) RO 2.9 SRO 4.6*

Given a set of initial conditions and the EAL Flow Path, the candidate must classify the appropriate Emergency Action Level for the event in progress.



rec'd 3/25/13

| | |
|---|---|
| Facility: Harris Nuclear Plant | Date of Examination: 09/09/2013 |
| Exam Level: RO SRO-I **SRO-U (bold)** | Operating Test No.: 05000400/2013301 |

Control Room Systems@ (8 for RO); (7 for SRO-I); (**2 or 3 for SRO-U, including 1 ESF - bold**)

| | System / JPM Title | Type Code* | Safety Function |
|---|---|---|---|
| **a.** | **Perform Control Rod and Rod Position Indicator Exercise per OST-1005 (JPM-CR-256-d)** **K/A 001 A2.11** | **A, N, S** | **1** |
| b. | Respond to the loss of the running CSIP (JPM-CR-038-a) *K/A APE 022 AA1.01* | A, D, S | 2 |
| c. | Pressurizer Pressure Master Controller Failure (AOP-019) (JPM-CR-252-a) *K/A APE 027 AA1.03* | A, P, S | 3 |
| **d.** | **Loss of Power to the TDAFW Pump control system (E-0 and OP-137) (JPM-CR-281-a)** ***K/A 054 AA2.04*** | **A, EN, L, N, S** | **4S** |
| e. | Return the Containment Fan Coolers to normal following an SI actuation. (OP-169) (JPM CR-260-a) **RO Only** *K/A 022 A4.01* | D, EN, L, S | 5 |
| **f.** | **Restore Off-site Power to an Emergency Bus (OP-156.02) (JPM-CR-027-b)** ***K/A 062 A4.01*** | **A, D, EN, S** | **6** |
| g. | Restore an Excore NI Channel to service (at power, NI failed low) (OWP-RP-25) (JPM-CR-278-a) *K/A 015 A4.03* | N, S | 7 |
| h. | Align CCW to Support RHR System (OP-145) (JPM CR-085-a) *K/A 008 A4.01* | D, L, S | 8 |



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| In-Plant Systems@ (3 for RO); (3 for SRO-I); (**2 or 3 for SRO-U - BOLD**) | | | |
|---|---|---|---|
| **i.** | **Place the ASI System in Standby Alignment (OP-185)**<br>**(JPM-IP-277-a)**<br>***K/A 004 A4.11*** | **L, N, R** | **2** |
| j. | Local Inspection of Annunciator Cabinets (AOP-037)<br>(JPM IP-273-a)<br>*K/A 016 A2.02* | D, E | 7 |
| **k.** | **Perform an Instrument Air System Leak Isolation Locally (Turbine Bldg / Yard)**<br>**(JPM-IP-161-a)**<br>***K/A APE 065 AA2.03*** | **D, E, L** | **8** |

| | |
|---|---|
| @ | All RO and SRO-I control room (and in-plant) systems must be different and serve different safety functions; all 5 SRO-U systems must serve different safety functions; in-plant systems and functions may overlap those tested in the control room. |

| * Type Codes | Criteria for RO / SRO-I / SRO-U | |
|---|---|---|
| (A)lternate path | 4-6 / 4-6 / 2-3 | (5, 5, 3) |
| (C)ontrol room | 6 2 | |
| (D)irect from bank | ≤ 9 / ≤ 8 / ≤ 4 | (7, 6, 2) |
| (E)mergency or abnormal in-plant | ≥ 1 / ≥ 1 / ≥ 1 | (2, 2, 1) |
| (EN)gineered safety feature | - / - / ≥1 | (3, 2, 2) |
| (L)ow-Power / Shutdown | ≥ 1 / ≥ 1 / ≥ 1 | (5, 4, 3) |
| (N)ew or (M)odified from bank including 1(A) | ≥ 2 / ≥ 2 / ≥ 1 | (4, 4, 3) |
| (P)revious 2 exams | ≤ 3 / ≤ 3 / ≤ 2 | (1, 1, 1) |
| (R)CA | ≥ 1 / ≥ 1 / ≥ 1 | (1, 1, 1) |
| (S)imulator | | |

# 2013 NRC Control Room/In-Plant JPM Summary

**<u>JPM a</u> – Perform Control Rod and Rod Position Indicator Exercise per OST-1005 (JPM-CR-256-d) SRO Upgrade NEW**

*K/A 001 A2.11 – Ability to (a) predict the impacts of the following malfunction or operations on the CRDS- and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Situations requiring a reactor trip (CFR: 41.5/43.5/45.3/45.13) RO 4.4 / SRO 4.7*

The candidate will assume the watch with the unit operating at 100% power and will be directed to perform OST-1005 commencing with Control Bank D in section 7.2. The candidate will insert and withdraw Control Bank D 10 steps as required. The candidate will continue OST-1005 and select the next Control Bank and insert the Control Bank 10 steps as required. Once the candidate begins to insert the next selected Control Bank, the **Alternate Path** will begin and a malfunction of the rod control system will result in the Control Rods continuing to insert once the demand for rod motion has stopped. This will cause RCS Tavg, and Reactor Power will lower in response to the control rods inserting and the Control Rod step counter will continue to lower. The candidate should recognize the failure of the rod control system and perform AOP-001 immediate actions to place Rod Control in manual. The candidate may or may not select the manual position. Rod Control is considered to be in manual as long as the Auto position is not selected and being in Control Bank A satisfies this step. Rod motion will continue in either case requiring the candidate to perform the RNO action and initiate a manual reactor trip. The candidate will announce the Reactor is tripped and begin to perform the immediate actions of E-0. Once the candidate announces entry into E-0, evaluation on this JPM is complete.

**<u>JPM b</u>** – Respond to the loss of the running CSIP (JPM-CR-038-a)

*K/A APE 022 AA1.01 - Ability to operate and / or monitor the following as they apply to the Loss of Reactor Coolant Makeup: CVCS letdown and charging (CFR: 41.7 / 45.5 / 45.6) RO 3.4 / SRO 3.3*

With the plant at 100% power and the ASI system OOS for planned maintenance the candidate will assume the Operator at the Controls (OAC) responsibilities. The A CSIP will trip requiring the candidate to enter AOP-018. AOP-018 will direct the candidate to isolate letdown in response to the loss of charging flow. While the candidate is assessing letdown, RCP Thermal Barrier Flow Control valve (1CC-252) will shut. Following the isolation of letdown the candidate will evaluate the status of component cooling water to the RCP Thermal Barrier and determine that flow is isolated. The **Alternate Path** of this JPM will begin and the candidate should evaluate Attachment 1 for RCP trip limits and determine trip limit 4 Loss of all RCP seal injection (including ASI) is met due to the loss of CCW flow to the RCP Thermal Barrier Hx and trip the reactor. Once the immediate actions of EOP-E-0 are complete the candidate will return to AOP-018 to stop all RCPs and shut the PRZ Spray controllers for RCS loops A and B. Once the candidate has stopped all RCPs and shut the PRZ Spray controllers for RCS loops A and B, evaluation on this JPM is complete.



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## 2013 NRC Control Room/In-Plant JPM Summary

**JPM c** – Pressurizer Pressure Master Controller Failure (AOP-019)
(JPM-CR-252-a) Previous NRC Exam – 2012* randomly selected from bank

*K/A APE 027 AA1.03 – Ability to operate and / or monitor the following as they apply to the Pressurizer Pressure Control Malfunctions: Pressure control when on a steam bubble (CFR 41.7 / 45.5 / 45.6) RO 3.6 / SRO 3.5*

The candidate will assume the Operator at the Controls (OAC) responsibilities and be directed to maintain current plant conditions of 100% steady state power. Soon after assuming the watch the Pressurizer Pressure Master Controller PK-444B will begin to fail in Automatic to 100%. This will cause BOTH Pressurizer Spray valves to go from full closed to the full open position. The candidate should identify the failure and enter AOP-019. While performing the immediate actions the candidate should complete the **Alternate Path** (Take manual control of the Pressurizer Master Controller and lower the output to close the Pressurizer Spray Valves.) IF the candidate takes manual of control of BOTH Pressurizer Spray valves and NOT PK-444B then the master controller will continue to fail and Pressurizer PORV 444B will go full open. When the RCS pressure is < 2000 psig an auto shut signal will be sent to PORV 444B but by this time the pressure excursion will be so great that it will most likely cause an automatic Reactor Trip on OT$\Delta$T and Safety Injection on Low Pressurizer Pressure (at 1850 psig). Once the candidate places the Pressurizer Master Controller is in manual OR both Pressurizer Spray Valves are manually shut AND PORV 444B is shut, evaluation on this JPM is complete.

**JPM d – Loss of Power to the TDAFW Pump control system (E-0 and OP-137)**
**(JPM-CR-281-a) SRO Upgrade**
**NEW**

*K/A APE 054 AA2.04 – Ability to operate and / or monitor the following as they apply to the Loss of Main Feedwater (MFW): Proper operation of AFW pumps and regulating valves (CFR: 43.5 / 45.13) RO 4.2 / SRO 4.3*

The candidate is informed that a Small Break LOCA has occurred, subsequently a leak developed in the Condensate Storage Tank (CST) and the CST level has decreased to less than 10%. The candidate is directed to supply ESW from the A Header to both the A AFW Pump and the Turbine Driven AFW pumps. This will require shutting down the 'A' Train of Containment Fan Coolers. ESW will be aligned to the 'A' AFW Pump. While aligning service water to the Turbine Drive AFW Pump the 'Aux Feedwater Pump Turbine Gov Control Power Failure' alarm (ALB-017-7/3) will annunciate. The **Alternate Path** of this JPM will begin, as the TDAFW Pump speed will begin to increase, TDAFW pump flow and pump discharge pressure will increase. These indications will require the candidate to shut the steam supply valves per the alarm response procedure. Once the A AFW Pump is supplied from the ESW system and both TDAFW pump steam supply valves are shut, evaluation on this JPM is complete.



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# 2013 NRC Control Room/In-Plant JPM Summary

**JPM e** – Return the Containment Fan Coolers to normal following an SI actuation. (OP-169) (JPM CR-260-a) **RO Only**

*K/A 026 A4.01 Ability to manually operate and/or monitor in the control room: CCS fans (CFR: 41.7 / 45.5 to 45.8) RO 3.6 / SRO 3.6*

The candidate is informed an inadvertent SI initiation has occurred and the control room staff has entered EOP-E-0 and EOP-ES-1.1. Attachment 1 of EOP-ES-1.1 is being performed to realign plant systems. The candidate is directed to realign containment fan coolers IAW Attachment 1 step 6.a using OP-169, Containment Cooling And Ventilation, Section 8.4. The candidate will be directed to align the A Train of CNMT Fan Coolers for normal service. The candidate will secure both A Train CNMT Fan Coolers and verify proper damper alignment for the secured fans. The candidate will restart the A Train Fans per section 5.1 of OP-169. To minimize the starting current required for Hi-Speed operation the fans are initially started in Lo-Speed, then stopped and restarted in Hi-Speed. The candidate will return to section 8.4 to secure the B Train of CNMT Fan Coolers. Once the B Train of CNMT Fan Coolers are in standby and the determination is made that Maximum Cooling Mode is NOT required, evaluation on this JPM is complete.

**JPM f – Restore Off-site Power to an Emergency Bus (OP-156.02) (JPM-CR-027-b) SRO Upgrade**

*K/A 062 A4.01 Ability to manually operate and/or monitor in the control room: All breakers (including available switchyard)*
*(CFR: 41.7 / 45.5 / to 45.8) RO 3.3 / SRO 3.1*

The candidate is informed that the plant has tripped due to a LOOP and both EDGs have energized 1A-SA and 1B-SB 6.9kV Emergency Busses. The candidate will be directed by the CRS to restore power to the 1A-SA 6.9kV Emergency Bus IAW OP-156.02, AC Electrical Distribution. The candidate will re-energize the Aux Bus D from Start Up transformer A. With Aux Bus D energized the candidate will close the first Aux Bus D supply breaker to the 1A-SA 6.9kV Emergency Bus. The **Alternate Path** of this JPM will begin after the candidate determines the status of the EDG output breaker. Because the EDG output breaker is shut the candidate must transition to OP-155 to synchronize and transfer the 1A-SA 6.9kV Emergency Bus to offsite power. The candidate will operate the voltage and governor controls to parallel the EDG and offsite power. Once parallel operations have been achieved and the candidate transitions to OP-155 section 7.1 and start to unload the EDG, evaluation on this JPM is complete.



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## 2013 NRC Control Room/In-Plant JPM Summary

**JPM g** – Restore an Excore NI Channel to service (at power, NI failed) (OWP-RP-25) (JPM-CR-278-a)
**NEW**

*K/A 015 A4.03 – Ability to manually operate and/or monitor in the control room: Trip bypasses (CFR: 41.7 / 45.5 to 45.8) RO 3.8 / SRO 3.9*

New JPM to restore previously repaired failed NI-43 to service.
The candidate will assume the watch with the plant at 100% steady state power and the PRNI channel NI-43 which failed downscale earlier repaired. The candidate will be required to return NI-43 to service IAW OWP-RP-25. OWP-RP-25 ensures the components that have NI-43 as an input, Rod Control and SG Feedwater regulating bypass valves are in manual control to prevent spurious movement or uncontrolled changes in level. The candidate will verify the controllers are in manual. The OWP will require the candidate to contact maintenance (I&C personnel) to return the two previously trip bistables for the Channel III OTΔT signals to normal in the Process Instrument Cabinet 3 (PIC-3).
The candidate will return the following items to NORMAL

- At the Detector Current Comparator Drawer: Both upper and lower sections of NI-43
- At the Comparator and Rate Drawer: Comparator Channel Defeat switch

The candidate will return the following items to OPERATE

- At the Miscellaneous Control and Indication Panel: Power Mismatch Bypass switch and the Rod Stop Bypass switch.

The candidate will have to contact maintenance (I&C personnel) a second time and direct them to re-connect the NI-43 power supply leads to the NI drawer. After the I&C personnel re-connect the NI-43 power supply leads, the candidate will verify proper bi-stable and annunciator configuration for the restoration of NI-43 to service. Finally the candidate will have to restore the plant computer (ERFIS) point to processing and document the position of MCB components for the current plant conditions with NI-43. Once the candidate reports that OWP-RP-25 is complete to the CRS, evaluation on this JPM is complete.

**JPM h** – Align CCW to Support RHR System (OP-145) (JPM CR-085-a)

*K/A 008 A4.10 Ability to manually operate and/or monitor in the control room: Conditions that require the operation of two CCW coolers (CFR: 41.7 / 45.5) RO 3.3 / SRO 3.1*

The plant is in Mode 4 and a cool down is in progress. The CRS directs the candidate to align CCW to support RHR operation IAW OP-145 section 8.9. After reviewing section 8.9 the candidate determines a second CCW pump is required to be started and transitions to section 5.2. The candidate starts the B CCW pump IAW section 5.2 and returns to section 8.9 and isolates the A train essential header of the CCW from the non essential header. The candidate will align the B train essential header to supply RHR HX B. The candidate will verify both trains of the CCW system operating parameters are within the required band on the MCB indicators. The candidate will contact a non license operator (NLO) to locally verify the CCW flow to the Gross Failed Fuel Detector is within the required band. Once the candidate contacts the NLO to verify CCW flow locally then evaluation on this JPM is complete.



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# 2013 NRC Control Room/In-Plant JPM Summary

**<u>JPM i</u> – Place the ASI System in Standby Alignment (OP-185)**
**(JPM-IP-277-a) SRO Upgrade**
**NEW**

*K/A 004 A4.11 Ability to manually operate and/or monitor in the control room: RCP Seal injection flow (CFR: 41.7 / 45.5 to 45.8) RO 3.4 / SRO 3.3*

NOTE: This JPM is inside the RCA.

The plant is in Mode 4 and a heat up is in progress. The CRS directs the candidate to place the ASI system in automatic standby alignment IAW OP-185 section 5.1. The candidate will verify the ASI supply header isolation valves are open and the status of the ASI system control panel. The candidate will realign the ASI pump to automatic and return the Squib valve bypass control switches to normal alignment on the ASI control panel. The candidate will turn on the ASI system control panel feeder supply breaker and the ASI pump power supply breaker. The candidate will recheck the indications on the ASI system control panel for the proper standby alignment of the system. Once the candidate proceeds to section 5.1.3, Automatic Standby alignment configuration control closeout, then evaluation on this JPM is complete.

<u>JPM j</u> – Local Inspection of Annunciator Cabinets (AOP-037)
(JPM IP-273-a)

*K/A 016 A2.02 Ability to (a) predict the impacts of the following malfunctions or operations on the NNIS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of power supply (CFR: 41.5 / 43.5 / 45.3 / 45.5) RO 2.9 / SRO 3.2*

The candidate is informed that the control room annunciator System 2 power failure alarm has been received and the CRS has entered AOP-037. The CRS will direct the candidate to check the status of System 2 annunciator power supplies per AOP-037 Attachment 2. The candidate will perform Attachment 2 and obtain the annunciator cabinet key. The JPM cues include information of the proper status of the power supply light indications. The candidate will initial for the indications that remain lit. The candidate will determine based on the cues that one of the System 2, Bay 1, 12 VDC power supplies, one of the System 2, Bay 3, 12 VDC power supplies and the System 2, Bay 5, 24 VDC power supplies are de-energized. Once the CRS is notified that AOP-037, Attachment 2 is complete and the correct de-energized power supplies have been identified then evaluation on this JPM is complete.

**<u>JPM k</u> – Perform an Instrument Air System Leak Isolation Locally (Turbine Bldg / Yard) (AOP-017)**
**(JPM-IP-161-a) SRO Upgrade**

*K/A APE 065 AA2.03 Ability to determine and interpret the following as they apply to the Loss of Instrument Air: Location and isolation of leaks (CFR: 43.5 / 45.13) RO 2.6 / SRO 2.9*

The candidate is informed that the plant was operating at 100% when the plant was tripped due to lowering instrument air pressure. AOP-017 is being performed. The CRS directs the operator to perform Attachment 3 of AOP-017 to reduce instrument air header loads. They will be required to isolate individual sections of the instrument air system within the Turbine Building and contact the Main Control room staff following the completion of each action to determine if the prior actions have successfully isolated the instrument air leak. The JPM cues include information of the proper sequence of actions that must be taken in order reposition the valves and due to the valve locations a description of the nearest ladder location is given to simulate climbing to the valve location. Once notified by the Main Control room that the instrument air header pressure has stabilized then evaluation on this JPM is complete.



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| | | | |
|---|---|---|---|
| Facility: SHEARON-HARRIS | Scenario No.: 1 | Op Test No.: 05000400/2013301 | |
| Examiners: | | Operators: | SRO: |
| | | | RO: |
| | | | BOP: |

Initial Conditions:
- IC-5, BOL, 55% power
- RHR pump A-SA is under clearance for pump seal replacement
- 1SI-4, Boron Injection Tank Outlet valve is under clearance for breaker repairs
- 'B' Condenser Vacuum Pump is under clearance for makeup water supply valve problems
- Boric Acid Transfer Pump A-SA is under clearance for motor replacement

Turnover:
- Plant is at approximately 55% power. Plant startup is in progress IAW GP-005 step 135.e. After taking shift continue plant startup at 4 DEH Units/min.

Critical Tasks:
- Isolate AFW flow to 'C' Steam Generator prior to exiting EOP E-2
- Shut BIT Outlet valve 1SI-3 prior to reaching water relief from PZR SRV's

| Event No. | Malf. No. | Event Type* | Event Description |
|---|---|---|---|
| 1 | N/A | R – RO/SRO<br>N – BOP/SRO | Continue plant startup to 100% power |
| 2 | pt:495 | I - BOP/SRO<br>TS - SRO | Failure of the 'C' SG Pressure Transmitter PT-495 to 0% |
| 3 | lt:112 | I RO/SRO | VCT LT-112 fails high, letdown full divert to RHT (AOP-003) |
| 4 | rms007<br>zcr744 | I - BOP/SRO<br>TS - SRO | Radiation Monitor 3502A fails high and alarms, Containment Purge fails to isolate automatically (AOP-005) |
| 5 | prs14b | I - RO/SRO | Pressurizer Spray Valve, PCV-444D, fails Open (with Manual Control available) (AOP-019) |
| 6 | mss01c | M - All | Steam line Break on 'C' SG inside Containment |
| 7 | zrpk617a<br>zrpk617b | I – BOP/SRO | Failure of Auto AFW Isolation on 'C' SG |
| 8 | nis06b | I – RO/SRO | SR Nuclear Instruments fail to energize post trip due to IR NI-36 undercompensated |

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor

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## SCENARIO SUMMARY: 2013 NRC EXAM SCENARIO 1

Turnover provided to the crew is - The plant is operating at ~55% power in BOL. Criticality was achieved 8 hours ago, 74 hours after a trip from 100% power. A plant startup IAW GP-005 is in progress. After initial turbine loading, the load increase has been performed at 4 Units/min. The unit is currently at ~55% power 82 hours post-trip.

A startup has commenced and the crew has been directed to continue the power increase using GP-005, Power Operation Mode 2 to Mode 1, to 100% power at a ramp rate of 4 DEH Units/Minute.

The following equipment is under clearance:

- RHR Pump A-SA is under clearance for pump packing repairs. The pump has been inoperable for 12 hours and is expected to be repaired in the next 24 hours. The pump must be restored to operable status within the next 60 hours. Tech Spec 3.5.2 LCO Action **a.** and Tech Spec 3.3.3.5.b Action **c.** applies. OWP-RH-01 has been completed.

EMERGENCY CORE COOLING SYSTEMS

3/4.5.2 ECCS SUBSYSTEMS - $T_{avg}$ GREATER THAN OR EQUAL TO 350°F

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent Emergency Core Cooling System (ECCS) subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE Charging/safety injection pump.

b. One OPERABLE RHR heat exchanger.

c. One OPERABLE RHR pump, and

d. An OPERABLE flow path capable of taking suction from the refueling water storage tank on a Safety Injection signal and, upon being manually-aligned, transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.



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## SCENARIO SUMMARY: 2013 NRC EXAM SCENARIO 1 (continued)

Tech Specs associated with inoperable RHR Pump A-SA continued

INSTRUMENTATION

REMOTE SHUTDOWN SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.3.5.a The Remote Shutdown System monitoring instrumentation channels shown in Table 3.3-9 shall be OPERABLE.

3.3.3.5.b All transfer switches, Auxiliary Control Panel Controls and Auxiliary Transfer Panel Controls for the OPERABILITY of those components required by the SHNPP Safe Shutdown Analysis to (1) remove decay heat via auxiliary feedwater flow and steam generator power-operated relief valve flow from steam generators A and B, (2) control RCS inventory through the normal charging flow path, (3) control RCS pressure, (4) control reactivity, and (5) remove decay heat via the RHR system shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

c. With one or more inoperable Remote Shutdown System transfer switches, power, or control circuits required by 3.3.3.5.b, restore the inoperable switch(s)/circuit(s) to OPERABLE status within 7 days, or be in HOT STANDBY within the next 12 hours.

- 'B' Condenser Vacuum Pump is under clearance for makeup water supply valve problems
- 1SI-4, Boron Injection Tank Outlet valve is under clearance for breaker repairs. Tech Spec 3.5.2 Action **a** applies. OWP-SI-01 has been completed.

EMERGENCY CORE COOLING SYSTEMS

3/4.5.2 ECCS SUBSYSTEMS - $T_{avg}$ GREATER THAN OR EQUAL TO 350°F

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent Emergency Core Cooling System (ECCS) subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE Charging/safety injection pump.

b. One OPERABLE RHR heat exchanger,

c. One OPERABLE RHR pump, and

d. An OPERABLE flow path capable of taking suction from the refueling water storage tank on a Safety Injection signal and, upon being manually aligned, transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

- Boron Injection Pump A-SA is under clearance for motor replacement. Tech Spec 3.1.2.2 applies (tracking only). OWP-CS-04 has been completed.

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## SCENARIO SUMMARY: 2013 NRC EXAM SCENARIO 1 (continued)

**Event 1**: Continue plant startup to 100% power - Crew performs a power increase of approximately 5%-10% power (Lead Examiners discretion). For this reactivity manipulation it is expected that the SRO will conduct a reactivity brief, the RO will dilute and monitor auto rod withdrawal per the reactivity plan and the BOP will operate the DEH Controls as necessary to raise Main Turbine power.

**Event 2**: Failure of the 'C' SG Pressure Transmitter PT-495 to 0%. This event will require the BOP to place the 'C' SG level control to manual and control SG level within Reactor trip limits. The SRO should provide level band and trip guidance IAW OMM-001. The crew will take the channel out of service using OWP-ESF-04, Protection channel III Steam flow. The SRO should evaluate Tech Spec 3.3.1 action 6, Tech Spec 3.3.3.6 Accident Monitoring Instrumentation Action **a,** and Tech Spec 3.3.2 1.e Steam Line Press Low Action 19 applies.

T.S. 3.3.1: As a minimum, the Reactor Trip System instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE

REACTOR TRIP SYSTEM INSTRUMENTATION

| FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| 14. Steam Generator Water Level--Low Coincident With Steam/ Feedwater Flow Mismatch | 2 stm. gen. level and 2 stm./feed-water flow mismatch in each stm. gen. | 1 stm. gen. level coincident with 1 stm./feedwater flow mismatch in same stm. gen. | 1 stm. gen. level and 2 stm./feed-water flow mismatch in same stm. gen. or 2 stm. gen. level and 1 stm./feedwater flow mismatch in same stm. gen. | 1, 2 | 6 |

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours, and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1.

TS 3.3.3.6 The Accident Monitoring instrumentation channels shown in Table 3.3-10 shall be OPERABLE

- ACTION a. - With the number of OPERABLE accident monitoring instrumentation channels except In Core Thermocouples and Reactor Vessel Level less than the Total Required Number of Channels requirements shown in Table 3.3-10 restore the inoperable channel(s) to OPERABLE status within 7 days. or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

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## SCENARIO SUMMARY: 2013 NRC EXAM SCENARIO 1 (continued)

Tech Spec 3.3.2

INSTRUMENTATION

3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The Engineered Safety Features Actuation System (ESFAS) instrumentation channels and interlocks shown in Table 3.3-3 shall be OPERABLE with their Trip Setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4.

APPLICABILITY: As shown in Table 3.3-3.

TABLE 3.3-3

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

| FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| 1. Safety Injection (Reactor Trip, Feedwater Isolation, Control Room Isolation, Start Diesel Generators, Containment Ventilation Isolation, Phase A Containment Isolation, Start Auxiliary Feedwater System Motor-Driven Pumps, Start Containment Fan Coolers, Start Emergency Service Water Pumps, Start Emergency Service Water Booster Pumps) | | | | | |
| e. Steam Line Pressure--Low | 3/steam line | 2/steam line in any steam line | 2/steam line | 1, 2, 3# | 19 |

ACTION 19 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours, and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.2.1.

The SRO should also prepare OMM-001, Attachment 5 Equipment Problem Checklist for the failure.

rec'd 3/25/13

## SCENARIO SUMMARY: 2013 NRC EXAM SCENARIO 1 (continued)

**Event 3**: LT-112 fails HIGH – full divert to RHT. Enter AOP-003. This will require entry into AOP-003, Malfunction of Reactor Makeup Control (no immediate actions). A failure of LT-112 high will cause 1CS-120, Letdown VCT/Hold Up Tank valve to shift to the Hold Up Tank. The RO will have to return the MCB switch to the VCT position. Since VCT level has failed HIGH auto CSIP suction switch over on 5% VCT level to the RWST will not occur until Maintenance has lifted the leads associated with LT-112. The operator will have to monitor VCT level and communicate with Maintenance to resolve this failure.

The SRO should also prepare OMM-001, Attachment 5 Equipment Problem Checklist for the failure.

**Event 4**: Radiation Monitor 3502A high alarm, Containment Purge fails to isolate automatically. This failure will cause the radiation monitor output to immediately fail high and RM-11 to go into High Alarm. The automatic response to isolate Normal Containment Purge fails to occur due to a failed relay. The crew should respond to the alarms and enter AOP-005, Radiation Monitoring (no immediate actions). AOP-005 Attachment 1 will direct verifying that the automatic response for this alarm has occurred (other procedure options are available and detailed in exercise guide). This will also require the SRO to evaluate Tech Spec 3.3.2 for the failed Containment Isolation and Tech Spec 3.4.6.1, Leakage Detection Systems.

Tech Spec 3.4.6.1, Leakage Detection Systems, Action **a**

REACTOR COOLANT SYSTEM

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

---

3.4.6.1 The following Reactor Coolant System Leakage Detection Systems shall be OPERABLE:

a. The Containment Airborne Gaseous Radioactivity Monitoring System,
b. The Reactor Cavity Sump Level and Flow Monitoring System, and
c. The Containment Airborne Particulate Radioactivity Monitoring System.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With a. or c. of the above required Leakage Detection Systems INOPERABLE, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed for airborne gaseous and particulate radioactivity at least once per 24 hours when the required Airborne Gaseous or Particulate Radioactivity Monitoring System is inoperable; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.



rec'd 3/25/13

## SCENARIO SUMMARY: 2013 NRC EXAM SCENARIO 1 (continued)

Tech Spec 3.3.3.1 - Table 3.3-6 item 1.b.1) Airborne Gaseous Radioactivity – RCS leakage Detection Actions 26 and 27

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS

| INSTRUMENT | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ALARM/TRIP SETPOINT | ACTION |
|---|---|---|---|---|---|
| 1. Containment Radioactivity-- | | | | | |
| b. Airborne Gaseous Radioactivity | | | | | |
| 1) RCS Leakage Detection | 1 | 1 | 1, 2, 3, 4 | $\leq 1.0\times10^{-3}$ μCi/ml | 26, 27 |
| 2) Pre-entry Purge | 1 | 1 | ## | $\leq 2.0\times10^{-3}$ μCi/ml | 30 |

ACTION 26 - Must satisfy the ACTION requirement for Specification 3.4.6.1.

ACTION 27 - With less than the Minimum Channels OPERABLE requirement, operation may continue provided the containment purge makeup and exhaust isolation valves are maintained closed.

The SRO should also prepare OMM-001, Attachment 5 Equipment Problem Checklist for the failure.

**Event 5**: Pressurizer Spray Valve, PCV-444D, fails to maximum output in Auto (with manual control available). This failure will cause one of the Pressurizer spray valve to fail to 100% open while the other valve closes to 0% open. The crew should respond to multiple alarms and enter AOP-019, Malfunction of RCS Pressure Control. The RO should complete the immediate actions by gaining control of the Pressurizer Spray Valves. If RCS pressure decreases to < 2202 psig during the event the SRO will have to evaluate Tech Spec 3.2.5

POWER DISTRIBUTION LIMITS

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB-related parameters shall be maintained within the following limits:

a. Reactor Coolant System $T_{avg} \leq 594.8°F$ after addition for instrument uncertainty, and

b. Pressurizer Pressure $\geq$ 2185 psig* after subtraction for instrument uncertainty, and

c. RCS total flow rate $\geq$ 293,540 gpm after subtraction for instrument uncertainty.

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters not within its specified limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 6 hours.

The SRO should also prepare OMM-001, Attachment 5 Equipment Problem Checklist for the failure.

Rec'd 3/25/13

## SCENARIO SUMMARY: 2013 NRC EXAM SCENARIO 1 (continued)

**Event 6: MAJOR** – Steam line Break on 'C' SG inside Containment. Once RCS pressure control has been established to the satisfaction of the Lead Examiner a Steam Line Break inside Containment on the 'C' SG will occur. The crew should enter and carry out the immediate actions of EOP E-0.

The crew should diagnose that a LOCA is NOT in progress and transition from EOP E-0 to EOP E-2, Faulted Steam Generator Isolation.

While the crew is performing actions of E-2 the Containment pressure will continue to rise beyond 10 psig which will actuate a Containment isolation Phase B signal and require ALL RCPs to be secured.

**Event 7**: Failure of Auto AFW Isolation on 'C' SG. The crew should identify that an actuation signal for AFW Auto Isolation has failed on the 'C' SG and manually isolate AFW flow to the 'C' SG.

**Event 8**: Source Range channels will fail to energize post trip due to IR NI-36 under compensation. The crew will need to identify the failure of the SR instrumentation MCB indication and audible counts. They will then manually energize the SR channels to establish an audio count rate.

The scenario is ended when Safety Injection has been terminated and the crew transitions to EOP-ES-1.1, SI Termination.



Rec'd 3/25/13

## CRITICAL TASK JUSTIFICATION:

1. Isolate AFW flow to 'C' Steam Generator prior to exiting EOP E-2.

   Failure to isolate a faulted Steam Generator that can be isolated causes challenges to the Critical Safety Functions beyond those irreparably introduced by the postulated conditions. Also, depending upon the plant conditions, it could constitute a demonstrated inability by the crew to recognize a failure of the automatic actuation of an ESF system or component. This critical task requires the crew to recognize an automatic actuation should have occurred of an ESF system or component but has not and then take manual operator actions to perform the isolation.

2. Shutting BIT outlet valves 1SI-3 prior to water relief through the PZR Safety Relief Valves (SRV's).

   FSAR Section 15.1.5.2 (page 15.1.5-7) states the operator will secure one of the two CSIPs to facilitate PZR level indication remaining on scale and controllable. Additionally, shutting the BIT outlet valves is the first steps in realigning normal charging to the RCS. Continued flow through the BIT will cause the PZR level to reach solid conditions potentially causing the SRV's to lift. At low fluid temperature (like those present in the PZR at this time) the SRV's may not reset after fluid operation. This operator action is critical to prevent a challenge to plant safety



Rec'd 3/25/13

# HARRIS 2013 NRC SCENARIO 2

Facility: SHEARON-HARRIS Scenario No.: 2 Op Test No.: 05000400/2013301

Examiners: ______ ______ ______ Operators: SRO: ______ RO: ______ BOP: ______

Initial Conditions:
- IC-26, MOL, ~88% power
- RHR pump A-SA is under clearance for pump seal replacement
- 1SI-4, Boron Injection Tank Outlet valve is under clearance for breaker repairs
- 'B' Condenser Vacuum Pump is under clearance for makeup water supply valve problems
- Boric Acid Transfer Pump A-SA is under clearance for motor replacement

Turnover:
- A shutdown is in progress due to problems encountered during the repairs on the 'A' RHR pump. Repairs will not be able to be completed prior to the LCO expiring. The plant is operating at ~88% power in MOL. When turnover is complete continue the power reduction at 4 DEH units/min to support being in Mode 3 within the next 6 hours. GP-006 is currently in progress with step 8 completed. 300 gallons of boric acid have been injected IAW the reactivity plan. Control Bank D Rods have just stepped IN. All required notifications have been made to individuals concerning the reason for the shutdown.

Critical Tasks:
- Insert negative reactivity into the core by inserting control rods during an ATWS before completing the immediate action steps of FR-S.1
- Trip RCPs once RCP Trip Foldout Criteria is met and prior to exiting E-1

| Event No. | Malf. No. | Event Type* | Event Description |
|---|---|---|---|
| 1 | N/A | R – RO/SRO<br>N – BOP/SRO | Continue plant shutdown at 4 DEH Units/min |
| 2 | crf14b | C – RO/SRO | Rods fail to operate in AUTO (manual control works- setting up for ATWS) Enter AOP-001 |
| 3 | ft:497 | I – BOP/SRO<br>TS- SRO | Feed flow transmitter on 'C' SG FT-497 Channel IV (selected for 1C SG) fails low – OMM-001 |
| 4 | idi xd1i142<br>ilo xd1o142w<br>ian xn27e05 | C - BOP/SRO | Reactor Primary Shield Fan S2-A-SA Failure |
| 5 | cvc05a | C - RO/SRO<br>TS-SRO | CSIP Trip - 1 available, requiring AOP-018 entry<br>ASI Pump start / Respond to Boron addition to RCS from ASI |
| 6 | N/A | N - RO/SRO | Restore letdown IAW OP-107 |
| 7 | rcs18b | R – RO/SRO<br>N – BOP/SRO | ~30 gpm RCS leak to Containment. Enter AOP-016 increase ramp rate IAW AOP-038 |
| 8 | rcs18b<br>rps01b | M – ALL<br>ATWS | RCS leakage exceeds VCT makeup capability E-0 manual Rx Trip with ATWS Reactor Trip Breakers fail to open in auto or manual |



Rec'd 7/25/13

**SCENARIO 2 continued**

| Event No. | Malf. No. | Event Type* | Event Description |
|---|---|---|---|
| 9 | rcs18b | M – ALL | RCS leakage increases to SBLOCA ~650 gpm |
| 10 | zdsq2:6b<br>jpb9101b | C – RO/SRO | 'B' ESW pump fails to auto start from the Sequencer or from low pressure |
| * (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor | | | |

## SCENARIO SUMMARY: 2013 NRC EXAM SCENARIO 2

Turnover provided to the crew is – A shutdown is in progress due to problems encountered during the repairs on the 'A' RHR pump. Repairs will not be able to be completed prior to the LCO expiring. The plant is operating at ~88% power in MOL. Control Bank D is at 200 steps. The latest RCS Boron sample was 1067 ppm. GP-006 step 9 is in progress. When turnover is complete continue the power reduction at 4 DEH units/min to support being in Mode 3 within the next 6 hours. All required notifications have been made to individuals concerning the reason for the shutdown.

The following equipment is under clearance:

- RHR Pump A-SA is under clearance for pump packing repairs. The pump has been inoperable for 12 hours and must be restored to operable status within the next 60 hours. Tech Spec 3.5.2 LCO Action **a** and Tech Spec 3.3.3.5.b Action **c** applies. OWP-RH-01 has been completed.

EMERGENCY CORE COOLING SYSTEMS

3/4.5.2 ECCS SUBSYSTEMS - $T_{avg}$ GREATER THAN OR EQUAL TO 350°F

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent Emergency Core Cooling System (ECCS) subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE Charging/safety injection pump,

b. One OPERABLE RHR heat exchanger,

c. One OPERABLE RHR pump, and

d. An OPERABLE flow path capable of taking suction from the refueling water storage tank on a Safety Injection signal and, upon being manually aligned, transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.



Rec'd 3/25/13

## SCENARIO SUMMARY: 2013 NRC EXAM SCENARIO 2 (continued)

Tech Specs associated with inoperable RHR Pump A-SA continued

INSTRUMENTATION

REMOTE SHUTDOWN SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.3.5.b All transfer switches, Auxiliary Control Panel Controls and Auxiliary Transfer Panel Controls for the OPERABILITY of those components required by the SHNPP Safe Shutdown Analysis to (1) remove decay heat via auxiliary feedwater flow and steam generator power-operated relief valve flow from steam generators A and B, (2) control RCS inventory through the normal charging flow path, (3) control RCS pressure, (4) control reactivity, and (5) remove decay heat via the RHR system shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

c. With one or more inoperable Remote Shutdown System transfer switches, power, or control circuits required by 3.3.3.5.b, restore the inoperable switch(s)/circuit(s) to OPERABLE status within 7 days, or be in HOT STANDBY within the next 12 hours.

- 'B' Condenser Vacuum Pump is under clearance for makeup water supply valve problems
- 1SI-4, Boron Injection Tank Outlet valve is under clearance for breaker repairs. Tech Spec 3.5.2 Action **a** applies. OWP-SI-01 has been completed.

EMERGENCY CORE COOLING SYSTEMS

3/4.5.2 ECCS SUBSYSTEMS - $T_{avg}$ GREATER THAN OR EQUAL TO 350°F

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent Emergency Core Cooling System (ECCS) subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE Charging/safety injection pump,

b. One OPERABLE RHR heat exchanger,

c. One OPERABLE RHR pump, and

d. An OPERABLE flow path capable of taking suction from the refueling water storage tank on a Safety Injection signal and, upon being manually aligned, transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

- Boron Injection Pump A-SA is under clearance for motor replacement. Tech Spec 3.1.2.2 applies (tracking only). OWP-CS-04 has been completed.



Rec'd 3/25/13

## SCENARIO SUMMARY: 2013 NRC EXAM SCENARIO 2 continued

**Event 1**: Continue plant shutdown at 4 DEH Units/min - Crew performs a power decrease of approximately 5%-10% power (Lead Examiners discretion). For this reactivity manipulation it is expected that the SRO will conduct a reactivity brief, the RO will borate and monitor auto rod insertion per the reactivity plan and the BOP will operate the DEH Controls as necessary to reduce Main Turbine load.

**Event 2:** Rods do not move in AUTO (manual control works- setting up for ATWS) Enter AOP-001. During the downpower the rod control system will fail to operate in AUTO and the crew will be required to enter AOP-001, Malfunction of Rod Control and Indication System. The crew must take immediate actions and place rods in MANUAL. The SRO will have a TS check of 3.1.3.1, 3.1.3.5 and determine that all rods are trippable per Attachment 5 of AOP-001. System Engineering will be contacted. The crew will be expected to continue with the power reduction. If necessary, to get the crew moving, prompting by the Manager of Operations can be used.

TS 3.1.3.1 All shutdown and control rods shall be OPERABLE and positioned within + 12 steps (indicated position) of their group step counter demand position.

Applicability: Modes 1 and 2

TS 3.1.3.4 All shutdown rods shall be fully withdrawn as specified in the CORE OPERATING LIMITS REPORT (COLR), plant procedure PLP-106.

Applicability: Modes 1 and 2

AOP-001 Attachment 5

MALFUNCTION OF ROD CONTROL AND INDICATION SYSTEM

Attachment 5 - Determination of Control Rod Trippability
Sheet 1 of 1

The following guidance is provided for making the determination of control rod trippability:

A control rod may be considered trippable under any of the following circumstances:

- Rod Control System URGENT FAILURE alarm exists
- Inspection of the affected system cabinets reveals obvious electrical problems (for example, blown fuses)
- All rods of a particular group or bank are simultaneously affected
- NO control rod motion is possible

If none of the four conditions exist the rod must be considered untrippable until proven otherwise.



Rec'd 3/25/13

## SCENARIO SUMMARY: 2013 NRC EXAM SCENARIO 2 continued

**Event 3:** Feed flow transmitter on 'C' SG FT-497 Channel IV (selected for 1C SG) fails low – OMM-001. When the plant is in a stable condition, the Lead Evaluator can cue the SG 'C' Feed Flow channel failure. The crew should respond in accordance with the alarm response procedure and OMM-001. The BOP will be controlling SG 'C' level with the FRV in MANUAL and may switch controlling FF channels to restore control to AUTO. The crew will take the channel out of service using OWP-RP-10, SF/FF Loop 3. The SRO should evaluate the following Tech Specs for failure of FT-497:

T.S. 3.3.1: As a minimum, the Reactor Trip System instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE

REACTOR TRIP SYSTEM INSTRUMENTATION

| FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| 14. Steam Generator Water Level--Low Coincident With Steam/ Feedwater Flow Mismatch | 2 stm. gen. level and 2 stm./feed-water flow mismatch in each stm. gen. | 1 stm. gen. level coincident with 1 stm./feedwater flow mismatch in same stm. gen. | 1 stm. gen. level and 2 stm./feed-water flow mismatch in same stm. gen. or 2 stm. gen. level and 1 stm./feedwater flow mismatch in same stm. gen. | 1, 2 | 6 |

**ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:**

**a. The inoperable channel is placed in the tripped condition within 6 hours, and**

**b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1.**

The crew should implement OWP-RP-10 for this failure.

**Event 4:** Reactor Primary Shield Fan S2-A-SA Failure. When event 3 concludes to the satisfaction of the Lead Examiner then event 4 "Reactor Primary Shield Cooling Fan S2-A-SA failure" can be inserted. The crew will respond to ALB 027-05-5, Reactor Primary Shield Clg Fans S2 Low Flow - O/L and evaluate the condition. The crew should identify that the running fan has tripped, start the standby fan IAW OP-169, Containment Cooling and Ventilation, and dispatch an operator to check the status of the breaker. If flow cannot be established by starting the standby fan then APP-ALB-027-05-5 will direct a Reactor Shutdown to be initiated within 1 hour and cooldown to <350°F within 6 hours would be required. OP-169 P&L 4.0.3 states One Reactor Primary Shield Fan is required to be in operation anytime RCS temperature is greater than 140°F.



Rec'd 3/25/13

**SCENARIO SUMMARY: 2013 NRC EXAM SCENARIO 2 continued**

**Event 5:** Trip of the running 'A' Charging Pump breaker. The crew will enter AOP-018 and carry out the immediate actions. The crew should isolate letdown and then implement actions to place the 'B' Charging Pump in service. The crew will have to secure the ASI pump after the CSIP is started and evaluate the boration caused by the ASI pump running. The efficiency that the crew has in progressing through AOP-018 to the point of securing the ASI pump will determine the amount of boric acid added to the RCS through the RCP seals. This could require the SRO to direct the RO and BOP to coordinate Reactor and Turbine controls (dilute, rod movement and / or Turbine reduction) to accommodate the boron addition for Tavg/Tref stabilization.
The SRO should evaluate the loss of the CSIP in accordance with Tech Specs 3.1.2.2, 3.1.2.4 and 3.5.2
TS 3.1.2.2 - At least two of the following three boron injection flow paths shall be OPERABLE:

b. Two flow paths from the refueling water storage tank via charging/safety injection pumps to the RCS.

ACTION: With only one of the above required boron injection flow paths to the RCS OPERABLE, restore at least two boron injection flow paths to the RCS to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN as specified in the CORE OPERATING LIMITS REPORT (COLR), plant procedure PLP-I06 at 200°F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in HOT SHUTDOWN within the next 6 hours.

TS 3.1.2 4 - With only one Charging/safety injection pump OPERABLE restore at least two charging/safety injection pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN as specified in the CORE OPERATING LIMITS REPORT (COLR) plant procedure PLP-106 at 200°F within t he next 6 hours: restore at least two charging/safety injection pumps to OPERABLE status within the next 7 days or be in HOT SHUTDOWN within the next 6 hours.

TS 3.5.2 - Two independent Emergency Core Cooling System (ECCS) subsystems shall be OPERABLE with each subsystem comprised of:

One OPERABLE Charging/safety injection pump
One OPERABLE RHR heat exchanger
One OPERABLE RHR pump and
An OPERABLE flow path capable of taking suction from the refueling water storage tank on a Safety Injection signal and. upon being manually aligned transferring suction to the containment sump during the recirculation phase of operation.

ACTION: a. With one ECCS subsystem inoperable restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

The SRO should also prepare OMM-001, Attachment 5 Equipment Problem Checklist for the failure



Rec'd 3/25/13

## SCENARIO SUMMARY: 2013 NRC EXAM SCENARIO 2 (continued)

**Event 6**: Restore letdown IAW OP-107 - once the 'B' Charging Pump is in service, the crew will restore letdown IAW OP-107 Section 5.4 to establish inventory control. Once letdown has been restored, Tech Specs have been evaluated for the loss of the 'A' CSIP, and the crew response to the boron addition from the ASI system have been addressed to the satisfaction of the Lead Examiner the next event can be initiated.

**Event 7:** ~30 gpm RCS leak to Containment. Enter AOP-016 increase ramp rate IAW AOP-038. This will be the initiating event for the Major event. The crew will identify that leakage is present by radiation monitor alarms associated with increasing Containment sump leakage. Pressurizer level will decrease and Charging flow will increase. The crew should enter AOP-016, Excessive Primary Plant Leakage (no immediate actions) based on the indications of unidentified RCS leakage. The crew should determine that RCS leakage is now exceeding TS 3.4.6.2 leakage of 1 gpm Unidentified Leakage but RCS leakage is within VCT makeup capability $\leq$120 gpm). They should determine that Attachment 7 for leakage inside Containment is the applicable attachment and stop Containment Purge. Since a shutdown is already in progress IAW GP-006, the crew should evaluate that a more rapid means of plant shutdown is required and enter AOP-038 The SRO should implement AOP-038, Rapid Downpower and direct an increase of the ramp rate to some value $\geq$ 5 DEH Units/min in an attempt to rapidly remove the unit from service.

**Event 8:** RCS leakage exceeds VCT makeup capability E-0 manual Rx Trip with ATWS Reactor Trip Breakers fail to open in auto or manual. The crew will continue actions of AOP-038 and attempt a manual Reactor trip when leakage exceeds VCT makeup capability IAW AOP-016 continuous action step 4. When the RO attempts to perform a Manual Reactor trip he/she will find that the Reactor will not trip from either of the MCB trip switches. The crew will enter FR-S.1, Response to Nuclear Power Generation/ATWS and perform the immediate actions of manually inserting control rods (auto rod insertion is not available due to event 2 malfunction), tripping the main Turbine, starting AFW and directing an AO to locally trip the Reactor. The Booth Operator acting as the field operator will locally trip the Reactor trip breakers when the crew initiates emergency boration of the RCS. The crew will then continue with FR-S.1 until completing step 10 or when the foldout for Reactor Subcriticality Criteria is met. At that time they will transition from FR-S.1 to E-0.

**Event 9:** RCS leakage increases to SBLOCA ~650 gpm. After the crew transitions from FR-S.1 to E-0 they will verify the E-0 immediate actions and actuate SI if automatic SI initiation (PRZ Pressure less than 1850 psig) did not occur after the Reactor trip breakers were opened. The RCS leakage will increase in magnitude. RCS pressure will decrease to less than 1400 psig which will require the crew to trip the RCP's based on E-0 foldout criteria of RCS pressure less than 1400 psig and SI flow greater than 200 gpm.



Rec'd 3/25/1

## SCENARIO SUMMARY: 2013 NRC EXAM SCENARIO 2 (continued)

**Event 10:** 'B' ESW pump fails to start from the sequencer or from low pressure. This failure requires the crew to manually start the 'B' ESW pump. The crew should identify that the 'B' ESW pump did not start from the B sequencer. Either the RO should identify that the 'B' ESW pump is not running by observation of the MCB OR the BOP should identify that the B sequencer has skipped the 'B' ESW pump start. IF neither operator identifies that the 'B' ESW pump did not start then the BOP should identify that it is NOT running when performing EOP E-0 Attachment 3. Step 4 of the attachment has the operator "verify" that ALL ESW AND ESW Booster Pumps are running. By the time the operator performs this step in attachment 4 the actions of AOP-022 to secure both the 'B' CSIP and the 'B' EDG will have been met. The reason for securing the CSIP and EDG can be found in AOP-022 Basis Document. It states that both the CSIPs and EDGs are considered an essential load and requires the component to be stopped to protect against equipment damage due to overheating.

Securing the 1B SB EDG will remove the emergency power supply to the 'B' RHR pump. The 'B' RHR pump will continue to run since it was started by the 'B' sequencer but will only have the normal power source available to it.

The scenario can be terminated when directed to initiate RCS cooldown in ES-1.2.



Rec'd 3/25/13

## CRITICAL TASK JUSTIFICATION:

1. Insert negative reactivity into the core by inserting control rods during an ATWS before completing the immediate action steps of FR-S.1.

   There are four (4) immediate actions steps of FR-S.1 at HNP with step 4 directing an operator to contact OR report to the MCR to receive instructions to locally trip the reactor. Control rod insertion to add negative reactivity to the core attempting to bring the reactor core subcritical is crucial to prevent the possibility of core damage. Not performing rod insertion prior to completing the immediate actions of FR-S.1 is demonstrating the lack of ability to complete a required operator action during a function restoration procedure.

2. Trip RCPs once RCP Trip Foldout Criteria is met and prior to exiting E-1.

   Securing RCPs during a SB LOCA event will prevent depleting the RCS to a critical inventory by pumping more mass through the break than would occur if the RCP operation were ceased. (Critical inventory is defined as the amount of inventory remaining in the RCS when the break completely uncovers and the break flow changes from a mixture of liquid and steam to all steam.) Both E-0 and E-1 foldout criteria requires RCP's to be secured when SI flow of > 200 gpm is established and when RCS pressure is < 1400 psig. IF the crew continues to allow the RCPs to operate then RCS inventory will continue to deplete. Manually tripping the RCPs before depletion below the critical inventory conservatively ensures that Peak Clad Temperature remains below 2200°F. If the crew does NOT secure the RCP's prior to transition from E-1 then RCP trip criteria will no longer exist and the RCP's could continue to run.



Rec'd 3/25/1

Facility: SHEARON-HARRIS Scenario No.: 3 Op Test No.: 05000400/2013301

Examiners: ______ ______ ______ Operators: ______ ______ ______

Initial Conditions: IC-19, MOL, 100% power

- 'B' MD AFW Pump under clearance 12 hours ago for pump packing replacement, due back in 24 hours, TS 3.7.1.2 – 72 hour LCO
- 1SI-3, Boron Injection Tank Outlet valve is under clearance for breaker repairs
- 'B' Condenser Vacuum Pump is under clearance for makeup water supply valve problems
- Boric Acid Transfer Pump B-SB is under clearance for motor replacement

Turnover: Plant is operating at 100% steady state power. Maintain present conditions.

Critical Tasks:

- Isolate AFW flow to the ruptured 'C' SG prior to entering ECA-3.1, SGTR with Loss of Reactor Coolant: Subcooled Recovery
- Isolate 1MS-72 Prior to entering ECA-3.1, SGTR with Loss of Reactor Coolant: Subcooled Recovery
- Shut 'A' and 'B' MSIV's prior to exiting E-2, Faulted SG Isolation

| Event No. | Malf. No. | Event Type* | Event Description |
|---|---|---|---|
| 1 | prs06a | C – RO/SRO<br>TS – SRO | Pressurizer PORV 445A Leakage |
| 2 | nis03a | I – BOP/SRO | NI-31 high voltage block failure and energization at power (OP-105) |
| 3 | eps05a<br>zdsq94:6a | C – BOP/SRO<br>TS – SRO | Loss of 1A-SA Emergency Bus with failure of 'A' CSIP Room HVAC to restart (AOP-025) |
| 4 | ccw01a<br>ccw047 | C – RO/SRO<br>TS – SRO | Trip of 'A' CCW Pump on O/C during 'A' EDG sequencer start with standby CCW pump failure to auto start (AOP-014) |
| 5 | pt:308c | R – RO/SRO<br>N – BOP/SRO | SG 'C' PORV Pressure Instrument fails high and the PORV stays open requiring the crew to reduce power below 100%. (AOP-042) |
| 6 | sgn05c | M – ALL | 'C' Steam Generator tube rupture ~ 420 gpm (AOP-016) |
| 7 | mss11 | M – ALL | Main Steam Header break outside Containment (downstream of MSIVs) |
| 8 | zrpk504a<br>zrpk504a<br>mss05c | C – BOP/SRO | Main Steam Line Isolation Signal Fails, 'C' MSIV fails to shut |

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor



Rec'd 3/25/13

## SCENARIO SUMMARY: 2013 NRC EXAM SCENARIO 3

The plant is operating at ~100% power in MOL. When turnover is complete the crew will be directed to maintain the current plant conditions.

The following equipment is under clearance:

- 'B' MDAFW Pump is under clearance for pump packing repairs. The pump has been inoperable for 12 hours and will be restored to operable status within the next 24 hours. Tech Spec 3.7.1.2 LCO Action **a** and Tech Spec 3.3.3.5.b Action **c** applies.

PLANT SYSTEMS

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three independent steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE with:

a. Two motor-driven auxiliary feedwater pumps, each capable of being powered from separate emergency buses, and

b. One steam turbine-driven auxiliary feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

INSTRUMENTATION

REMOTE SHUTDOWN SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.3.5.b All transfer switches, Auxiliary Control Panel Controls and Auxiliary Transfer Panel Controls for the OPERABILITY of those components required by the SHNPP Safe Shutdown Analysis to (1) remove decay heat via auxiliary feedwater flow and steam generator power-operated relief valve flow from steam generators A and B, (2) control RCS inventory through the normal charging flow path, (3) control RCS pressure, (4) control reactivity, and (5) remove decay heat via the RHR system shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

c. With one or more inoperable Remote Shutdown System transfer switches, power, or control circuits required by 3.3.3.5.b, restore the inoperable switch(s)/circuit(s) to OPERABLE status within 7 days, or be in HOT STANDBY within the next 12 hours.



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## SCENARIO SUMMARY: 2013 NRC EXAM SCENARIO 3 continued

- 'B' Condenser Vacuum Pump is under clearance for makeup water supply valve problems
- 1SI-3, Boron Injection Tank Outlet valve is under clearance for breaker repairs. Tech Spec 3.5.2 Action **a** applies. OWP-SI-01 has been completed.

EMERGENCY CORE COOLING SYSTEMS

3/4.5.2 ECCS SUBSYSTEMS $T_{avg}$ GREATER THAN OR EQUAL TO 350°F

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent Emergency Core Cooling System (ECCS) subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE Charging/safety injection pump,

b. One OPERABLE RHR heat exchanger,

c. One OPERABLE RHR pump, and

d. An OPERABLE flow path capable of taking suction from the refueling water storage tank on a Safety Injection signal and, upon being manually-aligned, transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

- Boron Injection Pump B-SB is under clearance for motor replacement. Tech Spec 3.1.2.2 applies (tracking only). OWP-CS-05 has been completed.



Rec'd 3/25/13

## SCENARIO SUMMARY: 2013 NRC EXAM SCENARIO 3 continued

**Event 1**: Pressurizer PORV 445A Leakage: Pressurizer PORV 445A leakage. This failure will cause PRZ PORV 445A to leak, resulting in rising PRT pressure and level. PORV Line Temp indicator TI-463 will increase as observed on the MCB and the crew will respond IAW ALB 009-8-2, PRESSURIZER RELIEF DISCHARGE HIGH TEMP. The crew may utilize AOP-016 Attachment 5 to determine which PORV is leaking. The SRO will evaluate Tech Spec 3.4.4, Reactor Coolant System – Relief Valves.

TS 3.4.4 applicable LCO is Action **a,** 1 hour to restore

REACTOR COOLANT SYSTEM

3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.4 All power-operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3

ACTION:

a. With one or more PORV(s) inoperable, because of excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) with power maintained to the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.



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## SCENARIO SUMMARY: 2013 NRC EXAM SCENARIO 3 continued

**Event 2:** NI-31 high voltage block failure and energization at power (OP-105). The high voltage block on NI-31 will fail and cause the Source Range to suddenly energize and the audio count rate to become audible. The crew should respond to the failure by implementing OP-105, Excore Nuclear Instrumentation, Section 8.2 – Inadvertent Source Range Detector Energization at Power. They will promptly de-energize NI-31 by removing the 118V 5A instrument power fuse. The SRO should refer to TS 3.3.1 (no action required above P-6) and direct the implementation of OWP-RP-19.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the Reactor Trip System instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-1.

ACTION: As shown in Table 3.3-1.

| FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| 6. Source Range, Neutron Flux | | | | | |
| a. Startup | 2 | 1 | 2 | 2## | 4 |
| b. Shutdown | 2 | 1 | 2 | 3, 4, 5 | 5 |

##Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.



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## SCENARIO SUMMARY: 2013 NRC EXAM SCENARIO 3 continued

**Event 3**: Loss of 1A-SA Emergency Bus with failure of CSIP 'A' Room HVAC to restart (AOP-025). Emergency Bus A SA to Aux Bus D Tie Breaker 105 SA trips open causing a loss of power to the Emergency Bus A. The Emergency Diesel Generator 'A' starts and loads the bus. The crew will enter AOP-025 and perform the immediate action of checking any CSIP running. The crew should NOT perform the RNO action to isolate letdown since guidance is provided in AOP-025 Basis Document stating that this should only be done if the sequencer does not start the CSIP. IF the letdown flow is secured then the crew will have to restore letdown IAW OP-107. During the recovery steps of AOP-025 the BOP should identify that the CSIP 'A' room HVAC is NOT running. Since this is a "Verify" step the BOP should start the fan. The SRO will be directed by AOP-025 to review multiple Tech Specs due to the loss of power to Tech Spec related systems. Tech Spec 3.0.3 is the most limiting action due to the isolation of the CNMT vacuum relief valves caused by 2/4 Radiation monitors failing high after losing power.

☐4. **REFER TO** the following Tech Specs:

- 3.0.3 (Due to loss of 2/4 containment rad monitors and CVIS affect on CNMT vacuum reliefs)
- 3.3.3.1 Radiation Monitoring for Plant Operations (Due to inoperable Control Room Outside Air Intake Monitors)
- 3.4.6.1 RCS Leak Detection (Due to RM-3502A inop)
- 3.6.5 Vacuum Relief System
- 3.8.1.1 AC Sources Operating
- 3.8.1.2 AC Sources Shutdown
- 3.8.2.1 DC Sources Operating
- 3.8.3.1 Onsite Power Distribution - Operating
- 3.8.3.2 Onsite Power Distribution - Shutdown



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## SCENARIO SUMMARY: 2013 NRC EXAM SCENARIO 3 continued

**Event 4**: Trip of "A" CCW Pump on O/C during EDG 'A' sequencer start with standby CCW pump failure to auto start (AOP-014). During the loss of power to 1A-SA Emergency Bus the 'A' CCW Pump will be started by the sequencer and after 20 seconds it will trip on overcurrent. The standby 'B' CCW Pump fails to Auto Start due to a pressure transmitter failure (instrument is isolated therefore pressure decrease is not sensed). The crew should recognize the loss and enter AOP-014, Loss of Component Cooling Water in conjunction with AOP-025. AOP-025 additionally, has direction to recover the CCW pump IAW AOP-014. AOP-014 will direct the restoration of the CCW system. The RO will be directed by the SRO to manually start the 'B' CCW (or will have started it IAW OPS-NGGC-1000 when it did not auto start). The SRO should also evaluate Tech Spec 3.7.3, Component Cooling Water System.

PLANT SYSTEMS

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3 At least two component cooling water (CCW) pumps*, heat exchangers and essential flow paths shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With only one component cooling water flow path OPERABLE, restore at least two flow paths to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

*The breaker for CCW pump 1C-SAB shall not be racked into either power source (SA or SB) unless the breaker from the applicable CCW pump (1A-SA or 1B-SB) is racked out.



Rec'd 3/25/13

## SCENARIO SUMMARY: 2013 NRC EXAM SCENARIO 3 continued

**Event 5**: SG 'C' PORV Pressure Instrument fails high and the PORV stays open requiring the crew to reduce power below 100% (AOP-042). A transmitter failure will cause the 'C' SG PORV to fail 100% open. Steam Generator 'C' PORV Pressure transmitter PT-308c fails high causing the 'C' SG PORV to open in automatic. The crew should identify this failure by annunciator ALB-014-8-5, Computer Alarm Steam Generators alarming and status light indications for the 'C' SG PORV. Note: The PT-308c does not have MCB indications. The BOP will be directed by the SRO to take manual control of the PORV and shut it. The valve control will NOT respond and an Aux Operator will be dispatched to locally shut the isolation valve to the PORV. During this time Reactor Power will exceed 100%. The combination of the steam leak and Reactor power will cause the crew to enter AOP-042, Secondary Steam Leak/Efficiency Loss. AOP-042 is used to rapidly reduce Reactor power to < 100% at a ramp rate of up to 45 MW/min. When Reactor power has been stabilized below 100% the AO that was dispatched to locally isolate the PORV will report back that the valve has been shut. When power is stabilized to the satisfaction of the Lead Examiner event 2 can be introduced.
6

The SRO should evaluate Tech Specs 3.6.3, Containment Isolation Valves and PLP-106 Technical Specification Equipment List Program and Core Operating Limits Report. If the Tech Specs are not referred to during the scenario then if required ask a follow up question at the end of the scenario dealing with the LCO.

TS 3.6.3 – Action **c**, isolate the affected penetration within 4 hours. The redundant manual isolation valve per PLP-106 is Containment Isolation valve 1MS-63.

CONTAINMENT SYSTEMS

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 Each containment isolation valve specified in the Technical Specification Equipment List Program, plant procedure PLP-106, shall be OPERABLE with isolation times less than or equal to required isolation times.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one or more of the containment isolation valve(s) inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and:

a. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or

b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position, or

c. Isolate each affected penetration within 4 hours by use of at least one closed manual valve or blind flange, or

d. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

Rec'd 3/25/13

## SCENARIO SUMMARY: 2013 NRC EXAM SCENARIO 3 continued

**Event 6**: 'C' Steam Generator tube rupture ~ 420 gpm (AOP-016). 'C' Steam Generator (SGTR) one tube sheared. Break flow of 420 gpm. The crew should recognize the presence of a large leak in the primary and announce entry into AOP-016. Due to the leak size the crew will promptly recognize that the leak is beyond CVCS makeup capability and the RCS pressure is being rapidly reduced. Prior to reaching the Reactor trip setpoint on Pressurizer Low Pressure (1960 psig) they should manually trip the Reactor, carry out the immediate actions of E-0 and time permitting manually initiate Safety Injection (an automatic Safety Injection may occur if actions are not promptly taken).

**Event 7**: Main Steam Line Header break outside Containment. Five minutes after the Reactor is tripped a Main Steam line break on the main steam header outside Containment will occur. It is expected that the crew transition from E-0 to E-3 to address the ruptured Steam Generator. While in E-3 the faulted Steam Generator will become apparent by the rapid reduction in Steam Generator pressure.

**Event 8**: Main Steam Line Isolation Signal Fails, 'C' MSIV fails to shut. The crew should attempt to manually actuate the Main Steam Line Isolation due to approaching the criteria 'Any SG Pressure - Less Than or Equal to 601 psig'. With this failure the crew will attempt to shut the MSIV's manually on the MCB, but only 'A' and 'B' MSIV can be manually shut. 'C' MSIV will not shut from the MCB or locally. The crew should use the Secondary Integrity Foldout Criteria to address the faulted 'C' Steam Generator and transition to E-2, Faulted Steam Generator Isolation, (if isolation attempts were not performed during the implementation of E-3). After entry into E-2 for "C"' Steam Generator isolation the crew will return to E-3.

The scenario will end after the crew transitions to ECA-3.1, SGTR with Loss of Reactor Coolant: Subcooled Recovery and initiates an RCS Cooldown.



Rec'd 3/25/13

## CRITICAL TASK JUSTIFICATION:

1. Isolate AFW flow to the ruptured 'C' Steam Generator prior to entering ECA-3.1, SGTR with Loss of Reactor Coolant: Subcooled Recovery.

   Failure to isolate the ruptured SG causes a loss of differential pressure between the ruptured SG and the intact SGs. Upon a loss of differential pressure, the crew must transition to a contingency procedure that constitutes an incorrect performance that necessitates the crew taking compensating action which complicates the event mitigation strategy. This critical task requires the crew to isolate the feedwater flow to the ruptured SG. Any delay in the isolation of feedwater adds additional inventory along with the primary to secondary leakage. If the primary to secondary leakage is not stopped, SG inventory increase leads to water release through the PORV or safety valve which has the potential for an unmonitored radiation release. Continued filling of the SG could lead to SG overfill which could fill the SG steam lines with water and potentially cause a steam line pipe failure due to the additional weight of the water in the steam lines.

2. Isolate 1MS-72 prior to entering ECA-3.1, SGTR with Loss of Reactor Coolant: Subcooled Recovery.

   Failure to isolate the ruptured SG causes a loss of differential pressure between the ruptured SG and the intact SGs. Upon a loss of differential pressure, the crew must transition to a contingency procedure that constitutes an incorrect performance that necessitates the crew taking compensating action which complicates the event mitigation strategy. This critical task requires the crew to isolate the steam flow from the ruptured SG. Isolation of the ruptured SG is necessary because the crew cannot start RCS cooldown to establish RCS subcooling margin. Delaying isolation of the ruptured SG steam flow further depressurizes the ruptured SG increasing the differential pressure between the RCS and ruptured SG requiring the RCS to be depressurized to a lower value. This delay increases the duration of the primary to secondary leakage.

3. Shut 'A' and 'B' MSIV's prior to exiting E-2, Faulted SG Isolation.

   Failure to isolate a faulted Steam Generator that can be isolated causes challenges to the Critical Safety Functions beyond those irreparably introduced by the postulated conditions. Also, depending upon the plant conditions, it could constitute a demonstrated inability by the crew to recognize a failure of the automatic actuation of an ESF system or component. This critical task requires the crew to recognize an automatic actuation should have occurred of an ESF system or component but has not and then take manual operator actions to perform the isolation.



Rec'd 3/25/13

| | | | |
|---|---|---|---|
| Facility: | SHEARON-HARRIS | Scenario No.: 4 | Op Test No.: 05000400/2013301 |
| Examiners: | ______ | Operators: | SRO: |
| | ______ | | RO: |
| | ______ | | BOP: |

Initial Conditions:

- IC-19, MOL, 100% power
- 'B' MD AFW Pump is under clearance for pump packing replacement
- 1SI-3, Boron Injection Tank Outlet valve is under clearance for breaker repairs
- 'B' Condenser Vacuum Pump is under clearance for makeup water supply valve problems
- Boric Acid Transfer Pump B-SB is under clearance for motor replacement

Turnover:

- A plant shutdown is required due to problems encountered during the repairs on the 'B' MDAFW pump. Repairs will not be able to be completed prior to the LCO expiring. The plant is operating at ~100% power in MOL. When turnover is complete a power reduction at 4 DEH units/min must be started to support being in Mode 3 within the next 6 hours. All required notifications have been made to individuals concerning the reason for the shutdown.

Critical Task:

- Open 1MS-70 or 1MS-72 to establish a minimum of 210 KPPH AFW flow to the Steam Generators prior to exiting ECA-0.0
- Energize "B" AC emergency bus when offsite power becomes available prior to aligning equipment for extended power loss (step 11 of ECA-0.0)

| Event No. | Malf. No. | Event Type* | Event Description |
|---|---|---|---|
| 1 | N/A | R – RO/SRO<br>N – BOP/SRO | Plant Shutdown (GP-006) |
| 2 | nis08b | I – RO/SRO<br>TS – SRO | PR NIS Channel N-42 fails HIGH (AOP-001) |
| 3 | gen01 | C – BOP/SRO | Generator Voltage Regulator Failure (APP-ALB-022) |
| 4 | lt:459 | I – RO/SRO<br>TS – SRO | Controlling Pressurizer Level Channel, LT-459, fails HIGH (APP-ALB-009) |
| 5 | hva04 | C – BOP/SRO<br>TS – SRO | "A" Emergency Services Chilled Water Pump Trip (AOP-026) |
| 6 | cfw17b | C – RO/SRO | Main Feedwater Pump 1B Recirculation Valve (1FW-39) fails OPEN |
| 7 | eps01a | M – ALL | Loss of Offsite Power, Reactor Trip |
| 8 | dsg38, dsg05b | C – BOP/SRO | EDG 'A' output breaker trips prior to Load Block 9<br>EDG 'B' fails to start<br>Loss of ALL power / restoration possible with offsite power |
| 9 | z1974tdi<br>z1975tdi | C – BOP/SRO | 1MS-70 and 1MS-72 fail to auto open<br>(Loss of all AFW until operator opens 1MS-70 or 72) |

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor



Rec'd 2/25/13

## SCENARIO SUMMARY: 2013 NRC EXAM SCENARIO 4

A plant shutdown is required due to problems encountered during the repairs on the 'B' MDAFW Pump. Repairs will not be able to be completed prior to the LCO expiring. The plant is operating at ~100% power in MOL. When turnover is complete a power reduction at 4 DEH units/min must be started to support being in Mode 3 within the next 6 hours. All required notifications have been made to individuals concerning the reason for the shutdown.

The following equipment is under clearance:

- 'B' MDAFW Pump is under clearance for pump packing repairs. The pump has been inoperable for 12 hours and will be restored to operable status within the next 24 hours. Tech Spec 3.7.1.2 LCO Action **a** and Tech Spec 3.3.3.5.b Action **c** applies.

PLANT SYSTEMS

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three independent steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE with:

a. Two motor-driven auxiliary feedwater pumps, each capable of being powered from separate emergency buses, and

b. One steam turbine-driven auxiliary feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

INSTRUMENTATION

REMOTE SHUTDOWN SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.3.5.b All transfer switches, Auxiliary Control Panel Controls and Auxiliary Transfer Panel Controls for the OPERABILITY of those components required by the SHNPP Safe Shutdown Analysis to (1) remove decay heat via auxiliary feedwater flow and steam generator power-operated relief valve flow from steam generators A and B, (2) control RCS inventory through the normal charging flow path, (3) control RCS pressure, (4) control reactivity, and (5) remove decay heat via the RHR system shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

c. With one or more inoperable Remote Shutdown System transfer switches, power, or control circuits required by 3.3.3.5.b, restore the inoperable switch(s)/circuit(s) to OPERABLE status within 7 days, or be in HOT STANDBY within the next 12 hours.



Rec'd 3/25/13

## SCENARIO SUMMARY: 2013 NRC EXAM SCENARIO 4 continued

The following equipment is under clearance (continued):

- 'B' Condenser Vacuum Pump is under clearance for makeup water supply valve problems
- 1SI-3, Boron Injection Tank Outlet valve is under clearance for breaker repairs. Tech Spec 3.5.2 Action **a** applies. OWP-SI-01 has been completed.

EMERGENCY CORE COOLING SYSTEMS

3/4.5.2 ECCS SUBSYSTEMS - $T_{avg}$ GREATER THAN OR EQUAL TO 350°F

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent Emergency Core Cooling System (ECCS) subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE Charging/safety injection pump,

b. One OPERABLE RHR heat exchanger,

c. One OPERABLE RHR pump, and

d. An OPERABLE flow path capable of taking suction from the refueling water storage tank on a Safety Injection signal and, upon being manually aligned, transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

- Boron Injection Pump B-SB is under clearance for motor replacement. Tech Spec 3.1.2.2 applies (tracking only). OWP-CS-05 has been completed.



Rec'd 3/25/13

## SCENARIO SUMMARY: 2013 NRC EXAM SCENARIO 4 continued

**Event 1**: Plant Shutdown (GP-006). Crew performs a power reduction IAW GP-006. For this reactivity manipulation it is expected that the SRO will conduct a reactivity brief, the RO will borate per the reactivity plan and the BOP will operate the DEH Controls as necessary to lower power.

**Event 2**: PRNIS Channel N-42 fails HIGH (AOP-001). This malfunction will cause rods to start stepping in at maximum speed (72 steps per minute). The crew should respond by entering AOP-001, Malfunction of Rod Control and Indication System and perform the immediate actions which will be placing the Rod Control selector switch to MANUAL. The crew will then perform the follow up actions of AOP-001, implement OWP-RP-24 and OP-104 Section 5.5 in order to restore Rod Control to automatic. The SRO will evaluate Tech. Spec 3.3.1 for any impact due to the failed instrument.

| | FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|---|
| 1. | Manual Reactor Trip | 2<br>2 | 1<br>1 | 2<br>2 | 1, 2<br>3*, 4*, 5* | 1<br>9 |
| 2. | Power Range, Neutron Flux<br>a. High Setpoint<br>b. Low Setpoint | <br>4<br>4 | <br>2<br>2 | <br>3<br>3 | <br>1, 2<br>1###, 2 | <br>2<br>2 |
| 3. | Power Range, Neutron Flux High Positive Rate | 4 | 2 | 3 | 1, 2 | 2 |
| 4. | Power Range, Neutron Flux, High Negative Rate | 4 | 2 | 3 | 1, 2 | 2 |

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours,

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1, and

c. Either, THERMAL POWER is restricted to less than or equal to 75% of RATED THERMAL POWER and the Power Range Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER within 4 hours; or, the QUADRANT POWER TILT RATIO is monitored at least once per 12 hours per Specification 4.2.4.2.

**Event 3**: Generator Voltage Regulator Failure (APP-ALB-022). The voltage regulator failure will cause generator MVARS to rise above the normal control band. ALB-22-9-4 Computer Alarm Gen/Exciter Systems, ALB-22-4-5 Generator Exciter Field Forcing and ERFIS indications will alert the operators to this condition if not detected earlier by changes in generator MVARS. Annunciator guidance will have the BOP operator attempting to control voltage with the voltage regulator in MANUAL, but attempts for this type of control will fail requiring the base adjuster to be used to reduce MVARs to a value within normal operational limits (75 to 175 MVARs). This failure will also require the crew to notify the Load Dispatcher that the voltage regulator is in Manual control within 30 minutes.



Rec'd 3/25/15

## SCENARIO SUMMARY: 2013 NRC EXAM SCENARIO 4 continued

**Event 4**: Controlling Pressurizer Level Channel, LT-459, fails HIGH (APP-ALB-009). The crew should respond in accordance with alarm response procedure APP-ALB-09-4-2 and window 2-2. The crew should take Charging FCV-122 to Manual and maintain pressurizer level within the control bands and trip limits of OMM-001 Attachment 13. They will shift level control to an alternate channel. The SRO will evaluate Tech. Spec 3.3.1 for any impact due to the failed instrument.

TS 3.3.1 As a minimum the Reactor Trip System instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE

Table 3.3-1

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

| FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| 11. Pressurizer Water Level--High (Above P-7) | 3 | 2 | 2 | 1 | 6 |

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours, and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1.



Rec'd 3/25/13

## SCENARIO SUMMARY: 2013 NRC EXAM SCENARIO 4 continued

**Event 5**: "A" Emergency Services Chilled Water Pump Trip (AOP-026). The crew will respond to various alarms on ALB-023, diagnose the event, and enter AOP-026, Loss of Essential Chill Water System (no immediate actions). The SRO will direct the BOP to start the 'B' Train ESCWS Chiller IAW OP-148, Essential Service Chilled Water System. The crew should implement OWP-ECW-01 for the ESCW Chiller 1A-SA failure. The SRO should evaluate Tech Spec 3.7.13, Essential Services Chilled Water System and PLP-114, Relocated Technical Specifications and Design Basis Requirements – Attachment 4 for Area Temperature Monitoring. Note that the 'A' Chiller will be inoperable for the remainder of the scenario and this will impact plant response during the Major Event in that this failure will prevent Load Block 9 on sequencer Train 'A' from energizing.

TS 3.7.13

PLANT SYSTEMS

3/4.7.13 ESSENTIAL SERVICES CHILLED WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.13 At least two independent Essential Services Chilled Water System loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With only one Essential Services Chilled Water System loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

PLP-114 Attachment 4

Attachment 4
Sheet 1 of 9

Area Temperature Monitoring

1.0 OPERATIONAL REQUIREMENTS

1.1 The temperature of each area shown in Table A shall not be exceeded for more than 8 hours or by more than 30°F.

APPLICABILITY: Whenever the equipment in an affected area is required to be OPERABLE.

ACTION:

a. With one or more areas exceeding the temperature limit(s) shown in Table A for more than 8 hours, prepare within 30 days an evaluation to demonstrate the continued OPERABILITY of the affected equipment.

b. With one or more areas exceeding the temperature limit(s) shown in Table A by more than 30°F, prepare an evaluation as required by Action a. above and within 4 hours either restore the area(s) to within the temperature limit(s) or declare the equipment in the affected area(s) inoperable.



Rec'd 3/25/13

## SCENARIO SUMMARY: 2013 NRC EXAM SCENARIO 4 continued

**Event 6**: Main Feedwater Pump 1B Recirculation Valve (1FW-39) fails OPEN - The crew should identify that the 1B FW pump recirc valve has failed open by MCB light changes from green to red, FW discharge pressure changes, SG Feedflow/Steam flow changes, SG level trends on the ERFIS computer screen displays and by level trends on the WR and NR level recorders. The BOP may attempt to close the valve when the incorrect position is observed but the valve will not close from the MCB. The crew may dispatch the Turbine Building AO immediately or when directed by AOP-010, Feedwater Malfunctions. When the crew enters AOP-010 they will initially perform the immediate action to verify that a FW pump has not tripped. The SRO work through procedure steps to have the recirc valve manually closed. The AO will not be successful with shutting the recirc valve and all SG levels will reach OMM-001 trip limits of 30% within approximately 5 minutes. When the Reactor trip is activated event 7 will be automatically inserted.

**Event 7 (Major)**: Loss of Offsite Power, Reactor Trip – Once the crew has activated the Reactor trip switch and the Reactor trip breakers open a loss of Offsite Power will occur. The crew will identify that Offsite power has occurred which leads to a loss of all AC when Event 8 occurs.

**Event 8**: EDG 'A' output breaker trips prior to Load Block 9, EDG 'B' fails to start, Loss of ALL power / restoration possible with offsite power. 'A' EDG will start but the output breaker will trip 30 seconds after energizing the bus. Additionally, Load Block 9 would not have been reached due to Event 5 when the "A" Chiller tripped.

The crew should enter ECA-0.0, Loss of All AC Power and perform the immediate actions of verifying a Reactor and Turbine trip. The crew will continue efforts to restore power to the station. The Load Dispatcher will inform the crew the fault that has caused the Offsite power failure has been isolated and power has been restored to the switch yard. The Load Dispatcher call is critical and should be performed while the crew is evaluating the status of the diesel generators so they may make a decision to restore offsite power in step 9. If the call for offsite power restoration is delayed the crew could continue with ECA-0.0 extended power loss recovery. They should be allowed to make a decision to either restore offsite power or line up systems for extended power loss (the second decision would not be appropriate). The crew should determine that the 'B' Emergency Bus is the best choice to restore power to since the 'A' diesel generator output breaker has tripped and an evaluation of the cause has not been completed. The crew should restore offsite power to the 'B' Emergency Bus IAW EOP-ECA-0.0 Attachment 1.



Rec'd 3/25/15

## SCENARIO SUMMARY: 2013 NRC EXAM SCENARIO 4 continued

**Event 9**: 1MS-70 and 1MS-72 fail to auto open (Loss of all AFW until operator opens 1MS-70 or 72). 'B' MD AFW Pump is under clearance and 'A' MD AFW Pump will not have power. The Turbine Driven AFW pump will not start due to failures on both 1MS-70 and 1MS-72 to automatically open. The loss of all FW to the Steam Generators will create a RED path on Heat Sink (FR-H.1). With a loss of AC Power ECA-0.0 is in effect. A caution prior to step 1 of ECA-0.0 states: Critical Safety Function Status Trees should be monitored for information only. Function Restoration Procedures should NOT be implemented unless directed by this procedure. The crew should remain in ECA-0.0 and NOT transition to FR-H.1 when a RED path exists. The crew should identify that there is no Feedwater flow to the SG's and open either 1MS-70 or 1MS-72 to establish a motive force to run the Turbine Driven AFW pump. Once flow has been established the TDAFW pump speed controller should be manually adjusted to obtain a minimum of 210 KPPH AFW flow to the Steam Generators.

The scenario is terminated when the crew transitions from ECA-0.0 to E-0, Reactor Trip or Safety Injection then transition from E-0 to ES-0.1, Reactor Trip Response.

## CRITICAL TASK JUSTIFICATION:

1. Open 1MS-70 or 1MS-72 to establish a minimum of 210 KPPH AFW flow to the Steam Generators prior to exiting ECA-0.0

    Failure to establish the minimum required AFW flow results in adverse consequences or significant degradation in the mitigative capability of the plant. This critical task requires the crew to recognize an automatic actuation of an ESF system or component should have occurred but has not and then take manual operator actions to restore the required flow.

2. Energize "B" AC emergency bus when offsite power becomes available prior to aligning equipment for extended power loss (step 11 of ECA-0.0).

    Failure to energize an AC emergency bus constitutes mis-operation or incorrect crew performance which leads to degraded emergency power capacity. Failure to perform this task also results in the needless degradation of a barrier to fission product release via the RCP seals. Energize at least one AC emergency bus before transition out of E-0, unless the transition is to ECA-0.0, in which case the critical task must be performed before placing safeguards equipment hand switches in the pull-to-lock position. For Harris station safeguards equipment cannot be placed in pull-to-lock so the task would be to energize the emergency bus prior to aligning equipment for extended power loss and locally de-energizing control power to the ESF pumps.



Rec'd 3/25/13

Facility: SHEARON-HARRIS Scenario No.: 5 Op Test No.: 05000400/2013301

Examiners: ____ ____ ____ Operators: SRO: RO: BOP:

Initial Conditions: IC-27, MOL, ~4% power

- Plant startup to full power on HOLD until 'B' Condensate Booster Pump is in service
- 'B' Condenser Vacuum Pump is under clearance for makeup water supply valve problems
- 1SI-3, Boron Injection Tank Outlet valve is under clearance for breaker repairs
- Boric Acid Transfer Pump B-SB is under clearance for motor replacement
- GP-005, Power Operation, step 95.c

Turnover:

- Power ascension is on hold for 'B' Condensate Booster pump oil system repairs. Repairs are now completed and the pump is ready for service.
- Start the Second 'B' Condensate Booster Pump IAW OP-134 Section 5.6.

Critical Tasks:

- With RCS pressure < 1400 psig, establish SI flow of >200 gpm using alternate high head safety injection to cold legs prior to securing RCPs
- Manually actuate Main Steam Line Isolation prior to Containment pressure exceeding 10 psig

| Event No. | Malf. No. | Event Type* | Event Description |
|---|---|---|---|
| 1 | N/A | N – BOP/SRO | Start the 'B' Condensate Booster Pump |
| 2 | tt:144<br>jtb143b | I – RO/SRO | Letdown Temperature Controller fails LD/Diversion Valve fails to bypass demineralizers |
| 3 | lt:496 | C – BOP/SRO<br>TS - SRO | Controlling 'C' Steam Generator Level Transmitter, LT-496, fails low |
| 4 | jfb7579<br>z2715tic | C-BOP/SRO<br>TS - SRO | AH-39 Containment Fan Coil Unit fan trip with back up auto start failure ('C' RCP cooling fan) |
| 5 | ccw08a | C – RO/SRO | Component Cooling Water system leak requiring AOP-014 entry and manual makeup to maintain level |
| 6 | rcs09a | C – RO/SRO | RCP "A" rising vibration requires manual Reactor trip and securing "A" RCP and associated PRZ spray valve after E-0 immediate actions are completed |
| 7 | rcs18a | M- ALL | SBLOCA inside containment (100% severity) |
| 8 | sis017<br>sis018 | C – RO/SRO | Failure of BIT outlet valve 1SI-4 to open requiring alternate high head injection flow path use |
| 9 | N/A | C – RO/SRO | Manually trip "B" and "C" RCP when RCP trip criteria are met IAW E-0 foldout |
| 10 | zrpk504a<br>zrpk504b | C – BOP/SRO | Failure of automatic Main Steam Line Isolation to occur when Containment pressure exceeds 3 psig |

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor



Rec'd 3/25/13

## SCENARIO SUMMARY: 2013 NRC EXAM SCENARIO 5

**Low power scenario** – Turnover to crew is the unit startup on hold. The plant is in Mode 2 with Reactor power less than 5%. Power ascension was on hold while 'B' Condensate Booster pump oil system leak repairs are completed. The repairs have just been completed and the pump checkout is completed and ready to be started. When the crew takes the shift the expectation is to start the 'B' Condensate Booster pump IAW OP-134, Condensate System, Section 5.6. After the pump is running they will continue with GP-005, Power Operation, to obtain rated power conditions.

The following equipment is under clearance:

- 'B' Condenser Vacuum Pump is under clearance for makeup water supply valve problems.
- 1SI-3, Boron Injection Tank Outlet valve is under clearance for breaker repairs. Tech Spec 3.5.2 Action **a** applies. OWP-SI-01 has been completed.

EMERGENCY CORE COOLING SYSTEMS

3/4.5.2 ECCS SUBSYSTEMS - $T_{avg}$ GREATER THAN OR EQUAL TO 350°F

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent Emergency Core Cooling System (ECCS) subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE Charging/safety injection pump,

b. One OPERABLE RHR heat exchanger,

c. One OPERABLE RHR pump, and

d. An OPERABLE flow path capable of taking suction from the refueling water storage tank on a Safety Injection signal and, upon being manually aligned, transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

- Boron Injection Pump B-SB is under clearance for motor replacement. Tech Spec 3.1.2.2 applies (tracking only). OWP-CS-05 has been completed.



Rec'd 3/25/13

## SCENARIO SUMMARY: 2013 NRC EXAM SCENARIO 5 continued

**Event 1**: Start the 'B' Condensate Booster Pump. Upon turnover and assuming the shift the crew will start the 'B' Condensate Booster pump IAW OP-134, Condensate System, Section 5.6 "Second Condensate Booster Pump Start up". After the pump is in operation the crew will discuss raising power IAW GP-005 to prepare to place the Main Feedwater Regulating valves in service. Prior to the power increase event 2 will occur.

**Event 2**: Letdown Temperature Controller fails LD/Diversion Valve fails to bypass demineralizers. This failure will cause temperature controller TK-144 output to decrease to zero. Without cooling to the letdown heat exchanger, temperatures observed on TI-143 will increase. At 135°F annunciator ALB-07-3-2, Demin Flow Diversion High Temp will alarm. The crew should respond IAW the alarm procedure. The RO should identify that the divert valve to the VCT has failed to respond. The RO should report the failure to the SRO. The SRO should direct manually bypassing the CVCS Demineralizers, and should also provide directions to the RO to restore letdown temperature to normal utilizing MANUAL control of TK-144. The SRO should provide a temperature band to the RO IAW OMM-001, Conduct of Operations, for operation of components in manual. The SRO can find this temperature band guidance in OP-107. With TK-144 controller not in auto the temperature band should be from 110 – 120°F. The CVCS Demineralizers should remain bypassed pending an evaluation for continued resin use. Soon after stabilizing from this temperature controller failure event 3 will occur.

**Event 3**: Controlling 'C' Steam Generator Level Transmitter, LT-496, fails low. The BOP should respond to multiple 'C' Steam Generator alarms on ALB-014 and take manual control of the 'C' FRV Bypass valve in accordance with the alarm response procedures and OMM-001, Conduct of Operations. The SRO should evaluate the following Tech Specs for failure of LT-496:

T.S. 3.3.1: As a minimum, the Reactor Trip System instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE

REACTOR TRIP SYSTEM INSTRUMENTATION

| FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| 13. Steam Generator Water Level--Low-Low | 3/stm. gen. | 2/stm. gen. in any operating stm. gen. | 2/stm. gen. each operating stm. gen. | 1, 2 | 6(1) |
| 14. Steam Generator Water Level--Low Coincident With Steam/ Feedwater Flow Mismatch | 2 stm. gen. level and 2 stm./feed-water flow mismatch in each stm. gen. | 1 stm. gen. level coincident with 1 stm./feedwater flow mismatch in same stm. gen. | 1 stm. gen. level and 2 stm./feed-water flow mismatch in same stm. gen. or 2 stm. gen. level and 1 stm./feedwater flow mismatch in same stm. gen. | 1, 2 | 6 |

(1)The applicable MODES for these channels noted in Table 3.3-3 are more restrictive and, therefore, applicable.

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## SCENARIO SUMMARY: 2013 NRC EXAM SCENARIO 5 continued

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours, and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1.

T.S. 3.3.2: The Engineered Safety Features Actuation System (ESFAS) instrumentation channels and interlocks shown in Table 3.3-3 shall be OPERABLE with their Trip Setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4.

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

| FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| 5. Turbine Trip and Feedwater Isolation | | | | | |
| b. Steam Generator Water Level--High-High (P-14) | 4/stm. gen. | 2/stm. gen. in any stm. gen. | 3/stm. gen. in each stm. gen. | 1, 2 | 19 |
| 6. Auxiliary Feedwater | | | | | |
| c. Steam Generator Water Level--Low-Low | | | | | |
| 1) Start Motor-Driven Pumps | 3/stm. gen. | 2/stm. gen. in any stm. gen. | 2/stm. gen. in each stm. gen. | 1, 2, 3 | 19 |
| 2) Start Turbine-Driven Pump | 3/stm. gen. | 2/stm. gen. in any 2 stm. gen. | 2/stm. gen. in each stm. gen. | 1, 2, 3 | 19 |

ACTION STATEMENTS (Continued)

ACTION 19 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours, and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.2.1.

The OWP is not required to be implemented in order to continue with the scenario. If the crew allows SG levels to decrease to < 30% they will be required to perform a manual Reactor Trip.



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## SCENARIO SUMMARY: 2013 NRC EXAM SCENARIO 5 (continued)

**Event 4**: AH-39 Containment Fan Coil Unit fan trip with back up auto start failure ('C' RCP cooling fan). The fan failure will cause annunciator ALB-029 4-5 "Containment Fan Coolers AH-39 Low Flow-O/L to alarm. The crew should identify that the standby fan did not auto start and start the standby fan. There are no Tech Spec actions are required for SRO evaluation for this failure.

**Event 5**: Component Cooling Water system leak requiring AOP-014 entry and manual makeup to maintain level. A CCW leak in the running pump suction header will develop. The leak will be within CCW Surge Tank makeup capability. The crew should identify the leak by observation of MCB indications for CCW Surge Tank level or MCB annunciators based on CCW Surge Tank low level. The crew should respond to the CCW Surge Tank level change and/or alarm and enter AOP-014, LOSS OF COMPONENT COOLING WATER. The RAB RO will be dispatched to investigate the leak. The crew will maintain CCW Surge Tank level in the normal operating range by opening the demin water make up valve 1DW-15, on the MCB. Shortly after being dispatched the leak will be identified as a leak in the suction header near the pump. The leak can be manually isolated by closing local isolation valves. The crew will then be required to start the standby 'B' CCW pump and secure the running 'A' CCW pump IAW OP-145. The SRO should evaluate TS 3.7.3.

PLANT SYSTEMS

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3 At least two component cooling water (CCW) pumps*, heat exchangers and essential flow paths shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With only one component cooling water flow path OPERABLE, restore at least two flow paths to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

*The breaker for CCW pump 1C-SAB shall not be racked into either power source (SA or SB) unless the breaker from the applicable CCW pump (1A-SA or 1B-SB) is racked out.

Rec'd 3/25/13

## SCENARIO SUMMARY: 2013 NRC EXAM SCENARIO 5 (continued)

**Event 6**: RCP "A" rising vibration requires manual Reactor trip and securing "A" RCP and associated PRZ spray valve after E-0 immediate actions are completed. During this event the 'A' RCP vibrations will begin to increase and over 3 minutes peak at 28 mils shaft. Note: the shaft vibration instrumentation reads up to 30 mils. The crew will respond to the 'A' RCP malfunction by either identifying rising vibrations or when ALB-010-3-1, RCP-A Trouble alarms. The crew should see the 'A' RCP vibration probe readings are increasing. The crew should enter AOP-018, Reactor Coolant Pump Abnormal Conditions and perform the immediate actions of checking any CSIP running (YES). Vibrations will continue to increase and exceed AOP-018 Attachment 1 RCP trip criteria of 20 mils shaft. Since the RCP is NOT operating within the trip limits and the Reactor is NOT tripped, the crew will have to Trip the Reactor, GO TO EOP-E-0, perform the immediate actions of E-0 and return to AOP-018 follow up actions of steps 5-8 when time permits. The SRO should address steps 5-8, stopping the affected RCP and shutting the associated PZR spray valve prior to the manual Reactor trip.

> **Note**: AOP-018 recent revisions now direct Tripping the Reactor prior to tripping a running RCP. ALL RCP's must be operating whenever the Reactor trip breakers are closed. Previously two loop power operation was allowed after securing one RCP if the initial power level was ≤49%.

The crew will then transition from EOP E-0 to ES-0.1, Reactor Trip Response. The Lead Examiner can allow the crew to stabilize the plant then insert the major event.

**Event 7**: **Major** – SBLOCA inside containment (100% severity). The major event is a SBLOCA (100% severity) on 'A' Loop. The crew should recognize a rapid decrease in Pressurizer level and RCS pressure. If the crew responds quickly to the event they may manually actuate a Safety Injection based on ES-0.1 foldout criteria of not being able to maintain Pressurizer level > 5% or RCS subcooling < 10°F. If they do not respond quickly an Automatic Safety Injection will occur. The crew will then transition from ES-0.1 back to E-0, Reactor Trip or Safety Injection. They will again carry out immediate actions of E-0.

**Event 8**: Failure of BIT outlet valve 1SI-4 to open requiring alternate high head injection flow path use. 1SI-4 will fail to automatically open with the Safety Injection signal and cannot be manually opened from the MCB switch. Additionally, 1SI-3 was under clearance and cannot be opened from the MCB due to control power being removed from the breaker. In order to obtain Safety Injection flow the crew will have to use the alternate high head injection flow path as directed by E-0 RNO actions. They should OPEN alternate high head Safety Injection to cold legs valve 1SI-52 SA and then identify Safety Injection flow exceeding 200 gpm.

**Event 9**: Manually trip 'B' and 'C' RCP when RCP trip criteria are met IAW E-0 foldout. Shortly after entering E-0, the crew should recognize that the RCS pressure is low enough to meet Foldout Criteria for securing all RCPs but there is no flow indicated on FI-943 (normal SI flow indication). The crew will have to establish SI flow by opening the alternate high head Safety Injection to cold legs valve 1SI-52 SA. After opening 1SI-52A adequate flow (> 200 gpm) will be indicated on FI-940 (alternate SI flow indication) to STOP the 'B' and 'C' RCP's.

Rec'd 3/25/13

**SCENARIO SUMMARY: 2013 NRC EXAM SCENARIO 5 (continued)**

**Event 10**: Failure of automatic Main Steam Line Isolation to occur when Containment pressure exceeds 3 psig. As the Small Break LOCA continues to flow RCS to the Containment the pressure in the Containment will continue to rise. An automatic Main Steam Isolation signal is generated when Containment pressure is $\geq$ 3.0 psig. The failure of this signal will require the crew to manually actuate Main Steam Line Isolation. The MCB switch will NOT function requiring the crew to shut ALL MSIVs.

The crew will transition from E-0, Reactor Trip or Safety Injection at step 30 when Containment pressure is checked and found to be NOT normal to E-1, Loss of Reactor or Secondary Coolant step 1. The crew will progress through E-1 based on crew performance they will reach a decision point at step 13.

They will transition from in E-1 to ES-1.2, Post LOCA Cooldown and Depressurization, based on RCS pressure > 230 psig and RHR HX header flow < 1000 gpm.

While in ES-1.2 based on RCS cooldown rate exceeding 100°F/HR they will have to wait prior to reducing RCS temperature further.

The scenario ends when the crew has determined that the 100°F/HR cooldown rate has been exceed.



Rec'd 3/25/15

## CRITICAL TASK JUSTIFICATION:

1. With RCS pressure < 1400 psig, establish SI flow of > 200 gpm using alternate high head safety injection to cold legs prior to securing RCPs

   Securing RCPs during a SB LOCA event will prevent depleting the RCS to a critical inventory by pumping more mass through the break than would occur if the RCP operation were ceased. (Critical inventory is defined as the amount of inventory remaining in the RCS when the break completely uncovers and the break flow changes from a mixture of liquid and steam to all steam.) The PRZ Steam Space LOCA event in this scenario is a SB LOCA that requires the RCPs to be secured when E-0 foldout conditions are met. IF the crew continues to allow the RCPs to operate due to lack of establishment of SI flow of > 200 gpm then RCS inventory will continue to deplete. Manually tripping the RCPs before depletion below the critical inventory conservatively ensures that Peak Clad Temperature remains below 2200°F. NOT establishing SI flow prior to RCS reaching 230 psig (RHR injection pressure) was chosen as an applicable plant parameter to use for grading criteria for the task of securing RCPs.

2. Manually actuate Main Steam Line Isolation prior to Containment pressure exceeding 10 psig

   Containment pressure increasing to ≥ 3 psig should have caused an automatic Main Steam Line Isolation to occur. Since it has not the crew should manually actuate this isolation to prevent the potential release of contamination outside of Containment from the event. At 10 psig a Containment Phase B signal is actuated which will cause the remaining Containment Isolation valves to automatically close thus preventing a release of potential contamination. Since the 10 psig Containment Phase B isolation signal is the last signal to automatically isolate potential Containment release pathways this pressure was chosen as an applicable plant parameter to use for grading criteria for the task of Manual Steam Line Isolation.



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